~~SPLB~~ SPSB
cc: Suzie (memo only)

NEI

NUCLEAR ENERGY INSTITUTE

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March 31, 2004

Ms. Suzanne C. Black
Director, Division of Safety Assessment and Analysis
U.S. Nuclear Regulatory Commission
Mail Stop O 10 A1
Washington, DC 20555-0001

SUBJECT: *Three Dimensional Rod Ejection Accident Peak Fuel Enthalpy Analysis Methodology*, EPRI Technical Report 1003385, November 2002

PROJECT 689

Dear Ms. Black:

I have enclosed two copies of the subject topical report for NRC review and approval. This report was developed by the EPRI Fuel Reliability Program (formerly Robust Fuel Program) to describe a rod ejection analysis (REA) methodology based upon a three dimensional kinetics code. This topical report is limited to the calculation of fuel enthalpy for the pressurized water reactor REA.

This report can be used by PWR licensees and fuel vendors to apply the latest computational techniques in their licensing basis accident analyses. We seek NRC staff review and endorsement as a means of exchanging information that is intended to support generic regulatory improvements. Therefore, we believe an exemption from any review fees is warranted based upon the criteria in footnote 4 of 10 CFR 170.21.

If you need any further information, please contact me at 202-739-8080; am@nei.org.

Sincerely,

Alexander Marion

Alexander Marion

Enclosure

c: Mr. J. S. Wermiel, USNRC

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EPRI

Three-Dimensional Rod Ejection Accident Peak Fuel Enthalpy Analysis Methodology

Technical Report

Three-Dimensional Rod Ejection Accident Peak Fuel Enthalpy Analysis Methodology

1003385

Final Report, November 2002

EPRI Project Manager
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Nuclear Management Company

Framatome Advanced Nuclear Power

Westinghouse Electric Corporation

Dominion Generation

Electricite de France

Duke Engineering & Services

ORDERING INFORMATION

Requests for copies of this report should be directed to EPRI Orders and Conferences, 1355 Willow Way, Suite 278, Concord, CA 94520, (800) 313-3774, press 2 or internally x5379, (925) 609-9169, (925) 609-1310 (fax).

CITATIONS

This report describes research sponsored by EPRI under the Robust Fuel Program. The work was conducted by the members of the REA 3D Analysis Focus Group:

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A. F. Dias, Utility Resource Associates Corporation (Appendix A)

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This report describes research sponsored by EPRI and Duke Energy Company.

The report is a corporate document that should be cited in the literature in the following manner:

Three-Dimensional Rod Ejection Accident Peak Fuel Enthalpy Analysis Methodology, EPRI, Palo Alto, CA, and Duke Energy Company, Charlotte, NC: 2002. 1003385.

REPORT SUMMARY

A consistent methodology for analyzing the effects of a pressurized water reactor (PWR) control rod ejection accident (REA) on fuel has been developed. The methodology is based on state-of-the-art three-dimensional (3D) neutron kinetics codes together with probability-based assumptions about core initial conditions. In licensing calculations, this methodology is expected to provide a more precise estimate of the number of fuel rods (if any) that can reach enthalpy levels high enough to result in fuel failures during an REA.

Background

Laboratory experiments on high-burnup light water reactor (LWR) fuel have raised concerns about the adequacy of current regulatory criteria for fuel failure and core coolability during a reactivity-initiated accident (such as an REA in PWRs). As a result, the industry, through the EPRI-managed Robust Fuel Program, proposed a reduction in the criteria at higher burnups, consistent with measured changes in fuel mechanical properties (EPRI report 1002865). As the criteria are reduced, however, the question of whether high-burnup fuel can reach high enough enthalpy levels to result in failures during an REA event becomes more important. Typical Nuclear Regulatory Commission (NRC) approved REA analysis methodologies have used overly conservative analytical approaches based on zero or one-dimensional kinetics models coupled with static neutronics to capture the 3D core power distribution effects. Such methodologies greatly overestimate fuel response since they must expand what, in reality, is a localized phenomenon on a core-wide basis. Recently, there have been significant advances in computational capabilities as well as in developing 3D kinetics methods to the point that a more realistic analysis of this event is now possible.

Objectives

- To develop a three-dimensional neutron kinetics methodology for analyzing rod ejection accidents in PWRs.
- To develop a consistent approach to licensing calculations that will be required for introducing burnup extensions or new fuel designs.

Approach

Issues relating to fuel response to transient events or to accidents are addressed under Working Group 2 of the Robust Fuel Program. Under the auspices of this group, an REA Analysis Focus Group was established and charged with developing a methodology based on the new 3D kinetics codes. The group's intent was to achieve more realistic results that can accommodate anticipated reductions in failure criteria. The focus group consisted of members from utilities and fuel vendors that have licensed traditional conservative analytical methods. The focus group examined current (zero- and one-dimensional) analysis methods and modeling assumptions that

are being made for licensing calculations. The group then evaluated how these assumptions would have to be changed when 3D methods are used and proposed a common set of assumptions.

Results

Focus group members developed a common methodology that is independent of individual member's currently licensed methods or codes. The proposed methodology is based on 3D kinetics codes as well as probability-based arguments to define the set of core initial conditions to be analyzed. The methodology also provides a two-tiered approach to treating analytical uncertainties as a user option. Key parameters with the most significant impact on analytic results are identified and an option for treating uncertainties in these parameters either statistically or deterministically is provided. Demonstrations of how the methodology would be used by different organizations for different types of PWRs are provided in appendices.

EPRI Perspective

This work is part of an industry-wide effort under the Robust Fuel Program aimed at extending fuel rod average burnup levels above currently licensed limits. The methodology described here will be submitted as a topical report to the NRC for acceptance on an industry-wide basis. It is anticipated that each organization intending to implement this common methodology will be required to submit for NRC review a separate REA methodology report. Such a report will provide details of their implementation relating to use of their specific codes and providing justification for any deviations. This two-step approach to licensing the new REA analysis technologies is intended to reduce each organization's effort as well as facilitate the NRC review process.

3D kinetics methodologies are considered necessary for demonstrating that high-burnup fuel will remain well below the revised failure criteria proposed in the topical EPRI report 1002865.

Keywords

Reactivity-initiated accident
PWR rod ejection analysis
Safety analysis
LWR fuel
Burnup extension
Robust fuel

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EXECUTIVE SUMMARY

The analysis of postulated rod ejection accidents (REA) is a part of the licensing basis accident analyses required for pressurized water reactors (PWR). The REA analysis simulates the reactor coolant system, core, fuel rod, and fuel pellet response to the transient induced by the rapid positive reactivity insertion. One part of this analysis is to calculate the fuel thermal response during the accident, which is typically reported as the peak radial average enthalpy (cal/gm) of the hottest fuel pellet. These results are then compared to the regulatory limits on peak enthalpy to ensure that the fuel design meets safety requirements.

The current regulatory limit on core coolability for PWR rod ejection accidents is 280 cal/gm. Recent experiments have shown that the current regulatory limit may be set too high, especially at higher fuel burnup values. This has led to a research program by the U.S. Nuclear Regulatory Commission (NRC), and the expectation that the 280 cal/gm regulatory limit will be lowered in the future.

Current vendor and licensee NRC-approved REA analysis methodologies use a variety of conservative analytical methods. These methodologies typically include point kinetics or one-dimensional kinetics models, and syntheses of these kinetics models with static neutronics models to capture the three-dimensional (3D) core power distribution effects. The traditional approach to include conservative values for all initial and boundary conditions is used, and contributes to the overall conservatism of the REA analysis. Recently there have been significant advances in analysis methods and computational speed, particularly in the use of 3D kinetics methods and high-powered workstations. These methods have been demonstrated to more realistically predict reactor power transients, especially in the case of a localized power excursion as occurs in the REA. The industry and NRC have increased the application of probability-based approaches in the resolution of various issues associated with nuclear power plant safety. Inclusion of a probability-based approach to the REA analysis has merit based on the recognized low probability of the REA event.

The Electric Power Research Institute (EPRI) has chartered a Robust Fuel Program to advance the design of nuclear fuel assemblies to meet the future needs of the industry. One element of the Robust Fuel Program, Working Group #2, is addressing the fuel response to transients and accidents. Under the auspices of Working Group #2, the REA 3D Analysis Focus Group has been charged with developing a REA analysis methodology based on a 3D kinetics code, with the intent of achieving less conservative analysis results that can accommodate a future reduction in the 280 cal/gm regulatory limit.

The REA 3D Analysis Focus Group membership consists of licensees and fuel vendors that have developed and licensed traditional conservative REA analytical methods. To meet the future reduction in the REA cal/gm regulatory limit, these existing methods are likely to need revision including a shift to a 3D kinetics-based method. To lessen the burden on each organization to

develop and license upgraded REA methods, the Focus Group members have come together to develop a common REA 3D methodology that is independent of individual members' licensed methods or codes. This report describes the proposed common methodology. The report will be submitted to the NRC for generic review and approval. Each organization intending to implement this NRC-approved methodology would then be required to submit for NRC review a separate REA methodology report that would specify the specific computer codes and details of the application of the generic methodology, and justify any deviations from the methodology. This two-step approach to licensing upgraded REA analysis methodologies is intended to lessen the development and licensing effort at each of the organizations, as well as to facilitate the NRC review process at less cost.

The REA analysis methodology described in this report is limited to the calculation of fuel enthalpy for the PWR REA. There are other aspects of REAs, such as the calculation of departure from nucleate boiling and the pressurization of the reactor coolant system, that are not addressed here. The methodology uses 3D kinetics methods. It uses probability-based arguments to define the set of core initial conditions to be analyzed. It also provides a two-tiered treatment of the analytical uncertainties, as an option to the user. In this approach, the user may choose to treat analytical uncertainties in a deterministic manner, or combine the uncertainties statistically. The significant analytical variables were found to be the worth of the ejected rod in dollars, the fuel Doppler temperature coefficient, the moderator temperature coefficient, and the nuclear peaking factor uncertainty. Uncertainties in these variables are treated either statistically or deterministically.

Appendices A through D of the report provide demonstration analyses using analytical methods from different organizations. Analyses of two-loop and four-loop Westinghouse reactors, a B&W reactor, and a Westinghouse/Combustion Engineering reactor are presented. These demonstration analyses are not intended to be bounding results for PWR designs or reload core designs. For example, the uncertainty values used in the demonstration analyses are for demonstration purposes only. The purpose of the demonstration analyses is solely to demonstrate the 3D REA analysis methodology presented in the body of the report.

Appendix E of the report provides proposed changes to the two primary regulatory documents that affect PWR REA analysis, Regulatory Guide 1.77 and Section 15.4.8 of the Standard Review Plan, NUREG-0800. The intent of providing this information is to highlight how the new analysis methodology relates to the current regulatory requirements and to facilitate NRC review of the new methodology.

The proposed REA analysis methodology based on 3D kinetics will result in improved analysis results that are expected to accommodate a future reduction in the REA cal/gm regulatory limit, while maintaining a level of conservatism appropriate for this low probability initiating event.

1
INTRODUCTION

1.1 Background

The licensing of light water reactors requires analysis of a set of design basis transients and accidents. One class of these accidents is the reactivity insertion accident (RIA), where the issues are the ability of the nuclear fuel to maintain its integrity in a fast power excursion, and the ability of the reactor coolant system to accommodate any resulting pressurization transient. The main concern is that in a fast power excursion enough energy could be deposited in the fuel rod in a very short period of time to cause cladding failure and result in a loss of coolable geometry.

In a pressurized water reactor (PWR), the most severe of this class of accidents is considered to be the rod ejection accident (REA). This accident is initiated by a sudden failure of a control rod drive housing in the reactor vessel head, resulting in the ejection of a control rod assembly due to the pressure differential. The resulting insertion of positive reactivity into the core region around the ejected control rod assembly can cause the deposition of a large amount of energy in the fuel within a few hundred milliseconds.

If the core is initially subcritical or at low power, and the rod ejection event does not cause the reactor to go prompt-critical, there is no risk of core and plant damage. If the reactor goes prompt-critical, irrespective of the initial condition of the core, the response will be characterized by a rapid power excursion. The magnitude of the increase in neutron flux will of course depend on the amount of positive reactivity inserted by the rod ejection event, and the delayed neutron fraction. The localized rapid increase in neutron flux will cause a power excursion in the fuel rods in that region of the core. The associated energy deposition in the fuel rods will cause a temperature increase. As the fuel temperature increases, there will be a negative reactivity feedback due to the Doppler broadening of the U-238 absorption cross-section. This Doppler effect provides the most significant negative reactivity feedback to counter the positive reactivity insertion. It provides sufficient negative reactivity to overcome the positive reactivity insertion caused by the ejected control rod, and terminates the power excursion. The moderator temperature responds to the power increase with a delay due to the fuel thermal time constant. Direct heating of the moderator also occurs. A moderator temperature increase provides additional negative reactivity feedback if the moderator temperature coefficient is negative. If the moderator temperature coefficient is positive, additional positive reactivity will be inserted. Termination of the event occurs when a reactor trip setpoint is reached, the control rods drop into the core, and the power and temperature decrease back to initial levels or lower.

In a typical REA initiated from a hot zero-power critical condition, a rapid power increase, associated with prompt criticality, occurs within a few hundred milliseconds. The fuel temperature increases at about the same rate. The negative fuel temperature coefficient limits the power excursion, and then power begins decreasing very quickly. The rapid decrease in power stabilizes the fuel temperature. The cladding temperature lags the fuel temperature due to the thermal time constant of the fuel rod.

The main issues are whether the fuel rod will withstand the temperature increase without experiencing cladding failure and the associated release of radioactivity, and whether fuel failure will lead to a loss of core coolability. Experimental data from RIA experiments in test reactors have been historically used to set regulatory limits on the amount of energy deposition in the fuel rod that could cause cladding failure and loss of coolability. The energy deposition is characterized in terms of the increase in the peak average enthalpy of the hottest fuel pellet. The key analysis result used to assess fuel integrity in the event of a REA in this report is the increase in peak average fuel pellet enthalpy (Δcal/gm). This Δcal/gm result is then added to the initial enthalpy to obtain the peak cal/gm result.

1.2 Accident Mitigation

PWRs have several design features that are relevant for mitigating rod ejection accidents. These include the mechanical design features, which prevent the accident from occurring in the first place, the general design criteria for the reactor which limit the reactivity associated with an ejected control rod, the inherent negative fuel Doppler temperature coefficient, and other design and protection features to mitigate the accident if it happens.

In terms of mechanical design, the control rod drive housing is designed and tested to the high standards of the ASME code. Hydro-testing is conducted after installation.

The control rods and core loading patterns are designed and operating limits are specified to limit the reactivity worth of an ejected rod to acceptable values. The reactor is normally operated with the rods inserted only partially at full power. Thus the amount of reactivity that could be inserted in a postulated REA at full power, which is the initial condition during most of the time that the reactor is critical, is minimized. At lower power levels the control rod insertion limits are based in part on limiting the ejected rod worth. The design, location, and grouping of the control rod banks are also selected to limit the reactivity addition.

The position of the control rod assemblies is continuously indicated in the control room. If a bank of assemblies approaches its insertion limit, or if an assembly deviates from its bank, an alarm will typically result and corrective actions will be taken.

The reactor protection system includes an automatic reactor trip to mitigate the REA and put the reactor in a subcritical state. The reactor trip keeps the core subcritical after the excursion is initially mitigated by the fuel Doppler feedback.

1.3 Regulatory Criteria

Regulatory criteria for assessing REA events are based on meeting the intent of General Design Criterion, GDC-28 - Reactivity Limits, of 10CFR50, Appendix A. Regulatory limits for analysis, as well as guidance on how to perform a conservative analysis suitable for licensing purposes, can be found in NRC Regulatory Guide 1.77 [1] and the NRC Standard Review Plan (NUREG-0800), Section 15.4.8 [2]. The regulatory concern is that in postulated REA events, there could be sufficient energy deposition in the fuel to cause rupture of the fuel pins and rapid fragmentation and dispersal of fuel material into the coolant. This would result in rapid heat transfer to the water from the finely dispersed fuel particles. Conversion of this energy to mechanical energy could conceivably disarrange the reactor core or breach the primary system.

Acceptance criteria for REA events are based on meeting GDC-28 requirements as it relates to the effects of postulated reactivity accidents neither resulting in damage to the reactor coolant pressure boundary greater than limited local yielding, nor causing sufficient damage to impair significantly the capacity to cool the core [2]. Specific acceptance criteria used for evaluating postulated REA events are [2]:

1. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components".

3. The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling (DNB) condition is an input to the radiological evaluation conducted to meet the radiological criteria given in Regulatory Guide 1.77, Appendix B [1].

These regulatory limits have been used for licensing and considered satisfactory for PWRs for many years since their publication in the regulations. However, beginning in 1993, RIA experiments at the CABRI test reactor in France indicated that cladding failure and fuel dispersal could occur for high burnup fuel at very low values of energy deposition, significantly lower than the 280 cal/gm regulatory limit. Other RIA test results were subsequently obtained at other test reactors in Japan and Russia. These new research results have caused renewed interest in this regulatory issue, and have led to discussions on whether the regulatory limits should be lowered.

In August 1994, the U.S. Nuclear Regulatory Commission issued Information Notice IN-94-64, notifying U.S. utilities about the recent experimental data, particularly from the CABRI test reactor in France. The NRC also assessed the safety significance of the test results with respect to operation of current reactors. It was concluded that the rod ejection event, with the current regulatory limits, would not have a significant impact on public health and safety because of the low probability of the event and the mitigation measures in place.

Since then, there has been considerable international research and investigation into reactivity insertion accidents. The NRC commissioned a study to identify and rank phenomena that could occur following a PWR REA. This study is focused on high burnup fuel. The study is expected to provide guidance for further research efforts. Concurrently, there is an ongoing effort to determine a revised set of acceptance limits for energy deposition in fuel rods as a consequence of this type of an accident. It appears likely that the acceptance limits will be lowered, and/or made burnup dependent.

The nuclear industry, through the Nuclear Energy Institute, has proposed new PWR REA cladding failure and core coolability acceptance limits for REA for high burnup fuel. The new proposed limits were developed by the Electric Power Research Institute (EPRI) on behalf of the industry, and are detailed in "Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel and Core Coolability Criteria [16]". This report was submitted to the NRC in June 2002.

1.4 Analytical Methods

NRC-approved REA analysis methodologies generally employ point kinetics or one-dimensional kinetics approaches in combination with static 3D neutronics models to calculate the transient core power distribution. It is more accurate and realistic to use 3D space-time kinetics models to calculate the power excursion and the power distribution. This is especially the case in the REA where the issue is one of calculating the local power in a small region of the core. The 3D methods essentially ensure that the calculation of the core power response utilizes the actual spatial and temporal distribution of nuclear parameters that affect the detailed response of the core and fuel during the REA. This avoids the averaging and approximations that are necessary with the less sophisticated simulation approaches.

A comparison of 3D methods versus 1D methods used by Westinghouse have been reported by Ray, et al. [3], and Risher, et al. [4]. The case chosen was representative of a limiting case, in the sense that the initial conditions represented hot zero power conditions, with rods inserted at the insertion limits typical for this condition. All other input parameters were assumed similar in the comparison analysis and were typical of assumptions used in current licensing analyses. The results showed that the peak fuel enthalpy reached during the transient by the peak power fuel rod, as calculated by the 3D model, was about 50% less than that predicted by the 1D model (about 100 cal/gm for the 3D analysis compared to about 200 cal/gm for the 1D analysis). Risher et al. [4] also showed that with more realistic input assumptions (compared to the conservative licensing assumptions), the peak fuel enthalpy would be further reduced by about 50%.

Dias, et al. [5] used 3D kinetics methods developed at EPRI to study the PWR REA. The study looked at a realistic set of input assumptions, for initial conditions corresponding to hot zero power (HZP) and hot full power (HFP). The limiting case, which was the HZP case, did not indicate a prompt jump when the accident was initiated, which then resulted in a gradual power increase until the accident was terminated by reactor trip. This case used a reactivity insertion of $0.88, which was considered realistic for the core studied. Typical reactivity insertion values assumed in some licensing analyses would be about $1.30. The limiting case was then re-analyzed with this reactivity insertion. The results this time did indicate a prompt power excursion, but the fuel enthalpy did not increase beyond 40 cal/gm. These results are consistent with the results reported by Risher, et al. [4].

It appears from these studies that:

1. Use of 3D kinetics models, even with conservative licensing-type input assumptions, would result in a significant reduction in the calculated peak fuel enthalpy. One study has shown this reduction to be about 50%.

2. Use of realistic input assumptions, combined with the use of 3D kinetics models, would also produce a significantly reduced calculation of the peak fuel enthalpy. One study has shown this effect to be about 50%.

1.5 Probability-Based Assumptions

Traditional transient and accident analysis methodologies have employed conservative scenarios and input parameter assumptions in conducting the analyses. In general, as a transient or accident scenario undergoes research, analytical methods become more refined, and additional data becomes available, it is justifiable to perform a less conservative analysis that still meets the general design criteria for nuclear plants. With the work that has been done in recent years by the industry and regulators, both domestically and internationally, on RIA testing and REA analysis, it is possible to more narrowly define the analysis scope than was done before. In particular, probabilistic-based arguments can be used to define and limit the scope of REA analyses.

1.6 Uncertainty Analysis

Given the low probability of an REA event, it is also reasonable to treat the effects of uncertainties in key variables statistically. The NRC sponsored PWR REA PIRT project has identified the important variables in the analysis. Recent work on REA (Ramos, et a. [6], Diamond, et al. [7]) have shown that it is possible to perform a conservative REA analysis without bounding all variables and combining them in a deterministic fashion. Hence, the analysis ought to be able to use statistical methods to combine uncertainties such that the overall uncertainty in the calculated fuel energy deposition is bounded at a reasonably high probability level. This approach has been applied successfully in other types of transient and accident analyses.

1.7 Scope of Methodology

U.S. nuclear plants desire to achieve higher burnup limits in fuel rods. This would allow longer fuel operating cycles which offers economic advantages. Recent tests conducted in France, Japan, and Russia have raised questions about whether the current regulatory limits for reactivity insertion accidents are sufficient. The NRC is actively engaged in re-examining these regulatory limits. At the same time, it has been well established that the current methods used for licensing analysis in this area are overly conservative. Considering these issues, EPRI Robust Fuel Program, Working Group #2 - Response to Transients, formed a REA 3D Analysis Focus Group in 1999. The Focus Group has representatives from fuel vendors and licensees. The members of the Focus Group are Westinghouse, Framatome Advanced Nuclear Power, Electricite de France, Dominion Generation, the Nuclear Management Company, and Duke Power.

The mission of the Focus Group is to develop a standard method for performing the PWR REA analysis using 3D core modeling which will be licensable in the U. S. The objective of this new methodology is to remove excess conservatism by providing a more realistic analysis for a low probability event. The methodology starts from the specification of the initial core and plant conditions, continues with all modeling assumptions (in particular the level of conservatism at each step of the methodology), and extends into the calculations necessary to show compliance with the regulatory limit.

The Focus Group has completed this assignment. This report describes a standard PWR REA 3D analysis methodology that is intended to be a template for upgrading the current REA analysis methods. The new methodology should gain sufficient margin to meet the expected future reduction in the REA peak fuel enthalpy regulatory limit.

2
OBJECTIVE OF NEW METHODOLOGY

The objective of this new methodology is to provide a generic PWR REA 3D analysis guideline for the calculation of the fuel enthalpy increase and peak fuel enthalpy. This report is intended to be a generic licensing document. The methodology is considered applicable to all current U.S. PWR core and fuel designs.

The methodology in this report is intentionally independent of any specific computer codes and can be used by any organization. This allows use of the methodology by various organizations. It is expected that each organization will develop appropriate computer codes for use with this methodology and will license the codes with the NRC as appropriate. This report is intended to serve as a reference for organizations performing REA analysis and to serve as a generically applicable standard method. This approach is expected to result in less licensing effort by each organization, and also facilitate NRC review.

The report describes a methodology for performing the PWR rod ejection analysis using 3D kinetics and probability-based assumptions for the purposes of meeting the future lower cal/gm acceptance limits that will be proposed by the NRC for high burnup fuel. The methodology incorporates less conservatism than has traditionally been reflected in the assumptions employed in the rod ejection analyses that have been licensed with the NRC and that are presented in Chapter 15 of current FSARs. The justification for lowering the conservatism is based on the improved accuracy of the 3D kinetics core model, the overall low probability of the REA event, and the use of a probability-based approach to justify some of the initial and boundary conditions.

The methodology starts with 3D core kinetics analysis of the REA event, and continues through to the calculation of the Δcal/gm and peak cal/gm in the hot fuel pellet. A two-tiered approach is proposed for the treatment of analytical uncertainties. The first tier will be a deterministic approach with less conservatism than in the past. The second tier will use a statistical combination of uncertainties for the key physics parameters only, which will provide more margin to the acceptance limits. The second tier will likely only be used by organizations that need additional margin.

The methodology has been tested by several organization running demonstration analyses. These analyses have used different computer codes for the physics and fuel rod response calculations. The report includes the results of these demonstration cases.

One recognized issue is that the future lower cal/gm acceptance limits (which may be a function of burnup) have not been finalized. It is expected that this proposed methodology will be successful in meeting the new acceptance criteria.

The report includes proposed revisions to Reg. Guide 1.77 [1] and Standard Review Plan (SRP) 15.4.8 [2]. The purpose of this is to facilitate NRC review and understanding of how the proposed REA methodology would affect the current regulatory guidance.

It is recognized that cladding integrity is affected by other issues in addition to the fuel enthalpy increase, such as the calculation of departure from nucleate boiling. There are also other issues associated with the rod ejection event, such as reactor coolant system over-pressurization. Those issues are not addressed by the methodology described in this report. The focus of this methodology is only to calculate the fuel enthalpy increase and peak enthalpy following a PWR REA.

3
DESCRIPTION OF METHOD

3.1 Introduction

This section describes the proposed methodology for PWR REA analysis, along with technical justification for the methodology and assumptions made. The focus of the methodology is on the fuel enthalpy calculation associated with the REA event. The computer codes used for the analysis may be used for other aspects of the REA analysis, such as DNBR calculation, but this methodology only deals with the application of these codes to the calculation of energy deposition in the fuel rod and the associated enthalpy increase.

The methodology is not specific to any computer code or suite of codes in use by various organizations. It assumes that the computer codes have been sufficiently benchmarked and have been approved by the NRC as necessary. The methodology is not constrained to any particular core designs, and is not limited to a particular value of fuel burnup. The intent of this methodology is to take advantage of 3D kinetics and a probability-based approach to define the scope of core initial conditions to provide more analytical margin than exists with currently licensed methods.

The methodology includes guidance for a full scope analysis for the first application to a particular reactor and core design, and an abbreviated scope of work required to assess REA events for core reloads.

Individual organizations will need to submit the plant-specific elements associated with this generic methodology to the NRC for review and approval. It is expected that this report will serve as a reference for these organizations. Use of this report as a reference is expected to result in a reduced scope of development and licensing work by each organization, as well as facilitate the NRC review.

3.2 Significant Variables and Uncertainty Analysis

As part of the regulatory and industry followup to the CABRI test results, the NRC convened a panel of experts to develop a phenomena identification and ranking table (PIRT) for the PWR REA. A primary purpose of the study was to identify the experimental and analytical uncertainties associated with this event to provide guidance for further research. Results of the study have been issued [8]. The study focused on the effects of high burnup fuel. In the study, the rod ejection event considered was initiated from hot zero power conditions. The analysis of the transient power distribution and the calculation of the fuel temperature increase were considered separately. In the calculation of pin power, the variables that were determined to be

the most important were the ejected rod worth, fuel temperature (Doppler) feedback, delayed neutron fraction, and fuel cycle design. In the calculation of fuel enthalpy increase, the heat capacities of fuel and cladding were the only parameters considered important. The study also notes that these variables are well-known variables, indicating that the uncertainty in these variables can be bounded at a high confidence level.

Diamond, et al. [7] studied the uncertainty in fuel enthalpy calculation for a PWR REA event. The analysis used engineering judgement and insights from point kinetics methods to determine that the significant variables in the analysis would be the ejected rod worth, the delayed neutron fraction, the fuel Doppler reactivity coefficient, and the fuel specific heat. Sensitivity studies were carried out using a best-estimate 3D physics code to determine the impact of these variables on the fuel enthalpy. It was determined that the uncertainty in the fuel enthalpy would be determined primarily by the uncertainties in the ejected rod worth and the delayed neutron fraction. Based on the sensitivity studies, Diamond, et al. [7] also concluded that only events with rod worth in excess of $1 (essentially prompt-critical events) need be considered in the analysis.

The recent studies noted above, conducted with 3D kinetics methods, have essentially identified only a few significant parameters in the calculation of fuel enthalpy increase for a PWR REA event. These variables are:

- Time-in-cycle
- Core peaking factor
- Ejected rod worth
- Delayed neutron fraction
- Fuel Doppler temperature coefficient
- Moderator temperature coefficient

Sensitivity studies have also been conducted as a part of developing currently used licensing analysis methods. This experience provides knowledge on the variation of the fuel enthalpy increase with other, less important variables. The current licensing analyses are generally conducted with a combination of nominal and bounding values for these less important variables.

The approach used by the Focus Group in developing this methodology benefited from knowledge of the REA event from current licensing analyses and the studies referenced above. In addition, the Focus Group performed additional sensitivity analyses to obtain results where previous results or knowledge were lacking. Based on this body of knowledge, Table 3-1 shows a list of parameters used in typical REA analysis, the judgement of the Focus Group regarding the sensitivity of the REA fuel enthalpy increase to each parameter, and the recommended analytical assumption for each parameter. The fuel heat capacity is judged to be sufficiently well-established that values from the literature can be used directly. The hot pin peaking factor is also judged to be well-established within each organization's analytical methods for predicting core power distributions. The ejected rod worth in dollars, the fuel Doppler temperature coefficient, and the moderator temperature coefficient are treated as uncertainty variables. In this approach, the uncertainty associated with the calculation of the transient power distribution

(code uncertainty) is addressed by including the standard uncertainty value for the core peaking factor as an uncertainty variable, and by crediting the conservatism in the transient core power distribution resulting from the uncertainty applied to the ejected rod worth.

The uncertainty analysis employs a combination of deterministic and statistical treatments to produce fuel enthalpy calculations that are bounded at a high probability level. This approach treats less important variables by assuming them to be at either nominal or bounding conditions, as done in current licensing analyses. Initial conditions that are not ruled out by probability-based arguments are treated explicitly. Uncertainties in the other significant variables (ejected rod worth in dollars, fuel Doppler temperature coefficient, moderator temperature coefficient, core peaking factor) may be treated deterministically by applying all uncertainties simultaneously, or by using a statistical combination.

3.3 Overview of Methodology

The REA analysis methodology starts with identifying the scope of core initial conditions to be analyzed. These initial conditions must lead to a significant Δcal/gm result. The probability-based approach is used to identify which off-normal core conditions must be considered. For the cases to be analyzed the initial conditions in the core and the fuel rod are then established. The 3D kinetics code and the hot rod code are then initialized to those conditions. The 3D kinetics model REA analyses are then run. The fuel rod model, either the model integral to the 3D code or a stand-alone hot rod model code, is then used to calculate the enthalpy increase in the hot fuel pellet as a function of time, and the peak enthalpy value. This is the key result of the calculation. This method can be done deterministically, with all of the uncertainties applied, or using the statistical combination approach. The methodology is described in greater detail as follows:

Step 1 – Identification of REA Scenarios with Potential for Large Δcal/gm Result

Core initial conditions for which a significant cal/gm increase can occur are identified. This is basically limited to core conditions for which there is the potential for the ejected rod worth to exceed one dollar of reactivity. Typically these core conditions will require some control rods to be substantially inserted and the reactor at or near a critical condition. Most likely, the hot zero power (HZP) initial condition will be the most limiting. Fuel burnup needs to be treated as a variable. Thus the analysis would need to consider beginning-of-life (BOL) as well as end-of-life (EOL) fuel rod initial conditions.

Step 2 – Identification of Off-Normal Core Initial Conditions

Technical specifications and core operating limit reports typically allow operation with deviations from nominal operating conditions. These off-normal conditions include:

- Core tilts
- Mis-positioned control rods
- Power distribution effects
- Adverse xenon distributions
- Long term operation with control rods inserted

In addition, intermediate power levels occur during startups and shutdowns, and can also occur due to equipment problems or planned operation at less than full power. All such core initial conditions that have the potential for worsening the core response to a rod ejection accident are to be identified.

Step 3 – Determine the Probability of Off-Normal Core Conditions

The rod ejection accident is recognized as a very low frequency event. For a rod ejection to occur during off-normal core initial conditions is even less probable. Below a frequency of 1.0E-07/RY, an initiating event does not warrant consideration as a design basis event from a probability-based perspective. Justification for this approach and the details of performing this probabilistic assessment are described in Chapter 4 of this report. Off-normal core initial conditions that cannot be screened out must be addressed in the rod ejection analysis.

Step 4 – Determine Uncertainty Values for Key Rod Ejection Analysis Parameters

Based on industry experience in analyzing rod ejection accidents with three-dimensional transient neutronics codes, the following core physics parameters (derived from three-dimensional modeling) have been identified as the key parameters that warrant consideration of an uncertainty allowance in the methodology:

- Ejected rod worth divided by beta-effective (ERWS)
- Doppler coefficient
- Moderator coefficient
- Core peaking factor (F_Q)

The moderator coefficient has a smaller effect relative to the other key parameters. However, it is labeled as a key parameter in the methodology to ensure that the presence of a positive moderator coefficient is addressed. Uncertainty values for these parameters over the range of core conditions that are valid for a particular rod ejection analysis core initial condition must be established. The basis for choosing these variables as the significant variables in this analysis is discussed in Section 3.2.

Step 5A – Conservative Deterministic Analysis Including Uncertainties

For the probable core initial conditions including off-normal considerations resulting from Steps 1-3, perform 3-D rod ejection analyses including the uncertainty values for the key parameters identified in Step 4. If the ERWS value with uncertainty is less than what is necessary to produce a significant Δcal/gm result, then that case can be screened out. The methods used for the 3D analysis need to be sufficiently robust for the purpose. Chapter 5 of this report further discusses requirements for the 3D kinetics method. Chapter 6 of this report discusses the hot fuel rod analysis method. It is expected that a simple fuel rod model can be used within the 3D core kinetics calculation provided that this approach is clearly conservative or has been justified by appropriate benchmarking. Organizations may also choose to use a separate more detailed hot fuel rod model to calculate the fuel enthalpy increase with boundary conditions from the 3D kinetics calculation.

The cases to be evaluated should include critical hot zero power initial conditions at beginning-of-cycle (BOC) and end-of-cycle (EOC). A spectrum of ejected rod locations is to be evaluated to ensure that the limiting cases are identified. The maximum Δcal/gm result and the peak enthalpy value are then determined based on the simulated fuel rod thermal response or based on deposited energy. The fuel rod model may be assumed to be adiabatic. If heat transfer from the fuel pellet is modeled, it needs to be justified.

The results of the analysis are then compared to the acceptance limits. If the acceptance limits are not exceeded, then the methodology is complete. If the acceptance limits are exceeded, then one of the following options can be taken: a) perform a hot rod analysis as a function of burnup to take advantage of more detailed modeling of the fuel rod transient response, b) re-design the core to obtain more favorable rod ejection analysis results, c) change allowable operating conditions, i.e., the rod insertion limit, or d) use the statistical methodology approach of Step 5B.

Step 5B –Statistical Combination of Uncertainties Analysis

For the probable core initial conditions including off-normal considerations resulting from Steps 1-3, perform a best-estimate rod ejection analysis without uncertainties. This is expected to include critical hot zero power initial conditions at BOC and EOC. A spectrum of ejected rods is to be analyzed to ensure that the limiting cases as a function of burnup are obtained. The best estimate Δcal/gm as a function of burnup is then determined for each case. The limiting cases, which will be referred to as the reference cases, are then determined from the initial case spectrum. For the reference case(s), sensitivity analyses that vary one key parameter at a time by including the uncertainty allowance are run. These sensitivity analyses are performed for the ejected rod worth in dollars, the fuel Doppler coefficient, and the moderator coefficient. The differences in the Δcal/gm for each sensitivity case are then determined relative to the reference case. The core peaking factor uncertainty sensitivity result is determined by multiplying the Δcal/gm result of the reference case by the core peaking factor uncertainty value. A Δ/Δcal/gm sensitivity result is then obtained for each of the four sensitivity results by subtracting the reference case Δcal result. These four individual Δ/Δcal/gm results are then combined using the Square-Root-Sum Squares (SRSS) method to obtain the statistical Δ/Δcal/gm result.

This statistical Δ/Δcal/gm result is then added to the reference case Δcal/gm result to obtain the statistical Δcal/gm result. This result is then added to the initial cal/gm value to obtain the statistical peak cal/gm result. If the acceptance limits are exceeded, then one of the following options can be taken: a) perform a hot rod model analysis as a function of burnup to take advantage of more detailed modeling of the fuel rod transient response, b) re-design the core to obtain more favorable rod ejection analysis results, or c) change allowable operating conditions such as the rod insertion limit.

Step 6 – Reload Core Checks

Each reload core must be evaluated to confirm that the basic assumptions (fuel design, reactor operation, control rod insertion limits, fuel rod burnup, etc.) in the REA analyses of record remain valid. Each reload can then be screened for the ejected rod worth in dollars (ERW$). If the ERW$ including uncertainty is too small to produce a significant Δcal/gm result, then the reload does not need to be analyzed. If the reload is not screened out by the above ERW$ check,

then the key REA physics parameters must be evaluated to determine if the REA analyses of record remain bounding. If this determination cannot be made with confidence, then the REA must be reanalyzed to bound the reload core and shown to meet the acceptance limits.

In addition, if the future regulatory limit on fuel enthalpy are a function of burnup, and if a sufficient margin to the limit does not exist for the REA at all burnups, then the spectrum of possible ejected rod locations will likely require reanalysis each reload to address the reload design-specific nature of the REA analysis results.

Table 3-1
Parameters for PWR 3D REA Fuel Enthalpy Analysis

Parameter	Sensitivity on Fuel Enthalpy	Analysis Value
Initial Conditions		
Initial power	High	HZP
Core design	High	Reactor and cycle-specific
Time-in-cycle	High	BOC and EOC
Initial core peaking factor	Low	Nominal
Rod position during depletion	High	Nominal
Xenon condition	High	Analysis-specific
Initial rod insertion	High	To the insertion limit
Coolant pressure	Low	Nominal
Coolant temperature	Low	Nominal
Coolant flow rate	Low	Nominal
Kinetics Parameters		
Ejected rod worth ($)	High	Key parameter - uncertainty value included
Fuel Doppler temperature coefficient	High	Key parameter - uncertainty value included
Moderator temperature coefficient	Low	Key parameter - uncertainty value included
Neutron velocity (3D equivalent to prompt neutron lifetime in point-kinetics)	Low	Nominal
Trip worth	Low	Bounding low
Trip insertion curves	Low	Bounding slow
Ejection time	Low	Bounding fast

Table 3-1
Parameters for PWR 3D REA Fuel Enthalpy Analysis (Continued)

Parameter	Sensitivity on Fuel Enthalpy	Analysis Value
Fuel Parameters		
Fuel conductivity	High	Nominal
Fuel heat capacity	High	Nominal
Cladding conductivity	Low	Nominal
Cladding heat capacity	Low	Nominal
Cladding-coolant heat transfer	Low	Bounding low
Time of DNB	Low	Bounding early
Initial gap conductivity	Low	Bounding low

4
PROBABILITY-BASED APPROACH FOR SELECTING INITIAL CONDITIONS

4.1 Background

The traditional approach to performing analyses of UFSAR Chapter 15 transients and accidents has relied on selecting initial conditions and boundary conditions that bound the design of the plant and the applicable modes of operation. This approach is intended to ensure an overall conservative analysis result by selecting conservative values for the key parameters that influence the transient response, and then setting the parameters of lesser importance to nominal values. The conservative values typically include an allowance for uncertainty, whereas the nominal values do not. This selection process includes consideration of the range of parameter values that are specified in such documents as the technical specifications, the core operating limits report, the UFSAR, plant procedures, and design basis documents. The range of plant operating conditions such as initial power level, time-in-cycle, status of control systems, and operator actions must also be considered. The large number of combinations of these possible plant conditions and parameter values must be systematically evaluated to determine a manageable number of cases to be analyzed. From the set of analyzed cases the limiting case or cases are then obtained and serve as the design basis analyses.

This traditional approach has always included consideration of the likelihood of the various plant operating conditions or values of key parameters. However, until recently this has typically been a qualitative process based on engineering judgement and experience. The use of quantitative probability-based arguments lends itself well to the selection and specification of inputs to UFSAR accident analyses. The extent to which application of a probability-based approach can add value is highly dependent on the complexity of a particular transient or accident and the objectives of the analysis. This technology can be applied in a limited way to establish, for example, the range of initial power levels that should be analyzed for a given accident scenario. It can also be applied to quantify the change in the core melt frequency, for the purpose of determining whether or not a particular scenario has a high enough probability to warrant inclusion in the design basis.

Given that the REA is a low probability event, there is merit in using a probability-based approach to define the scope of REA analysis. The range of possible applications of a probability-based approach to the REA analysis was evaluated based on the objectives of the REA 3D methodology and the nature of the REA event. One possible application was to determine if the rod ejection accident was of sufficiently low probability that it could be eliminated as a design basis accident. This concept was discussed in various meetings within the industry and with NRC as the CABRI test results were released and the safety significance of the test results was being characterized. The industry conclusion was that pursuing elimination of the

REA from the spectrum of design basis accidents was not widely supported. Consequently, the application of probability-based arguments in the REA 3D methodology is proposed for determining the core initial conditions that are analyzed. In particular, off-normal conditions that can occur during operation and that are permitted by the technical specifications and the core operating limits report will be included or deleted from the scope of the REA 3D methodology using a probability-based approach.

4.2 REA Initiating Event Frequency

A rod ejection event has never occurred in the history of commercial pressurized water reactor operation, and is generally recognized as a low probability event. The REA initiating event frequency will be based on zero occurrences in the history of commercial PWR operation using the following formula to estimate the mean value for the frequency [9]:

$$f = \frac{2s+1}{2T}$$

where,

f = frequency of the event

s = number of occurrences of the event

T = the time period

Given that "s", the number of occurrences of REA events, is zero, the frequency "f" reduces to 1/2T. The value of "T" is chosen as the total number of PWR years of operation during which a REA was possible. From Reference 10, the total world-wide experience for "western style" PWRs through 1997 is 3362 calendar years. From Reference 11, there are 256 total PWRs in operation world-wide, with 204 of "western-style" as of December 2000. With 4.25 calendar years after 1997 through December 2001, the total calendar years of PWR operation of interest is:

Western-style PWR years = 3362 + (204)(4.25) = 4229

To estimate the fraction of time that a PWR would be pressurized, and therefore capable of ejecting a control rod, a factor of 60% is chosen. This is a conservatively small value. This factor is applied to obtain the value for "T" and the REA initiating event frequency as follows:

T = (4229)(0.60) = 2537 reactor-years (RY) and pressurized

f = 1/2T = 1/(2)(2537) = 1.97E-4 /RY (This will be labeled "F-rea")

This value of the REA initiating event mean frequency "F-rea" will be used in the methodology along with the probability of other core conditions to define the scope of the initial conditions for the REA analysis.

4.3 Evaluation of REA Core Initial Conditions

The objective of the REA 3D methodology is to more accurately calculate the REA Δcal/gm analysis result in anticipation of a lower cal/gm regulatory acceptance limit in the future. This objective enables a narrow focus on the REA event as a core transient event rather than a plant transient event, since the Δcal/gm result of interest occurs in the first few seconds following the ejection of the control rod, and the plant response in that time period does not significantly influence the Δcal/gm result.

The most important parameter in the REA analysis is the reactivity worth of the ejected rod. First principles, industry REA analysis experience, and the results of sensitivity analyses performed by the EPRI REA 3D Focus Group in the development of this methodology, have confirmed that an ejected rod worth in excess of one dollar of reactivity is necessary to obtain a significant Δcal/gm result. This leads to a review of the core operating conditions for the reactor of interest to determine what conditions can produce a sufficiently large ejected rod worth. Typically the control rod must be fully or mostly inserted. The control rod insertion limits and nuclear analysis calculations of the ejected rod worth will determine the probability of an ejected rod event having a sufficient ejected rod worth to produce a significant Δcal/gm result.

Consistent with the requirement for the control rod to be fully or mostly inserted to have a sufficiently high ejected rod worth is the likelihood that the reactor will be at a low power level. This is in fact one of the key considerations in the setting of the control rod insertion limits, i.e. the ejected rod worth is limited as a function of power level by the rod insertion limits. This leads to a review of the operating history for the reactor of interest to determine the probability of operating at the low power levels and corresponding control rod positions that are required to produce a sufficiently large ejected rod worth. The probability of operating at these conditions of interest at the time that a rod ejection occurs will contribute to the overall probability.

4.4 Methodology

The methodology uses a frequency of 1.0E-7/RY as a screening criterion to determine which core initial conditions combined with the initiating frequency of the REA event are sufficiently probable to be included in the UFSAR REA analyses. For REA events from core initial conditions with a combined frequency of less than 1.0E-7/RY, the event is of sufficiently low frequency that it can be screened out or dropped from further consideration. The value of 1.0E-7/RY has previously been used in Reference 12 for the same purpose of quantifying REA sequences that can be neglected. ANSI/ANS 51.1-1983 (Reference 13) states the following. "If the frequency of occurrence of an event is shown to be <10-6/reactor year on a best estimate basis, this event shall not be considered for the design." These references show both a precedent and a margin of conservatism in the selection of the proposed REA screening criterion.

The REA initiating event frequency, "F-rea", described above, is then combined with the frequency or probability of the core conditions needed for a REA analysis to produce a significant Δcal/gm result.

The control rods must be fully or mostly inserted with the reactor critical or near critical for a REA of concern to occur. The probability based on the fraction of time that the core is at these conditions will be labeled "P-critical". For analysis purposes it is convenient to distinguish between the "P-critical" period of time at beginning-of-cycle (BOC) following a refueling outage (P-critical/BOC), and the other times during the fuel cycle (P-critical/not-BOC). The values of this parameter are obtained based on historical data for the number and duration of periods of operation at these core conditions for which a significant REA event could occur. The historical average number of hours per year at these conditions is then divided by 8760 (hours in one year) to obtain the probability.

For a given reactor design at the above "P-critical" core conditions of interest, not all control rods will have sufficient ejected rod worth to cause a power excursion that can result in a significant Δcal/gm result. Assuming that a rod ejection is equally probable for all of the control rods, the probability that a control rod can cause a significant Δcal/gm result during the "P-critical" core conditions of interest can be determined by nuclear analysis. This factor will be labeled "P-prompt" to indicate the probability of an ejected control rod causing a significant prompt-critical response.

Based on industry experience in REA analysis and the sensitivity studies performed by the EPRI REA 3D Focus Group, there exist a number of off-normal core conditions that can worsen the core response during a REA, and produce a higher Δcal/gm result. The most significant off-normal condition is an adverse xenon distribution. The effect of an adverse xenon distribution is to increase the ejected rod worth. The methodology addresses the effect of xenon in the following way. For the "P-critical/BOC" core condition, which is limited to the initial startup following a refueling outage, no xenon is present and therefore does not need to be modeled. For the "P-critical/not-BOC" core conditions, which include all other periods of operation with the control rods inserted or mostly inserted, and with the reactor critical or near-critical, the effect of xenon on the core initial conditions will be considered. A xenon penalty will be considered for the xenon distributions that can potentially exist, but can exclude the xenon distributions during the time to recover from a reactor trip or shutdown. This credits only the shortest time to restart the reactor based on current plant procedures, and excludes the xenon distributions while the reactor is shut down. In other words, the xenon distributions are to be consistent with the plant evolution prior to the initiation of the accident. An alternative acceptable approach is to simply assume a bounding xenon distribution for the REA analyses.

Other off-normal core conditions will be handled using the following approach. First, the possible off-normal conditions that could potentially worsen the REA Δcal/gm analysis results must be identified. These off-normal conditions are typically allowed by the plant technical specification or core operating limits report, and include:

- Sustained operation at less than full power
- Load follow operation
- Operation with a core power tilt
- Operation with control rods out of alignment

These off-normal conditions all have the potential for increasing the ejected rod worth and/or increasing the pre- or post-ejected power distribution, and thereby increase the Δcal/gm result. The fraction of time during the core conditions of concern that the off-normal conditions exist must then be quantified as a probability. The methodology addresses these off-normal core conditions by determining if the probability of these conditions (labeled "P-offnormal"), in combination with the product of the other terms defined above, is greater than or less than the REA screening criterion of 1.0E-7/RY. This can be expressed by the following total frequency labeled "F-total":

IF F-total = (F-rea)(P-critical)(P-prompt)(P-offnormal) <1.0E-7/RY

THEN the core conditions represented by "P-offnormal" can be excluded from the core conditions assumed in the UFSAR REA analysis.

Conversely,

IF F-total ≥ 1.0E-7/RY, **THEN** the core conditions represented by "P-offnormal" must be included in the core conditions assumed in the UFSAR REA analysis.

The off-normal core conditions are to be considered in combination during this step of the methodology, if applicable.

4.5 Demonstration of Methodology

This probability-based approach to selecting the core initial conditions to be analyzed in the REA 3D methodology is demonstrated here with some arbitrary values.

P-critical/BOC is obtained from the fuel cycle length and the duration of core conditions during the initial startup following the refueling outage. Assuming a two-year fuel cycle and a 20 hour period of time at which the core is critical or near-critical, and some control rods are fully or mostly inserted:

P-critical/BOC = 20 hours/(2 x 8760) = 1.14E-3

P-critical/not-BOC is obtained from the number of reactor startups per year excluding the initial startup following the refueling outage. Assume two startups per year and an 8 hour period of time at which the core is critical or near-critical, and some control rods are fully or mostly inserted:

P-critical/not-BOC = (2)(8)/8760 = 1.83E-3

P-prompt is obtained from a nuclear analysis of the ejected rod worth for all inserted or mostly-inserted control rods, for the P-critical core conditions above. The analysis determines how many control rods have sufficient ejected rod worth to cause a significant Δcal/gm result. Assume that there are 53 control rods, and that 8 can produce a sufficiently large ejected rod worth:

P-prompt = 8/53 = 0.15

The value of P-prompt will be zero if no control rods are capable of causing a significant Δcal/gm result. A zero value of P-prompt justifies that no REA analyses are necessary to show that the cal/gm regulatory limit is met.

The xenon distributions that must be considered are based on whether or not the core condition is the beginning-of-cycle startup following a refueling outage or not. For the BOC core condition no xenon is appropriate. Assume that the shortest time to restart the reactor following a shutdown or trip is 12 hours. Then, for the non-BOC core conditions being analyzed, exclude the xenon distributions that can only occur during the first 12 hours post-trip or post-shutdown.

To evaluate P-offnormal, assume an annual off-normal core condition during which the core is operated at 50% power for 30 days due to a reduced demand for power. Since this is not at BOC, the P-critical/not-BOC probability from above will be used. The value of P-offnormal is calculated by:

P-offnormal = 30 days / 365 days = 8.22E-2

To determine if the 50% power core initial condition must be analyzed as a REA initial core condition, calculate F-total:

F-total (50% power) = (F-rea) (P-critical/not-BOC)(P-prompt)(P-offnormal)

= (1.97E-4/RY) (1.83E-3) (0.15) (8.22E-2)

= 4.45E-9/RY

Since this frequency is less than the screening criterion of 1.0E-7/RY, the 50% power off-normal core conditions do not need to be analyzed. For this numerical example, since the product of the first three terms is less than the screening criterion, no off-normal core conditions must be analyzed. This result indicates that for the example conditions the probability of a large ejected rod worth is very small, and additional low probability off-normal core conditions do not need to be considered.

4.6 Confirmation of Continued Validity of the Approach

The probability-based approach involves quantifying plant-specific values for the inputs necessary to implement the methodology. These inputs include the number of unit startups per year, the number of hours per year that the core conditions necessary for a significant REA event to occur exist, the fraction of the control rods that can lead to a prompt-critical REA event, etc. These inputs to the methodology must be checked on a periodic basis to ensure continued validity. For example, if a switch from annual to 2-year fuel cycles is made, the value of P-critical/BOC will decrease by a factor of two. Similarly, if operating experience data shows that the number of unit startups per year changes, then the value of P-critical/not-BOC will change accordingly. The selection of the inputs can also be made with attention to the margin available in the results of the application of the methodology, so that there is less chance that the input value will be exceeded in the future. This periodic check can be incorporated into the review of the REA analysis that is required for each reload core design. If an input value is exceeded then an evaluation of the impact of the change must be performed. The results of the evaluation might include a reanalysis or implementation of other compensatory measures.

4.7 Alternative Approach

As an alternative approach to the probability-based methodology, the REA 3D analysis can assume conservative bounding values for analysis inputs without consideration of probability.

5
CORE MODEL

Three-dimensional (3D) core models for transient analysis are usually based on a similar 3D model used for static core analysis and fuel management. The core is usually divided into a number of radial and axial nodes. The radial nodalization of a fuel management model typically uses 2x2 radial nodes per assembly to correctly account for the effect of exposure gradients on the radial power distributions. For transient analysis, especially at end-of-cycle, a 1x1 radial node per assembly model with node properties homogenized from a 2x2 fuel management model may be adequate. Axial nodalization is usually a result of practical considerations such as the location of axial zone boundaries, which occur because of part length absorber rods, axial blankets, and other discontinuities. A typical PWR with a 12 foot fuel active length is typically divided into 16-24 axial nodes.

Starting with the 3D core model and the initial conditions for the REA analysis, ejected rod worths are determined from static calculations for the possible ejected rods. A transient REA analysis is then conducted for ejected rods of sufficient worth to produce a significant Δcal/gm result. The 3D transient REA calculation provides fuel temperature results for each node in the core. These results can be used directly to calculate fuel enthalpy on a node averaged basis. Due to local power peaking effects, the hottest pin in a node and the hottest pellet in the hot pin will have higher power than the node average. These local effects must be addressed in the calculation of the fuel enthalpy. The transient core power and power distribution obtained from the 3D model can also be used as initial and boundary conditions to drive a separate detailed hot rod model.

5.1 Code Requirements

The 3D kinetics code must solve the three-dimensional space-dependent reactor static and transient neutronic problem with thermal-hydraulic feedback.

The problem can be stated as a set of time and space-dependent coupled partial differential equations. In a multigroup diffusion theory approximation these are written in a matrix form with brackets denoting the matrices as:

$$\underline{\nabla} * [D(\underline{r},t)]\underline{\nabla}[\Phi(\underline{r},t)] - [\Sigma_T(\underline{r},t)][\Phi(\underline{r},t)]$$

$$+ (1-\beta)[\chi_p]\left[\frac{1}{\lambda}\nu\Sigma_f(\underline{r},t)\right]^T [\Phi(\underline{r},t)]$$

$$+ \sum_{d=1}^{D} \lambda_d [\chi_d] C_d(\underline{r},\underline{t}) = [u]^{-1}\frac{\partial}{\partial t}[\Phi(\underline{r},t)]$$

And,

$$\beta_d \left[\frac{1}{\lambda} \nu \Sigma_f(\underline{r},t)\right]^T [\Phi(\underline{r},t)] - \lambda_d C_d(\underline{r},t) = \frac{\partial}{\partial t} C_d(\underline{r},t); d = 1, 2 \ldots D$$

Where,

$D(\underline{r},t)$	=	diffusion coefficient
$\Phi(\underline{r},t)$	=	flux
$\Sigma_T(\underline{r},t)$	=	macroscopic total cross section minus scattering cross sections
β	=	total delayed neutron fraction
χ_p	=	prompt fission neutron fraction
λ	=	eigenvalue
ν	=	neutron yield per fission
Σ_f	=	macroscopic fission cross section
λ_d	=	delayed neutron decay constant
χ_d	=	delayed fission neutron fraction
$C_d(\underline{r},t)$	=	delayed neutron precursor concentration
$\lfloor u^{-1} \rfloor$	=	inverse neutron velocity
β_d	=	delayed neutron fraction

The above notation is standard except that the matrix $[\Sigma_T(\underline{r}, t)]$ contains the macroscopic total cross section minus scattering cross sections and β represents the total delayed neutron fraction, while β_d represents the delayed neutron fraction for each delayed neutron group.

This problem has been solved by fine-mesh diffusion theory methods, although these are very computer-resource intensive. Coarse-mesh diffusion theory methods are very inaccurate unless modified. Modified coarse-mesh diffusion theory methods have been successfully applied to this problem. Various finite-element models have also been successfully applied to this problem. Many current 3D models use an advanced nodal method, incorporating transverse –integrated leakages. Advanced nodal methods typically accept homogenized fuel assembly cross sections as

input, with assembly discontinuity factors to account for the heterogeneous fuel. The global flux distributions in the assembly can be readily calculated. This allows for overlaying local flux variations on the global solution in the assemblies to reconstruct pin power distributions. Pin power reconstruction from global fluxes and local variations is a useful capability.

Cross-section models must account for exposure, fuel and moderator temperatures, boron and fission product concentrations, temperature history and boron history effects, and control rod history effects.

The thermal-hydraulic feedback part of the calculation requires static and transient models of fuel temperature and water temperature or density. The essential feedback variables are the average fuel rod temperature, the average coolant temperature, and the average coolant density.

Very brief transients can be solved using an adiabatic assumption that no heat is transferred from the fuel to the coolant. This adiabatic assumption can potentially produce overly-conservative results. However, this assumption is non-conservative in the presence of a positive moderator coefficient.

The fuel rod model requires material properties and powers from the nuclear calculation, and coolant conditions and heat transfer coefficients as boundary conditions. The coolant model requires material properties, heat transfer coefficient models, and boundary conditions such as inlet flow, temperature, and exit pressure. Calculated fuel temperatures and moderator temperatures or densities are fed back to the cross section model to determine cross sections in the solution mesh.

The transient solution is typically obtained by breaking the transient up into time steps. The time steps may be completely specified by the user, or calculated by the code based on limiting the magnitude of change per time step for key parameters, or other algorithms. The solution in each time step is usually as follows:

1. Apply cross section changes due to external perturbations
2. Update thermal-hydraulic boundary conditions
3. Update coupling and advance flux solution
4. Update thermal-hydraulic variables
5. Update cross sections
6. Update delayed neutron precursors and prompt and delayed neutron solution
7. Perform edits and start next time step

Note that the nuclear and thermal-hydraulic calculations may have different limiting time steps, and both must be accounted for. Generally, time-step sensitivity studies will be necessary to assure numerical convergence. Higher Δcal/gm values are usually associated with larger power excursions or smaller pulse widths, and hence require smaller time steps.

The power is calculated for every node in the core at every time step in the course of the transient calculation. The power distribution for any assembly in the core can then be obtained. If pin power reconstruction techniques are used, or if a pin-to-box factor is applied, the power

distribution for any fuel pin in the core can be obtained. This result can also be used as a boundary condition for a hot rod model.

The 3D core model input typically consists of:

- Code options and iteration parameters
- Core neutronic boundary conditions
- Core composition and mesh spacing
- Cross sections and delayed neutron parameters
- Transient time step inputs
- Thermal-hydraulic parameters and conditions including inlet flow, inlet temperature, and core pressure
- Control rod positions
- Control rod, thermal-hydraulic and boron perturbations during the transient
- Fuel rod thermal parameters

Cross sections are almost always obtained from external library files. Delayed neutron parameters may also be obtained from external library files.

Reactivity parameters such as control rod worth, Doppler coefficient, and moderator coefficient are functions of the input geometry, initial state variables, and the cross section library. It is seldom possible to change one parameter without influencing others in a 3D model. For example, a change in control rod worth will also change peaking factors.

The 3D core model output for practical reasons consists of a chosen subset of the parameters calculated during the transient. It is possible to produce powers and state variables (such as fuel temperature) for every solution node for every time step in the core. Since this would be impossibly voluminous, a more limited set of solution nodes and time steps is normally chosen. Parameters such as the total core power, maximum peak-to-average power, and peak power location may be edited frequently, while the power distribution and/or the thermal-hydraulic parameter distribution may be edited at less frequent intervals.

Important variables in the analysis are the fuel arrangement, fuel cross sections and delayed neutron parameters, fuel node exposure and history parameters, initial control rod positions and initial thermal-hydraulic parameters.

5.2 Typical Initial and Boundary Conditions

The worst condition for a rod ejection accident is typically at the end-of-cycle (EOC) for the hot zero power (HZP) condition with control rods at the insertion limit. The HZP condition has the largest allowable rod insertion limits (maximizing potential ejected rod worth) and minimizes the benefit of thermal-hydraulic feedback on the event. The EOC exposure minimizes the value of beta, the delayed neutron fraction, thereby maximizing the ejected rod worth in dollars.

5.3 Rod Ejection Accident Modeling Elements

The REA 3D Analysis Focus Group developed a table to identify the elements to be considered in the development of the proposed REA 3D analysis methodology. As a starting point for identifying these elements, Regulatory Guide 1.77 [1] was used. Table 5-1 identifies the elements and the disposition of each element in the proposed REA 3D methodology as decided by the Focus Group. The elements that are related to the fuel rod model are shown in Chapter 6, Table 6-1. It is also noted that additional elements that were identified by the Focus Group that were not included in Regulatory Guide 1.77 were added to the table at the bottom.

Table 5-1
REA 3D Methodology / 3D Kinetics Model Elements

REA Analysis Considerations For Elements Of Reg. Guide 1.77	Focus Group Decision For Proposed REA 3d Methodology
A. Initial Core Conditions Zero power (BOL & EOL) Low power (BOL & EOL) Full power (BOL & EOL)	Use probability-based approach to determine what initial core conditions must be analyzed. As a minimum analyze HZP initial condition at BOC and EOC.
B. Loss of Primary System Integrity Effects to be included	RCS overpressure not part of the scope
C. Ejected Rod Worth	
a. Maximum inserted position based on power level	a. Consistent with rod insertion limits
b. Additional fully or partially inserted misaligned or inoperable rods if allowed	b. Use probability-based approach to address off-normal conditions
c. Increase worth to account for calculation uncertainties	c. **Key parameter** - uncertainty value included. The uncertainty in ejected rod worth will be included in combination with the uncertainty in beta-effective. This is based on how rod worths are measured.
d. Increase worth to account for xenon transients	d. No xenon penalty for HZP BOC case since no xenon should exist. For HZP EOC and other cases, xenon distributions are to be consistent with the plant evolution leading to the core initial conditions.
D. Reactivity Insertion Rate	
a. Based on differential rod worth curve and rod position vs. time curve	a. The 3D model will predict the reactivity worth as a function of ejected rod position.
b. Rate of ejection based on maximum ΔP and weight and cross-sectional area of the control rod and drive shaft	b. Considered in input to model. Typical ejection time is 0.1 second (from full insertion).

Table 5-1
REA 3D Methodology / 3D Kinetics Model Elements (Continued)

REA Analysis Considerations For Elements Of Reg. Guide 1.77	Focus Group Decision For Proposed REA 3d Methodology
E. Effective Delayed Neutron Fraction and Prompt Neutron Lifetime	
a. Use available data and average based on fission fractions	a. Use of 3D model allows nodal values to be used
b. Use minimum calculated value for the given reactor state	b. Values consistent with time-in-cycle will be used
c. Consider both the power excursion and the power reduction when selecting a conservative value	c. **Key parameter** - uncertainty value included. The uncertainty in beta-effective will be modeled in combination with the ejected rod worth uncertainty. The prompt neutron lifetime is replaced by the inverse velocity when using a 3D model. This parameter is not a key parameter as determined by sensitivity analyses, and nominal values will be used.
F. Initial Pressure, Flow, and Temperature	For the Δcal/gm analysis, nominal values are suitable for the 3D analysis. Use conservative values in the hot rod analysis. Ensure that flow is consistent with technical specifications.
G. Fuel Thermal Properties	
a. Fuel-cladding gap gas conductivity	a) Bounding low
b. Fuel thermal conductivity	b) Nominal
c. Direct moderator heating	c) Nominal Refer to Table 6-1 for Fuel Rod Model
H. UO_2 Specific Heat	Use nominal values in 3D code Refer to Table 6-1 for Fuel Rod Model
I. Moderator Reactivity Coefficient To include effects of voids, pressure, temperature, and boron	**Key parameter** - uncertainty value included. The uncertainty will be quantified and applied either deterministically or statistically.

Table 5-1
REA 3D Methodology / 3D Kinetics Model Elements (Continued)

REA Analysis Considerations For Elements Of Reg. Guide 1.77	Focus Group Decision For Proposed REA 3d Methodology
J. Doppler Coefficient To include corrections for pin shadowing and should compare conservatively to data. Uncertainty in fuel temperature to be included.	**Key parameter** - uncertainty value included. The uncertainty will be quantified and applied either deterministically or statistically.
K. Control Rod Reactivity Insertion on reactor trip to include correct initial position, differential worth curve, etc.	Use minimum trip worth and conservative rod position vs. time.
L. Reactor Trip Delay Time	Use conservative trip delay time.
M. Computer Code a. Coupled t-h/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	All addressed by use of 3D code.
N. Analytical Models and Computer Codes a. Documented and justified b. Conservatism evaluated by comparison with experiment or more sophisticated codes c. Changes in flux shapes should be investigated 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 e. Sensitivity studies on Doppler, power distribution, fuel heat transfer parameters and other relevant parameters should be included	These issues will be addressed by each organization in the submittal describing their specific application of the 3D method.
O. Pressure Surge	RCS overpressure not part of the scope.
P. Pin Census	Conventional approach to performing a pin census is acceptable

Table 5-1
REA 3D Methodology / 3D Kinetics Model Elements (Continued)

REA Analysis Considerations For Elements Of Reg. Guide 1.77	Focus Group Decision For Proposed REA 3d Methodology
Additional Items not Called Out in R.G. 1.77	
AA. Initial Power Distribution / Cross Sections	Initial power distribution and cross sections to be consistent with technical specifications and core operating limits. Modifications to cross sections are then implemented to obtain the desired changes in ejected rod worth, Doppler feedback, and moderator feedback. Off-normal conditions, such as core tilt and misaligned control rods to be evaluated by the probability-based approach in Chapter 4.
BB. Pin-to-Node Factor	The effects of local power peaking are to be addressed
CC. Reload Checks	Conventional approach will confirm that the analysis of record remains valid for each reload core. Otherwise the analysis must be revised, or the core must be re-designed.
DD. Pellet-Cladding Gap Model	Use nominal values in the 3D code, with static or dynamic gap model. Ensure modeling is appropriate if cal/gm results are sensitive. Refer to Table 6-1 for Fuel Rod Model
EE. Onset of DNB	Can be predicted by an approved CHF correlation or assumed to occur at a bounding time.
FF. Post-DNB Heat Transfer	Any credit for post-DNB heat transfer must be justified. Refer to Table 6-1 for Fuel Rod Model
GG. Calculation of cal/gm	Can use either the 3D code fuel rod model heat transfer solution, or a calculation of energy deposition. Need to include the local power distribution effect. Need to ensure that any heat transfer from the fuel pellet is conservatively modeled. Refer to Table 6-1 for Fuel Rod Model
HH. Fuel Pellet Radial Power Profile	The effect of the pellet radial power profile must be considered. Refer to Table 6-1 for Fuel Rod Model

6
FUEL ROD MODEL

The most important factors in the fuel rod analysis for calculating the fuel enthalpy increase are the power produced and the fuel heat capacity. Fuel rod models in rod ejection analysis have two purposes. One is to calculate fuel rod temperatures in the 3D nodal code to determine Doppler feedback. The other is to calculate the fuel enthalpy or to provide input for fuel rod enthalpy calculations in a hot rod model.

These models can range from simple adiabatic models, to lumped parameter models with user input parameters or input from material property routines, to more detailed heat conduction models which explicitly represent the fuel, clad, gap, coolant and surface heat transfer effects.

Using an adiabatic assumption for the surface heat transfer, the fuel temperature is found directly from the power and heat capacity such that:

$$\frac{\partial T}{\partial t} = \text{Power density/heat capacity}$$

In a lumped parameter model, an equivalent heat capacity and equivalent thermal conductivity for the fuel rod are found for the combined fuel, gap, and cladding. The heat transfer at the rod surface and the coolant properties are determined separately.

Node average models can be used to find the fuel rod enthalpy on a node average basis. Pin power peaking factors, obtained from pin-to-box factors or pin power reconstruction, are necessary for analysis of the highest power rod.

The time-dependent core power and power distribution obtained from a 3D nodal code can be used to drive a hot rod model. In a simple case, the hot rod model could be basically the same as the nodal model, but using the higher powers associated with the peak rod. In the extreme case, the hot rod model could be a transient fuel mechanical model with a complete dynamic representation of a fuel rod with fuel relocation, clad expansion, dynamic gap conductance, and other mechanical and thermal-hydraulic effects accounted for.

6.1 Code Requirements

The fuel rod model must calculate fuel temperatures, either node-averaged for use in Doppler feedback calculations, or pin-specific, to determine the fuel enthalpy values. The ultimate purpose of the fuel rod model is to produce fuel rod enthalpy in cal/gm or a Δcal/gm values.

The fuel enthalpy can be determined from a correlation of UO_2 enthalpy to UO_2 temperature, such as that of Kerrisk and Clifton [14].

The fuel heat capacity can be input by the user or found from material property routines such as MATPRO [15].

Heat transfer to the coolant during the event will affect the fuel temperature and enthalpy depending on whether the moderator coefficient is positive or negative. The Doppler feedback will increase with a decrease in heat transfer to the coolant, which would tend to decrease the peak power in the event. Moderator feedback can either increase or decrease reactivity and the peak power in the event. Heat transfer may be by conduction through the cladding and heat transfer at the fuel rod surface, or by direct moderator heating in the coolant from neutrons and gamma radiation.

6.2 Initial and Boundary Conditions

All models must have an initial value of fuel temperature. For fuel rod models with heat transfer to the coolant, the initial coolant conditions of flow, pressure, and enthalpy or temperature must be supplied. The pellet-cladding gap must be characterized in terms of width, gap conductivity, etc. The occurrence of a departure from nucleate boiling condition must be predicted with an appropriate critical heat flux correlation.

Assuming an adiabatic boundary condition for the fuel rod would tend to give more conservative results, as none of the heat produced in the rod during the event is lost. The only factors in an adiabatic fuel rod analysis are the power produced and the fuel heat capacity.

6.3 Rod Ejection Accident Modeling Elements

Similar to the information presented on the core physics modeling elements in Table 5-1, Table 6-1 shows the Focus Group position on the various elements of Regulatory Guide 1.77 [1] related to fuel rod modeling.

Table 6-1
REA 3D Methodology/Fuel Rod Model Elements

REA Analysis Considerations for Elements of Reg. Guide 1.77	Focus Group Decision for Proposed REA 3d Methodology
G. Fuel Thermal Properties	
a. Fuel-cladding gap gas conductivity	a. The gas conductivity is modeled as a bounding low value
b. Thermal conductivity	b. Use standard values for material properties
c. Direct moderator heating	c. Use nominal value
H. UO_2 Specific Heat	Use standard values for material properties.
DD. Pellet-Cladding Gap Model	Static or dynamic gap model can be used. Any increase in heat transfer, such as would occur due to gap closure, must be justified.
EE. Onset of DNB	Can be predicted by an approved CHF correlation or assumed to occur at a bounding early time.
FF. Post-DNB Heat Transfer	Any credit for post-DNB heat transfer must be justified
GG. Calculation of cal/gm	The effects of 3D local power peaking must be included.
HH. Fuel Pellet Radial Power Profile	The effect of the pellet radial power profile must be considered.

7
DEMONSTRATION ANALYSES

7.1 Overview

The objective of the REA 3D methodology is to provide a conservative but more realistic REA analysis methodology through the explicit 3D analysis of the REA core response. The proposed REA 3D methodology described in the previous chapters of this report is applied in demonstration analyses that are detailed in Appendices A, B, C, and D. These demonstration analyses are summarized in this chapter.

The methodology is demonstrated for four reactor types - a two-loop Westinghouse reactor (Appendix A), a two loop Westinghouse/CE reactor (Appendix B), a four-loop Westinghouse reactor (Appendix C), and a B&W reactor (Appendix D). The analyses employ three different 3D core kinetics codes. Some of the demonstration analyses also employ a separate hot fuel rod analysis code.

End-of-cycle was selected due to the typically higher value of ejected rod worth in dollars (ERW$), and due to interest in the effect of higher burnup fuel. In addition, all of the analyses assumed a value of 10% for the uncertainty in the ejected rod worth in dollars, the Doppler fuel temperature coefficient of reactivity, and the moderator coefficient of reactivity. The value of 10% uncertainty is chosen for demonstration purposes only.

None of the demonstration analyses include "off-normal" core conditions as described in the probability-based approach in Chapter 4. This can be interpreted as being consistent with having shown by following the methodology of Chapter 4 that the off-normal core conditions were of sufficient low probability that they could be excluded from the cases analyzed.

The statistical analysis approach described in Chapter 3 is applied for each of the demonstration analyses.

The ejected rod worth for some of the demonstration analyses was increased from the actual value to a value large enough to obtain a prompt-critical power excursion. This was accomplished by applying a multiplier to the appropriate cross section inputs. Increasing the ejected rod worth was necessary to demonstrate the methodology. Otherwise the core response to the actual ejected rod worth would be benign in most cases.

7.2 Summary of Results

The details of the analysis results are provided in the respective appendices. Some of the key observations are as follows, with key results summarized in Table 7-1.

1. For some cores, the ejected rod does not have sufficient worth to cause a significant prompt-critical core power response.

2. The ejected rod worth in dollars is the dominant input parameter. A value in the range of $1.40 is necessary to obtain a significant power excursion.

3. An ejected rod worth on the order of $1.50 will result in a peak enthalpy increase of about 45 Δcal/gm, and a peak fuel enthalpy of about 65 cal/gm.

4. The pulse width decreases with increasing ejected rod worth. Values ranged from 25 to 28 msec for the more limiting analyses.

5. The margin gained from the statistical analysis approach was small. For larger ejected rod worths or higher uncertainty values the margin gain can be larger.

6. A pin census analysis was not performed in the demonstration analyses because the peak cal/gm result was small.

At the time of publication of this report the future cal/gm regulatory acceptance limits are still a topic of discussion between industry and the NRC. Consequently, no conclusions can be made relative to the acceptability of the results of the demonstration analyses. Nevertheless, the proposed 3D methodology does provide a significant improvement in accuracy in the modeling of the REA in pressurized water reactors. The application of a probability-based approach to determine the core conditions to be analyzed is consistent with current regulatory initiatives. The combination of more accurate analytical methods and more realistic assumptions provides margin needed to address the current regulatory issue.

Table 7-1
Summary of Demonstration Analyses Results

Reactor Design	Reference Ejected Rod Worth ($)	Reference Case (Δcal/gm and Peak cal/gm)	Conservative Case (Δcal/gm and Peak cal/gm)	Statistical Case (Δcal/gm and Peak cal/gm)	Conservative Case Pulse Width (msec)
W Two-Loop	1.28	24 / 44	36 / 56	33 / 53	39
W/CE Two-Loop	1.45	30 / 48	43 / 61	38.1 / 56.1	32
W Four-Loop	1.32	24 / 39	33 / 49	31 / 47	28
B&W	1.39	31 / 46	42 / 57	39 / 54	25

8
CONCLUSIONS

A methodology for PWR REA analysis has been developed that uses 3D kinetics methods to more accurately represent the transient power distribution in the reactor core, and hence the energy deposition in the hot fuel rod. The methodology uses probability-based assumptions to limit the number of accident scenarios to be investigated. Key parameters in the analysis have been identified, and the uncertainties in these key parameters can be treated deterministically or statistically, as options to the user.

Using probability-based assumptions, an approach is provided to determine what initial conditions need to be considered in the scope of REA analysis for quantifying the margin to the cal/gm limit. This approach involves plant-specific data, but it is expected that most off-normal core initial conditions can be screened out based on low probability. The initial xenon distribution is also an important core initial condition that must be determined consistent with the evolution of plant operations at the time of the event. An approach for determining the range of initial xenon conditions to be considered is provided.

The key parameters in PWR REA analysis are determined to be the worth of the ejected rod in dollars, the fuel Doppler reactivity coefficient, the moderator reactivity coefficient, and the core power peaking uncertainty. Other parameters are known with good accuracy, and therefore nominal values can be assumed. A deterministic application of the methodology involves explicit analysis using conservative values for these key parameters. An analysis is performed for each of the core conditions that can result in a significant Δcal/gm result. In a statistical application of the methodology, a reference case calculation is performed by assuming nominal values for the key parameters. Then sensitivity cases are run which separately include the uncertainties in the key parameters. The results of these sensitivity cases are then combined using the SRSS methodology to obtain the statistical result.

A demonstration of the methodology has been undertaken for four different reactor types – a two-loop Westinghouse reactor, a two-loop Westinghouse/CE reactor, a four-loop Westinghouse reactor, and a B&W reactor. The demonstration analyses used similar initial conditions and assumptions for ease of comparison. The results of the demonstration cases should not be interpreted as bounding. For example, the uncertainty values used were assumed values only, and the probability-based element of the methodology was not performed. The results of the demonstration analyses do indicate that a substantial margin gain in the cal/gm result should be achieved relative to current licensing basis results.

The information presented in this report is considered to be useful as a standard REA 3D methodology approach to gain margin in the cal/gm result. Implementation of this methodology on a plant-specific basis will require additional methodology development and licensing actions to obtain NRC approval prior to implementation.

9
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A
DEMONSTRATION ANALYSIS FOR TWO-LOOP WESTINGHOUSE REACTOR

A.1 Overview

Prairie Island Units 1 and 2 have been in operation for more than 25 years. They represent an early Westinghouse core design consisting of a two-loop NSSS plant rated at 1650 MWth. The core consists of 121 14x14 fuel assemblies, currently all fabricated by Westinghouse. The core size, together with an optimized (aggressive, 4.95% U enrichment, 8% Gadolinium) core design and loading in order to achieve longer cycles, results in best-estimate ejected rod worths already above the $1 threshold. This REA study addresses the Prairie Island Unit 1 Cycle 19 core specifically. Figure A-1 shows the core loading pattern for that cycle. Assuming the core at hot zero power (HZP) and end-of-cycle (EOC), Figure A-2 identifies the ejected rod worths, assuming a $1/8^{th}$ core symmetry, based on the most limiting condition of all rods at the appropriate insertion limits. Note that these ejected rod worth values are obtained from a best-estimate model, no cross-section modification was needed. The nodalization scheme used for the REA simulations presented here specified 4 nodes assigned to each fuel assembly (2x2 radial nodalization per assembly) and 24 equally spaced axial planes.

CORETRAN [1] is an EPRI code developed for simulating light water reactor cores during both normal (e.g., depletion, core-follow) and transient operations. It is based on the Analytical Nodal Method (ANM) and allows a full three-dimensional time-dependent treatment of the reactor core. The ANM is derived from the 2-group time-dependent diffusion equations together with the assumption of six delayed neutron precursors families. All neutronic parameters (cross-sections, ADF's, Betas, velocities, pin-powers form functions) are treated as fuel type dependent and accordingly functionalized so that their dependence on both historical and transient feedback conditions can be emulated. CORETRAN offers 3 options for thermal-hydraulics: the Homogeneous Equilibrium Mixture (HEM) option (identical to the EPRI code VIPRE-01), the Closed-Channel Drift Flux option, and the Two-Fluid 6-equation option. For the study reported here, the second option was chosen where a nodal-averaged characteristic pin heat radial conduction is finite-mesh solved and coupled with the thermal-hydraulic condition of the local coolant. Fuel (UO_2) properties are based on MATPRO [2] correlations. CORETRAN is expected to be brought to the NRC for review in the near future. Some of its components, however, have already been accepted by the NRC as in the safety evaluation report [3] granted to Duke Power for their REA analysis and the recent safety evaluation report [4] granted to RETRAN-3D [5].

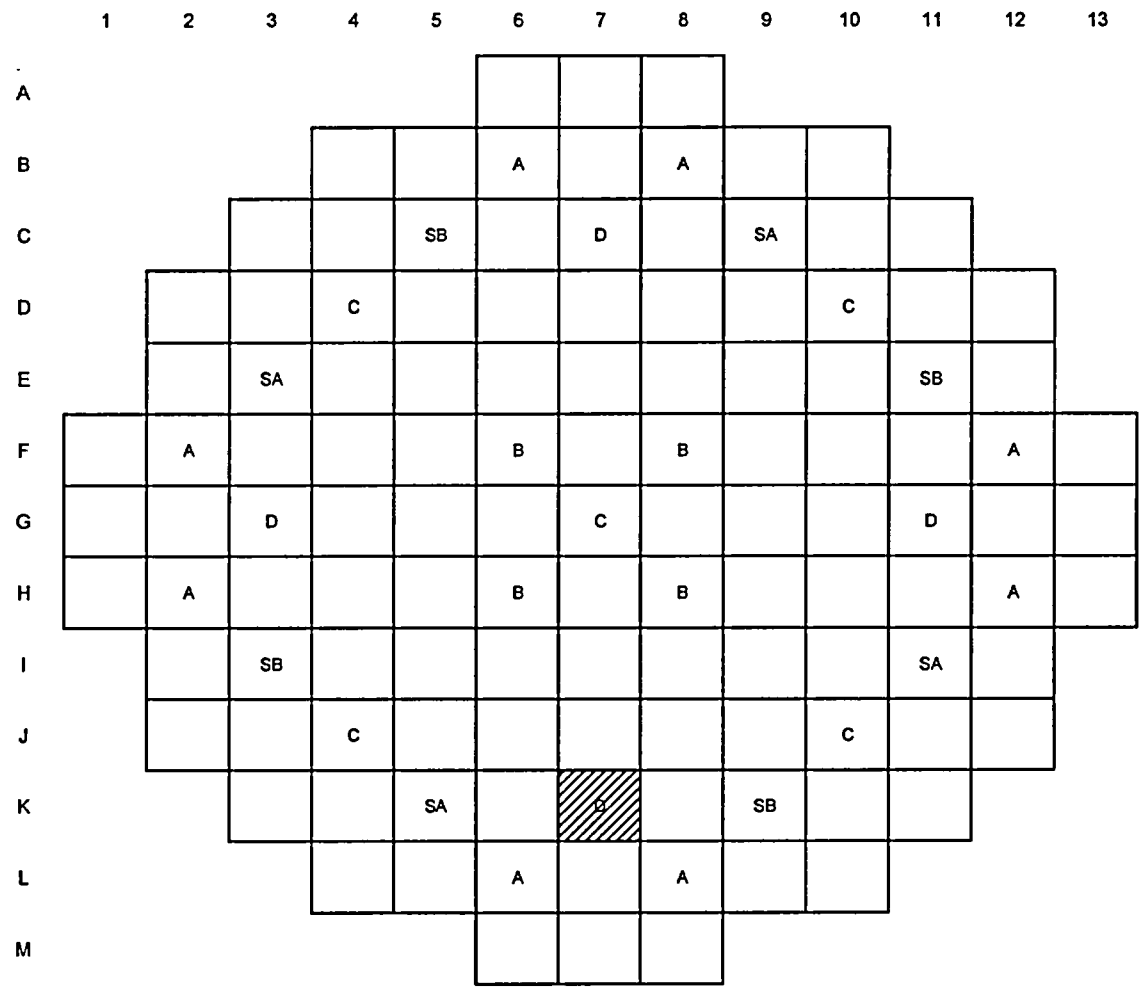
	1	2	3	4	5	6	7	8	9	10	11	12	13
A						55.695	55.842	55.684					
B				55.311	18.937	20.899	53.986	20.912	18.933	55.307			
C			49.126	22.317	45.500	49.270	25.591	49.258	45.490	22.294	49.122		
D		55.302	22.289	25.772	50.669	26.904	51.813	26.881	50.687	25.778	22.322	55.305	
E		18.932	45.489	50.682	51.565	48.012	48.094	48.009	51.572	50.680	45.507	18.946	
F	55.687	20.909	49.254	26.881	48.009	26.014	50.176	26.037	48.023	26.909	49.278	20.906	55.693
G	55.842	53.980	25.592	51.819	48.098	50.177	52.717	50.247	48.107	51.825	25.596	53.989	55.839
H	55.689	20.905	49.276	26.908	48.019	26.051	50.241	26.068	48.023	26.889	49.267	20.919	55.685
I		18.939	45.503	50.675	51.560	48.024	48.111	48.029	51.566	50.693	45.498	18.942	
J		55.316	22.322	25.781	50.689	26.888	51.831	26.913	50.685	25.787	22.300	55.301	
K			49.128	22.295	45.497	49.262	25.597	49.284	45.510	22.327	49.123		
L				55.296	18.941	20.916	53.982	20.911	18.948	55.311			
M						55.688	55.838	55.687					

Legend:

- first cycle
- second cycle
- third cycle
- third cycle (reinserted from Cycle 15)

Figure A-1
PI119 EOC Assembly Average Burnup Distribution

C 1



Insertion limtis at HZP:

bank SB - fully withdrawn
bank SA - fully withdrawn
bank A - fully withdrawn
bank B - 24% inserted
bank C - 82% inserted
bank D - fully inserted

predicted ejected rod worths:

bank	position	worth
C	G-7	$0.33
B	H-8	$0.09
C	J-10	$1.22
D	K-7	$1.28

Figure A-2
PI119 Control Rod Pattern at HZP Insertion Limits

CORETRAN calculates the fuel enthalpy on a nodal average basis, using a MATPRO correlation where fuel enthalpy is expressed as a function of fuel temperature. The same average fuel temperature (obtained from the volume-averaging of the fuel temperature resulting from the heat conduction solution over the fuel pellet) that is used for Doppler feedback purpose is therefore "expressed" as stored energy (cal/gm) within the fuel. In order to obtain the fuel enthalpy at the hottest pin and not the average pin, a small post-processing program was developed making use of the time-dependent nodal powers and pin-powers calculated by CORETRAN. Assuming an adiabatic condition, one can say that all the power developed within a pin during the transient is equivalent to a rise in stored energy; i.e. enthalpy. This approach leads to a relationship where

the ratio between the enthalpy rise and energy deposited/stored for a given pin is the same as the ratio for the average pin. The adiabatic approach is carried up to 0.5 sec into the transient, roughly corresponding to the point in time when the Doppler effect has controlled the power excursion. Based on the average time-constant for the heat to be conducted out of the fuel (1 to 2 seconds), the adiabatic assumption is a rather good approximation and yet conservative.

A.2 Description of Analysis

The initial conditions for the REA study are summarized in Table A-1 and, as identified in Figure A-2, the rod to be ejected is at position K7, with an ejected worth of $1.28. This is the "reference" case, no modification so far has been applied to the core model. Figure A-3 shows the average assembly power radial distribution at the onset of the rod ejection.

The rod is ejected in 0.1 seconds, assuming a constant ejection speed. No consideration is given to trip signals since, by the time control rods are scrammed into the core, the real "bang" of the transient has already taken place, with Doppler being responsible to control the power excursion. Peak fuel enthalpies occur well before the effect of inserted control rods. The transient is simulated over a 1 second period.

In order to establish the sensitivity of the "reference" case to key specific parameters (ejected rod worth, Doppler temperature coefficient, and moderator temperature coefficient), a series of extra REA simulations were performed where the "reference" case is slightly modified and the resulting peak enthalpy rise is then compared against the reference value Δcal/gm. This was a way of quantifying the effects of uncertainties in these key parameters and their effect upon the Δcal/gm results. By the use of specific multiplicative factors, the CORETRAN user can request changes to ejected rod worth, Doppler temperature coefficient, and moderator temperature coefficient, among other core integral parameters. Each of the three key parameters was adversely changed by 10%. Also performed was a "conservative" case REA simulation where all three key parameters are modified at the same time.

Table A-1
PI119 Core Initial Conditions

Parameter	Condition	Value
Time-in-cycle	EOC	500 EFPD
Power level	HZP	1.0 MWatt
Temperature	HZP T-inlet	547°F
Core flow	100% design flow	73.43 E6 lbm/hr
Control rods	HZP rod insertion limit	Bank D: 0 swd, Bank C: 41 swd, Bank B: 173 swd
Boron	Critical	349 ppm
Xenon	No xenon	

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						0.313	0.329	0.313					
B				0.439	1.087	1.146	0.685	1.145	1.086	0.438			
C			0.507	1.195	1.125	1.027	0.795	1.026	1.124	1.193	0.506		
D		0.439	1.196	1.036	1.082	1.555	1.107	1.555	1.079	1.032	1.191	0.437	
E		1.088	1.126	1.082	1.149	1.313	1.274	1.311	1.146	1.078	1.121	1.082	
F	0.313	1.146	1.027	1.556	1.313	1.579	1.150	1.575	1.308	1.549	1.022	1.140	0.311
G	0.329	0.684	0.795	1.106	1.273	1.149	0.642	1.145	1.269	1.102	0.791	0.682	0.328
H	0.312	1.144	1.025	1.552	1.310	1.574	1.145	1.571	1.306	1.548	1.021	1.140	0.311
I		1.085	1.122	1.079	1.145	1.308	1.268	1.306	1.142	1.075	1.119	1.081	
J		0.438	1.191	1.032	1.077	1.550	1.102	1.546	1.075	1.029	1.188	0.436	
K			0.505	1.191	1.121	1.022	0.791	1.020	1.118	1.187	0.504		
L				0.437	1.083	1.141	0.681	1.138	1.080	0.436			
M						0.311	0.328	0.311					

Legend:

- power > 1.4
- 1.2 < power < 1.4
- 1.0 < power < 1.2
- 0.5 < power < 1.0
- power < 0.5

Figure A-3
PI119 Initial Normalized Assembly Radial Power Distribution

C2

Finally, under a statistical analysis approach, a SRSS combination of uncertainties methodology is applied to the Δcal/gm results of the three sensitivity analyses plus a fourth value which is a 10% increase on the results of the "reference" case to account for the uncertainty in the prediction of the power peaking. The result of the statistical calculation is an alternative to the result of the "conservative" case in which the uncertainties in the key parameters are applied simultaneously. The Δcal/gm results are converted to peak enthalpy values by adding the initial enthalpy value of 20 cal/gm.

A.3 Results

Table A-2 summarizes the results from the sensitivity studies that were performed, including CORETRAN's prediction for Fq and FΔH. Note that the hot pin fuel enthalpy rise is less than 10% higher than the corresponding nodal average fuel enthalpy rise. The pulse widths were calculated at the half-height of the power pulses. The MTC perturbed case was not run (historically: one did not expect to see that much of an effect; instead, one decided to observe the compound effect of modifying both ERW and DTC). The effect of singly modifying the MTC can be inferred by comparing against the "conservative" case.

Table A-2
PI119 REA Sensitivity Cases

Cases	$\Delta\rho_{rod}$ $	Δcal/gm Nodal	Δcal/gm Pin	Peak cal/gm	Fq	FΔH	Pulse Width (msec)
Reference Case	1.28	22	24	44	7.38	3.93	56
+10% rod worth	1.40	30	32	52	8.19	4.21	40
-10% Doppler	1.28	24	26	46	7.40	3.93	56
+10% rod worth, -10% Doppler	1.40	32	35	55	8.21	4.22	40
+10% rod worth, -10% Doppler, -10% MTC	1.40	33	36	56	8.23	4.21	39

Figure A-4 shows the power excursion observed for the "reference" case. Figure A-5 is the resulting nodal fuel enthalpy rise also observed from the "reference case". Figure A-6 shows how the radial power was distributed at the time the power peaked during the "reference" cases simulation. As expected, there is a major shift of power towards the area where the rod is ejected.

Figures A-7 & A-8 are provided in order to compare the "reference case against the "conservative" case. Figure A-9 shows the radial power distribution at the time of the peak power for the "conservative" case. The concentration of power around the ejected rod is now even more prevalent, as expected.

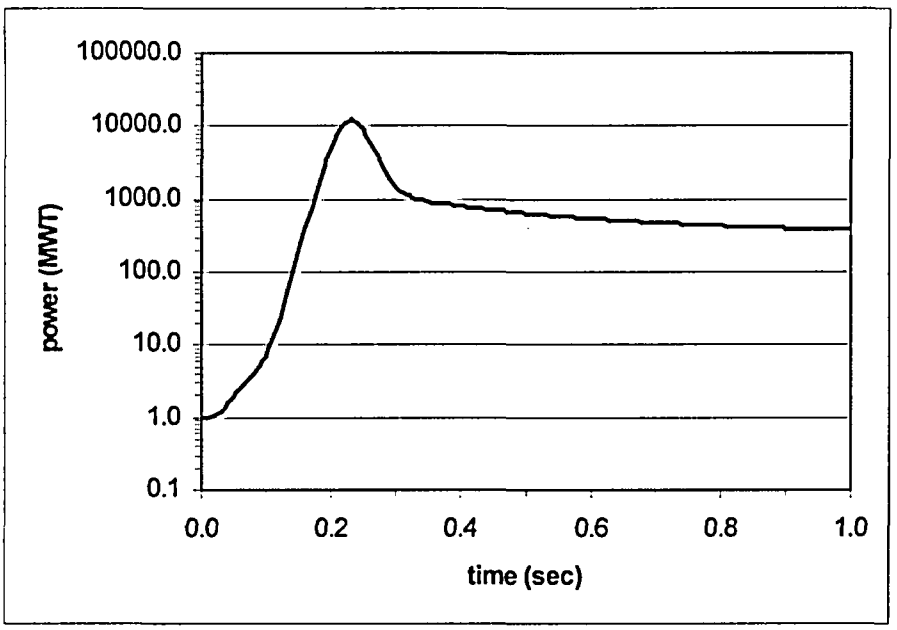


Figure A-4
Power Excursion for the "Reference" REA Case

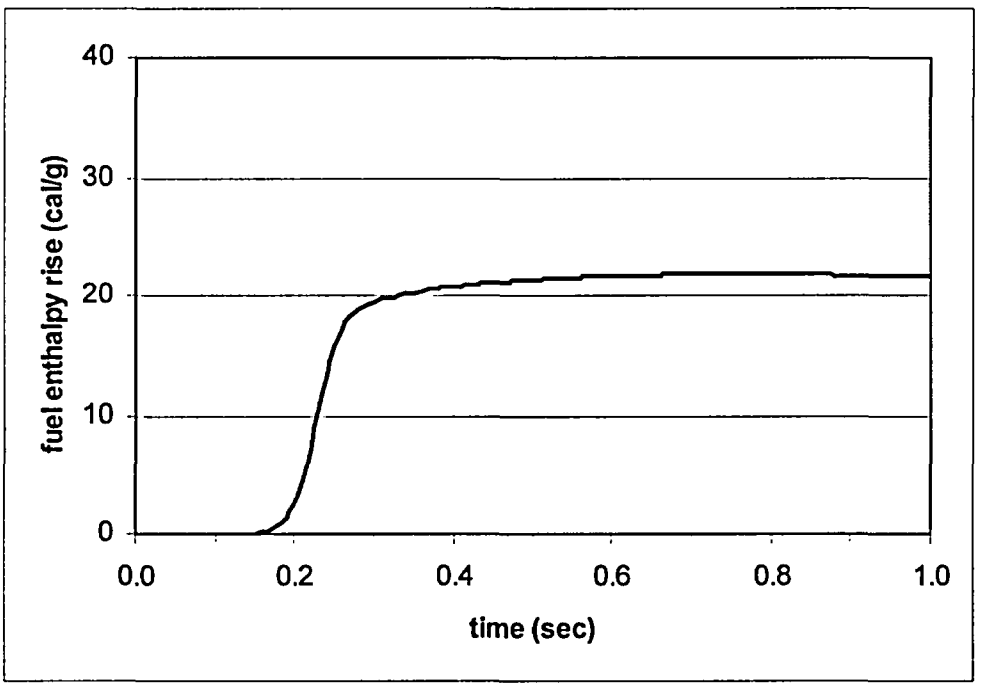


Figure A-5
Peak Nodal Fuel Enthalpy Rise for the "Reference" REA Case

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						0.165	0.173	0.165					
B				0.235	0.566	0.601	0.365	0.601	0.565	0.234			
C			0.277	0.627	0.600	0.560	0.438	0.560	0.599	0.626	0.277		
D		0.269	0.681	0.545	0.611	0.899	0.648	0.898	0.610	0.544	0.679	0.268	
E		0.692	0.707	0.678	0.717	0.813	0.791	0.813	0.716	0.677	0.705	0.690	
F	0.224	0.793	0.719	1.106	0.929	1.053	0.774	1.051	0.926	1.104	0.717	0.791	0.223
G	0.259	0.543	0.640	0.918	1.062	0.945	0.502	0.943	1.060	0.916	0.638	0.542	0.258
H	0.271	1.003	0.937	1.474	1.301	1.571	1.235	1.569	1.299	1.472	0.935	1.001	0.270
I		1.021	1.109	1.154	1.400	1.805	1.851	1.804	1.398	1.151	1.107	1.019	
J		0.458	1.289	1.184	1.637	2.843	2.307	2.842	1.636	1.182	1.287	0.457	
K			0.639	1.729	2.001	2.389	3.361	2.388	1.999	1.727	0.639		
L				0.748	2.106	2.698	1.918	2.696	2.104	0.747			
M						0.745	0.830	0.744					

Legend:

- power > 2.0
- 1.5 < power < 2.0
- 1.0 < power < 1.5
- 0.5 < power < 1.0
- power < 0.5

Figure A-6
PI119 Normalized Assembly Radial Power Distribution @ "Reference" Case Peak Power

C3

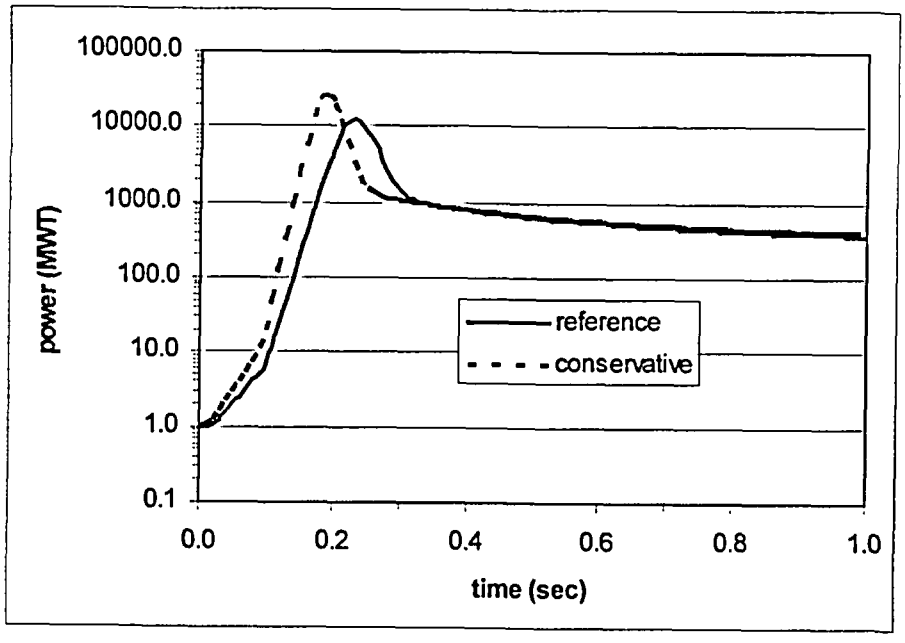


Figure A-7
Power Excursion for REA Cases

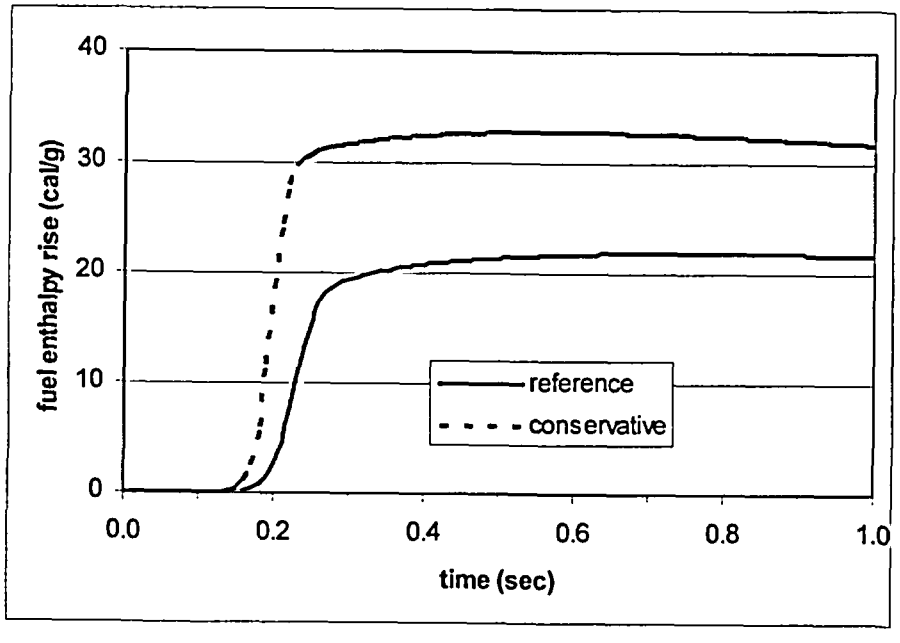


Figure A-8
Peak Nodal Fuel Enthalpy Rise for REA Cases

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						0.162	0.171	0.163					
B				0.232	0.560	0.595	0.362	0.595	0.559	0.231			
C			0.274	0.622	0.595	0.556	0.435	0.556	0.595	0.621	0.274		
D		0.265	0.674	0.542	0.608	0.894	0.645	0.893	0.607	0.540	0.672	0.265	
E		0.684	0.701	0.674	0.715	0.811	0.790	0.811	0.713	0.673	0.699	0.682	
F	0.220	0.785	0.714	1.101	0.927	1.053	0.774	1.052	0.924	1.098	0.712	0.783	0.220
G	0.256	0.539	0.635	0.915	1.062	0.947	0.504	0.945	1.059	0.913	0.634	0.537	0.256
H	0.267	0.993	0.931	1.469	1.301	1.578	1.241	1.576	1.300	1.467	0.929	0.991	0.267
I		1.011	1.103	1.152	1.403	1.813	1.862	1.812	1.401	1.149	1.101	1.009	
J		0.454	1.282	1.185	1.642	2.862	2.335	2.861	1.641	1.183	1.280	0.453	
K			0.636	1.725	2.005	2.415	3.458	2.414	2.004	1.723	0.636		
L				0.745	2.105	2.715	1.945	2.713	2.104	0.745			
M						0.748	0.837	0.747					

Legend:

- power > 2.0
- 1.5 < power < 2.0
- 1.0 < power < 1.5
- 0.5 < power < 1.0
- power < 0.5

Figure A-9
PI119 Normalized Assembly Radial Power Distribution @ "Conservative" Case Peak Power

C4

Figures A-10 & A-11 show together all the simulations performed for this study. It is rather clear that the most significant parameter is the Ejected Rod Worth, followed by the Doppler Temperature Coefficient and the Moderator Temperature Coefficient, in that order.

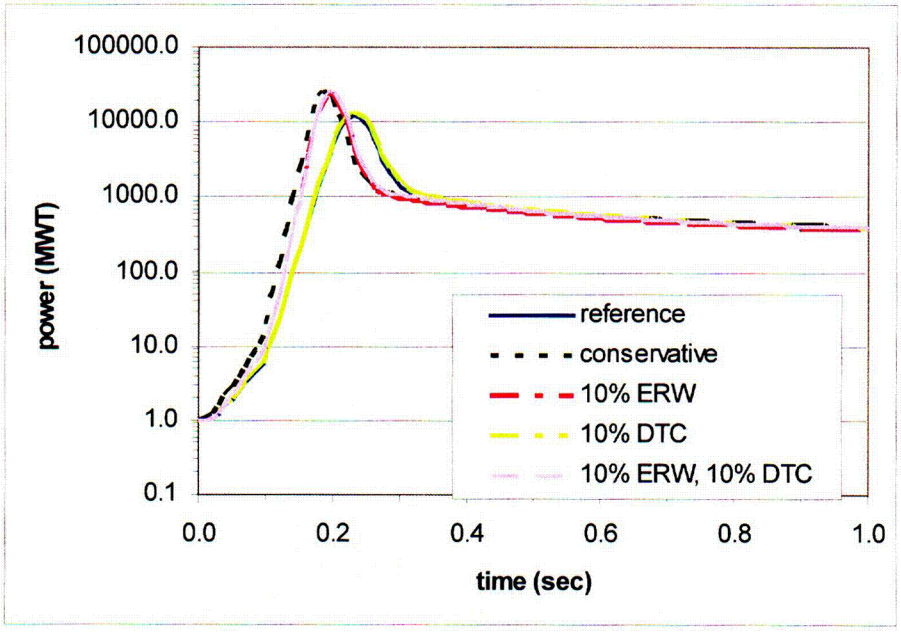


Figure A-10
Power Excursion for Sensitivity REA Cases

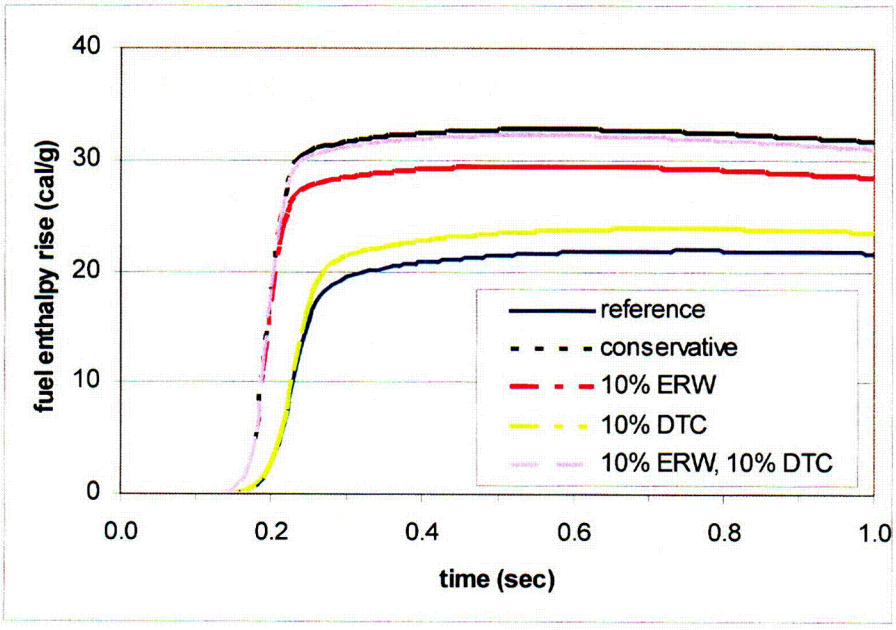


Figure A-11
Peak Nodal Fuel Enthalpy Rise for Sensitivity REA Cases

C5

The statistical approach combines the change (i.e. Δ/Δcal/gm), relative to the "reference" case result of 24 Δcal/gm (44 cal/gm), in the Δcal/gm results for the three sensitivity analyses, along with the +10% peaking uncertainty value based on the reference case result, using the SRSS combination of uncertainties methodology. This statistical result is then compared to the results of the "conservative" case (36 Δcal/gm / 56 cal/gm) to determine the potential margin gain with the statistical approach.

Result of +10% ERW sensitivity case = 32 Δcal/gm (Δ/Δcal/gm = 8)

Result of –10% DTC sensitivity case = 26 Δcal/gm (Δ/Δcal/gm = 2)

Result of –10% MTC sensitivity case = 25 Δcal/gm (Δ/Δcal/gm = 1) ⇐ (*inferred-see above)*

Result of +10% increase in F_q in reference case = (Δ/Δcal/gm = 24 x 1.1 = 2.4)

SRSS (Δ/Δcal/gm) = SQRT ($8^2 + 2^2 + 1^2 + 2.4^2$) = 9 Δ/Δcal/gm

Statistical result = reference case result + SRSS uncertainty = 24 + 9 = 33 Δcal/gm (53 cal/gm)

The results of the statistical analysis approach indicate that a reduction in the "conservative" REA result from 36 to 33 Δcal/gm (56 to 53 cal/gm) can be demonstrated.

A.4 References

1. "*CORETRAN-01: A Three-Dimensional Program for Reactor Physics and Thermal-Hydraulic Analysis,*" EPRI WO-3574, Revision 2, Electric Power Research Institute (Dec. 1999)

2. D.L. Hagrman, et al., "MATPRO – Version 11 (Revision 2): A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior", Idaho Falls, Idaho, Idaho National Engineering Laboratories, NUREG/CR-0497 TREE-1280, Revision 2 (Aug. 1981)

3. Safety Evaluation Report on Topical Report DPC-NE-3001, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters", U. S. NRC, November 15, 1991

4. Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "*RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems*", U. S. NRC, January 25, 2001.

5. "*RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems*", NP-7450(A), Revision 5, EPRI, July 2001

B
DEMONSTRATION ANALYSIS FOR TWO-LOOP WESTINGHOUSE/CE REACTOR

B.1 Overview

The Westinghouse/CE Rod Ejection Accident (REA) demonstration analysis was performed for a 2815 Mw(th) System 80 plant. These plants employ 177 fuel assemblies having a 16 X 16 fuel matrix, and have an active fuel length of 150 inches. The control rods that are allowed to be inserted in Mode 1 and 2 are comprised of four large B_4C fingers that are inserted into guide tubes that displace a 2 x 2 region of fuel. An equilibrium fuel cycle was modeled. Figure B-1 shows the fuel management layout.

All control rods that are allowed to be at least partially inserted at the HZP PDIL (Power Dependent Insertion Limit) are assumed to be fully inserted in the demonstration analysis. This both maximizes the worth of the ejected rod and eliminates the sensitivity of ejected worth to axial power shape that would occur if control rods were modeled as partially inserted. The locations of the control banks that were inserted in the demonstration calculation are shown in Figure B-1. Note that not all of the banks shown in this figure are actually allowed by the PDIL to be fully inserted at Hot Zero Power (HZP). Also, an additional control rod bank that actually is not allowed to be inserted at HZP was inserted near the core center. These control rods were added to ensure that future fuel managements were enveloped, and are shown in gray in Figure B-1. This extra bank increases ejected rod worth and peaking, ensures that the ejected rod is located towards the core periphery creating a greater power distribution asymmetry, and forces the highest power assembly to be adjacent to the ejected rod location.

The location of the highest worth control rod is shown in yellow on Figure B-1. The nominal worth of the highest worth ejected rod is approximately 0.42 $\%\Delta\rho$. This worth, in conjunction with a minimum value of β_{eff} (EOL) results in a nominal ejected rod worth < 1$, and consequently there would be no prompt power excursion. To accommodate the possibility of higher ejected rod worth in future cycles, the worth of the ejected rod for the Reference Case was increased to 0.64 $\%\Delta\rho$ by increasing control rod absorption cross sections. To accommodate future higher burnup cycles, the value of β_{eff} was also reduced to 0.0044, increasing the ejected rod worth of the Reference Case to $1.45.

The REA was performed using the HERMITE multi-dimensional space-time code. This code is described in detail in Reference 1. Reference 2 is the NRC issued SER approving HERMITE for solving the few-group transient diffusion equations. HERMITE has been referenced in a number of license applications and topical reports, including support for Reference 3, the C-E control rod methodology topical report. The HERMITE model employed four neutronic nodes per fuel assembly and 24 planes axially. Two neutron groups were employed with cross sections generated by the DIT code described in Reference 4.

					J* 1	K 2 1	J 3	K 4 1	J* 5					
			J* 6	K 7	K* 8 2	J* 9	K* 10 3	J* 11	K* 12 2	K 13	J* 14			
		J* 15 3	K/ 16	J 17	J* 18	K* 19	I 20	K* 21	J* 22	J 23	K/ 24	J* 25 3		
	J* 26	K/ 27	I* 28 4	K* 29	I* 30 P	I* 31	J 32 5	I* 33	I* 34 P	K* 35	I* 36 4	K/ 37	J* 38	
	K 39	J 40	K* 41	I/ 42	J/ 43	K* 44	I 45	K* 46	J/ 47	I/ 48	K* 49	J 50	K 51	
J* 52	K* 53 2	J* 54	I* 55 P	J/ 56	K* 57	I 58 1	K* 59	I 60 1	K* 61	J/ 62	I* 63 P	J* 64	K* 65 2	J* 66
K 67 1	J* 68	K* 69	I* 70	K* 71	I 72 1	J* 73 1	J* 74 1	J* 75 1	I 76 1	K* 77	I* 78	K* 79	J* 80	K 81 1
J 82	K* 83 3	I 84	J 85 5	I 86	K* 87	J* 88 1	I/ 89 4	J* 90 1	K* 91	I 92	J 93 5	I 94	K* 95 3	J 96
K 97 1	J* 98	K* 99	I* 100	K* 101	I 102 1	J* 103 1	J* 104 1	J* 105 1	I 106 1	K* 107	I* 108	K* 109	J* 110	K 111 1
J* 112	K* 113 2	J* 114	I* 115 P	J/ 116	K* 117	I 118 1	K* 119	I 120 1	K* 121	J/ 122	I* 123 P	J* 124	K* 125 2	J* 126
	K 127	J 128	K* 129	I/ 130	J/ 131	K* 132	I 133	K* 134	J/ 135	I/ 136	K* 137	J 138	K 139	
	J* 140	K/ 141	I* 142 4	K* 143	I* 144 P	I* 145	J 146 5	I* 147	I* 148 P	K* 149	I* 150 4	K/ 151	J* 152	
		J* 153 3	K/ 154	J 155	J* 156	K* 157	I 158	K* 159	J* 160	J 161	K/ 162	J* 163 3		
			J* 164	K 165	K* 166 2	J* 167	K* 168 3	J* 169	K* 170 2	K 171	J* 172			
					J* 173	K 174 1	J 175	K 176 1	J* 177					

5	Lead Regulating Bank
4	Second Regulating Bank
3	Third Regulating Bank
2	Fourth Regulating Bank

P	PSR Group
	Worst Ejected Rod
1	Bank 1 (This bank is not allowed to be inserted at HZP)

K **Once Exposed Fuel**
J **Twice Exposed Fuel**
I **Thrice Exposed Fuel**

Figure B-1
Fuel Loading and Control Rod Pattern

Q6

HERMITE contains options for both open and closed channel modeling of the coolant channels. Since little heat is transferred from the fuel rod to the coolant during the rapid initial part of the control rod ejection transient, the simpler closed-channel option was employed. Prompt heat-up of the coolant by gamma absorption and neutron interactions results in some (second order effect) T-H feedback, which serves to mitigate the control rod ejection transient. Such mitigating feedback has been neglected in the Reference calculation.

As described more fully in Reference 1, HERMITE incorporates a simplified fuel rod mode that solves the one-dimensional radial heat conduction equation for the temperature distribution at each axial node within a fuel pin. Power dependent gas-gap conductivity is input to this calculation. Sensitivity studies have shown that the change in enthalpy during the control rod ejection event is insensitive to the precise conductivity used. A conservative conductivity (200 BTU/Hr-ft^2-°F), somewhat lower than the minimum value calculated at beginning of cycle, was used. The resulting fuel temperatures are then used to calculate initial and final hot rod enthalpies (radial averaged enthalpy at he hot spot). Since the 3-D HERMITE calculates directly only nodal powers, an allowance is added to the calculation of Δcal/gm to including local (pin-to-node) peaking. This allowance, as well as other adjustments to the enthalpy increase, is discussed in more detail below.

B.2 Description of Analysis

The demonstration analysis was performed from hot zero power (HZP) initial conditions. HZP was selected because of the increased control rod insertion allowed by the PDIL. End-of-Cycle (EOC) was selected to minimize the value of β_{eff}; the combination of HZP and EOC results in the highest ejected rod worth (expressed in $). Other initial conditions are shown in Table B-1. The Reference calculation was performed for equilibrium, full-power xenon conditions (simulating a rapid power reduction from HFP); the xenon free condition (simulating an ejection after long term operation at HZP) also was addressed. Figure B-2 shows the initial axial power distribution. A highly top-skewed axial shape was selected, reflective of the tendency of the axial shape to shift towards the core top at HZP. Figure B-3 provides the radial power distribution prior to control rod ejection.

Table B-1
Core Initial Conditions Reference Case

Parameter	Condition	Value
Time-in-Cycle	EOC	See Note 1
Power Level	HZP Critical	1.0 E-5 MwTh
Temperature	HZP T-inlet	565 °F
100% Power	-----	2800 MwTh
Core Flow	Design Flow	-----
Boron	-----	0 ppm
Xenon	Equilibrium Xe	----
Control Rods	Augmented HZP rod insertion limit	See Note 2

Notes:

EOC is more adverse because of the lower value of β_{eff}. An artificially low value of β_{eff}, 0.0044, was employed to envelop extremely high burnup cores.

In order to envelop future operating cycles, control rod worth was increased, rods were inserted beyond the allowed zero power insertion limit, and extra banks were added. See text.

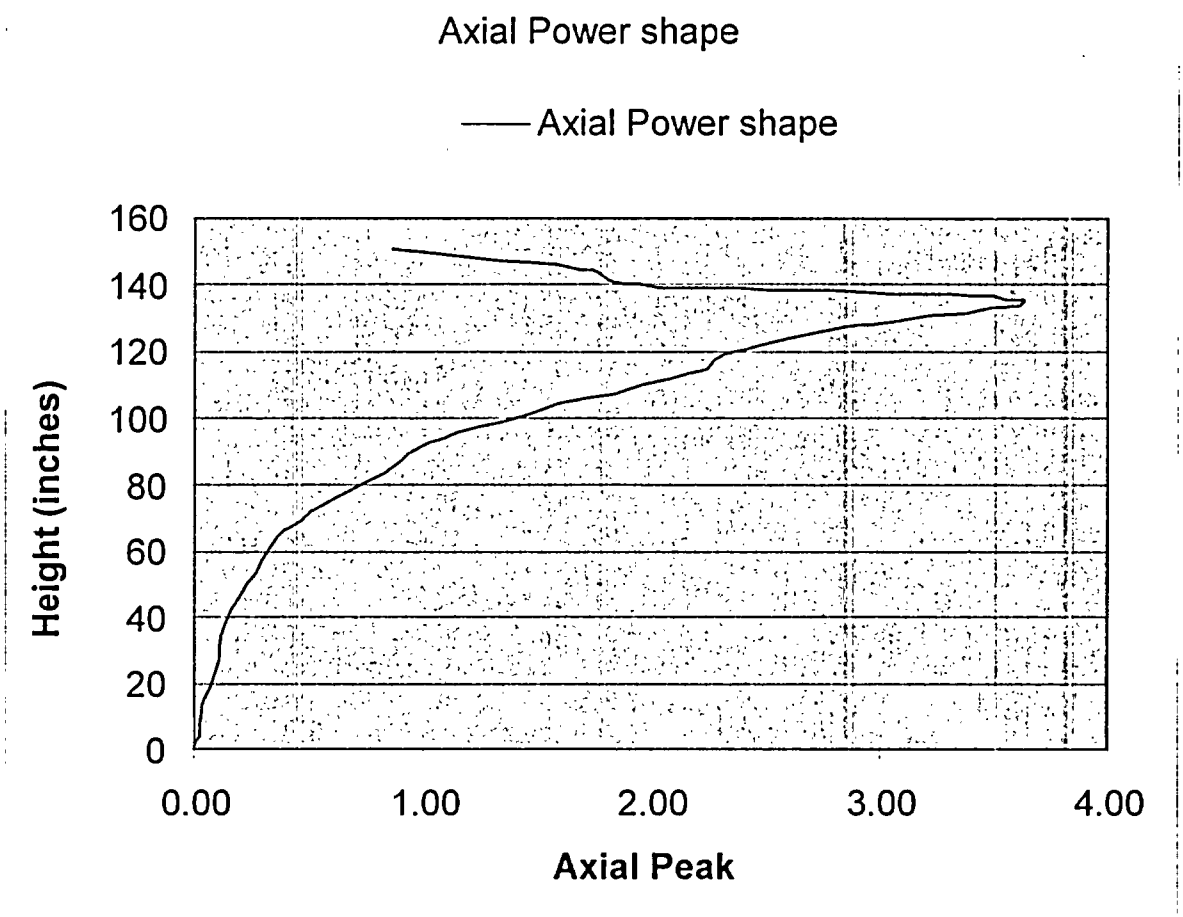


Figure B-2
Axial Power Shape

					J* 1 .2833	K 2 .3185	J 3 .3467	K 4 .3185	J* 5 .2835					
			J* 6 .6344	K 7 1.0408	K* 8 .7268	J* 9 .7128	K* 10 .6226	J* 11 .7130	K* 12 .7272	K 13 1.0409	J* 14 .6343			
		J* 15 .4251	K/ 16 1.2421	J 17 1.4218	J* 18 1.1798	K* 19 1.2947	I 20 .8526	K* 21 1.2944	J* 22 1.1805	J 23 1.4217	K/ 24 1.2412	J* 25 .4245		
	J* 26 .6355	K/ 27 .1.2434	I* 28 .8997	K* 29 1.7135	I* 30 1.2221	I* 31 1.1204	J 32 .8986	I* 33 1.1201	I* 34 1.2218	K* 35 1.7123	I* 36 .8984	K/ 37 1.2406	J* 38 .6338	
	K 39 1.0431	J 40 1.4245	K* 41 1.7153	I/ 42 1.3778	J/ 43 1.5920	K* 44 1.6433	I 45 1.1820	K* 46 1.6418	J/ 47 1.5908	I/ 48 1.3758	K* 49 1.7115	J 50 1.4204	K 51 1.0399	
J* 52 .2843	K* 53 .7290	J* 54 1.1833	I* 55 1.2245	J/ 56 1.5938	K* 57 1.6421	I 58 .8614	K* 59 1.1015	I 60 .8605	K* 61 1.6397	J/ 62 1.5904	I* 63 1.2211	J* 64 1.1790	K* 65 .7264	J* 66 .2831
K 67 .3195	J* 68 .7152	K* 69 1.2983	I* 70 1.1233	K* 71 1.6461	I 72 .8625	J* 73 .4027	J* 74 .3305	J* 75 .4022	I 76 .8609	K* 77 1.6428	I* 78 1.1202	K* 79 1.2946	J* 80 .7128	K 81 .3186
J 82 .3481	K* 83 .6250	I 84 .8559	J 85 .9020	I 86 1.1864	K* 87 1.1054	J* 88 .3314	I/ 89 .1785	J* 90 .3307	K* 91 1.1023	I 92 1.1829	J 93 .8993	I 94 .8532	K* 95 .6230	J 96 .3470
K 97 .3201	J* 98 .7163	K* 99 1.3010	I* 100 1.1259	K* 101 1.6513	I 102 .8654	J* 103 .4042	J* 104 .3317	J* 105 .4032	I 106 .8622	K* 107 1.6448	I* 108 1.1221	K* 109 1.2965	J* 110 .7141	K 111 .3189
J* 112 .2848	K* 113 .7309	J* 114 1.1865	I* 115 1.2291	J/ 116 1.6012	K* 117 1.6513	I 118 .8663	K* 119 1.1072	I 120 .8640	K* 121 1.6445	J/ 122 1.5948	I* 123 1.2247	J* 124 1.1831	K* 125 .7287	J* 126 .2841
	K 127 1.0472	J 128 1.4306	K* 129 1.7245	I/ 130 1.3871	J/ 131 1.6046	K* 132 1.6544	I 133 1.1887	K* 134 1.6494	J/ 135 1.5958	I/ 136 1.3799	K* 137 1.7169	J 138 1.4253	K 139 1.0434	
	J* 140 .6385	K/ 141 1.2502	I* 142 .9061	K* 143 1.7288	I* 144 1.2359	I* 145 1.1300	J 146 .9042	I* 147 1.1250	I* 148 1.2256	K* 149 1.7171	I* 150 .9011	K/ 151 1.2446	J* 152 .6359	
		J* 153 .4281	K/ 154 1.2526	J 155 1.4355	J* 156 1.1924	K* 157 1.3058	I 158 .8581	K* 159 1.3005	J* 160 1.1837	J 161 1.4254	K/ 162 1.2447	J* 163 .4258		
			J* 164 .6402	K 165 1.0508	K* 166 .7340	J* 167 .7190	K* 168 .6267	J* 169 .7162	K* 170 .7294	K 171 1.0437	J* 172 .6360			
					J* 173 .2861	K 174 .3210	J 175 .3490	K 176 .3201	J* 177 .2844					

Ejected Rod Location

Figure B-3
Pre-Ejection Power Map

The highest worth control rod, which is shown in Figure B-1 was assumed to be ejected from the full in position. The assumption that the control rod is fully inserted at the time of ejection maximizes the worth of the ejected rod. It also eliminates the sensitivity of ejected rod worth to axial power shape that occurs with partially inserted control rods. The control rod is assumed to be ejected over a period of 0.05 seconds; this is much more rapid than the physical time for ejection. The reactor trip was not modeled. Modeling of the reactor trip was unnecessary since, as illustrated in Figure B-5, peak power occurs prior to the time when control rods can act to mitigate the event (due to processing times, time for holding coil decay, and time for insertion, no credit is typically taken for reactor trip during the initial two seconds of the transient).

C7

As noted above, the Reference Case employed a rod worth increased by about 50% and a reduced β_{eff} to envelop future operating cycles. Control rod worth was increased by increase the control rod absorption cross section, while β_{eff} was reduced by uniformly reducing the six-group β_{eff}'s input to HERMITE. Enthalpy increase was found to be approximately linear with control rod worth ($), as is illustrated in Figure B-4. Increasing rod worth by increasing control rod cross section has a somewhat greater effect on enthalpy increase than does decreasing β_{eff} because the former also increases core peaking.

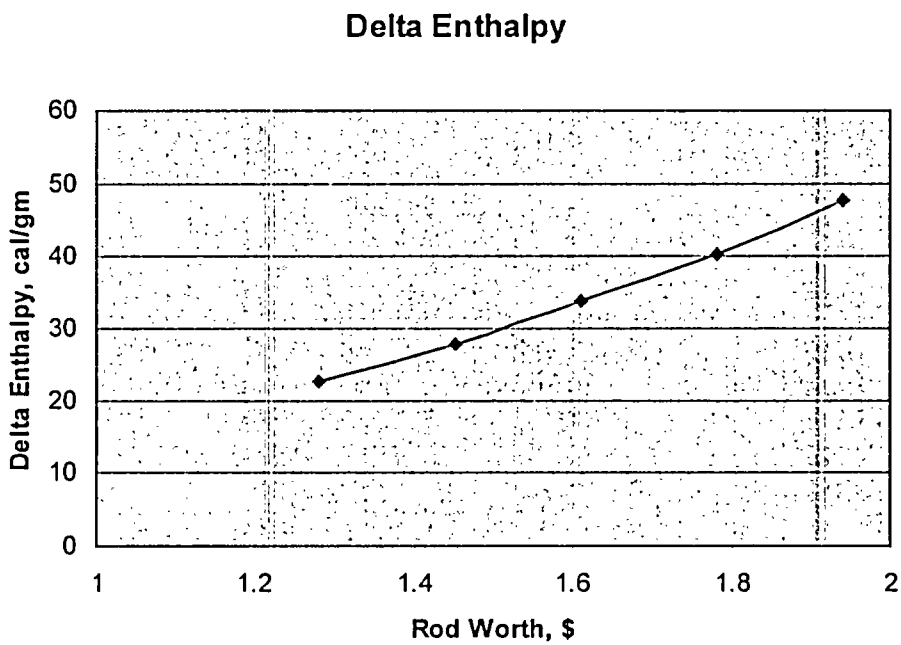
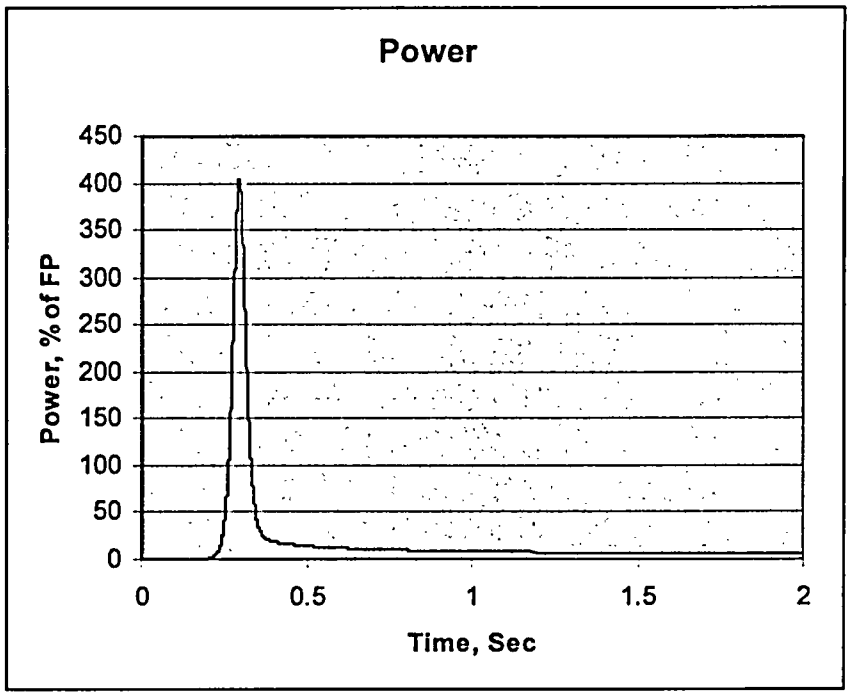


Figure B-4
Enthalpy vs. Ejected Rod Worth



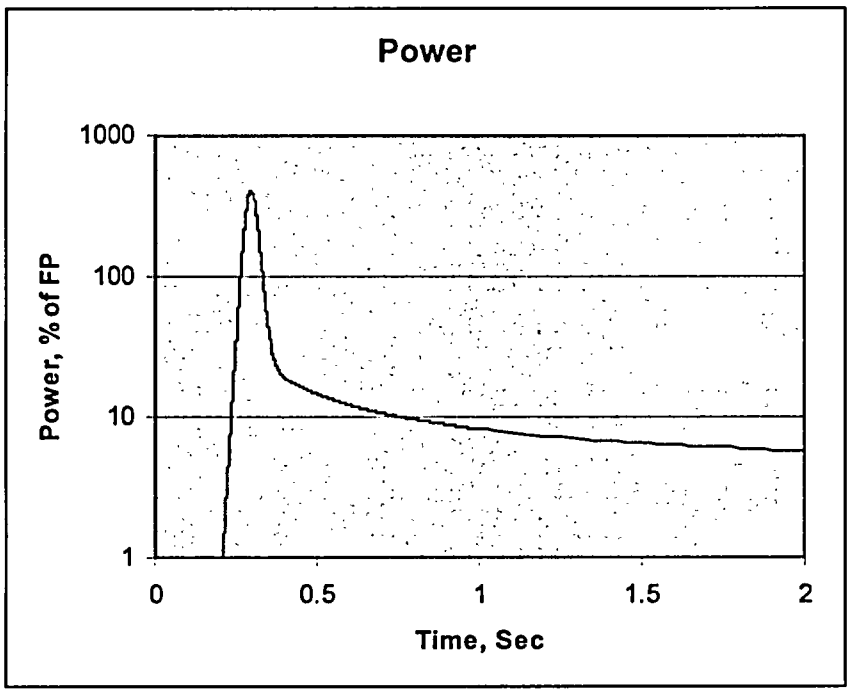


Figure B-5
Core Average Power vs Time - Reference Case

The sensitivity of the Reference Case to key physics parameters was also established. Each of these sensitivity analyses represents a quantification of the effect on enthalpy due to uncertainties in physics parameters. The capability to apply multiplicative factors to cross sections in HERMITE was employed to alter ejected rod worth, moderator temperature coefficient (MTC),

and fuel Doppler temperature coefficient (DTC) to the desired values. Specifically, the sensitivity to rod worth, MTC and DTC were established by changing each of these parameters by ±10%. The sensitivity to xenon was also determined by running cases with both zero and equilibrium xenon distributions. The REA sensitivity cases and conditions are listed in Table B-2. In addition, a "conservative" case was run that deterministically combines all sensitivity factors.

Table B-2
Demonstration Analysis Cases

Description	Notes
Reference Case	EWR=$1.45
+10% ejected rod worth (ERW) uncertainty sensitivity	Actually, 11% increase (EWR=$1.61)
-10% DTC uncertainty sensitivity	
-10% MTC uncertainty sensitivity	No significant effect
Xenon sensitivity	Change from equilibrium to no xenon
Conservative Case (+10% ERW, -10% DTC, no xenon, 10% power distribution uncertainty	Uncertainties deterministically included in calculation

B.3 Results

The results of the REA demonstration analysis are provided in Table B-3 and Figures B-5 through B-9. Figure B-5 shows the neutron power transient for the Reference Case. Peak power of about 400% occurs after approximately 300 msec. The pulse width is approximately 45 msec. Power is seen to decrease to about 6% after 2 sec. This near asymptotic power level is relatively low because of the low value of gap conductance employed (the power would return to zero for a true adiabatic heatup). The post-ejection radial power distribution is shown in Figure B-6. Note that the peak power location does not occur in the most highly irradiated fuel, but rather in adjacent first exposure fuel. Most appropriately, an acceptance criterion based on burnup should be employed and compared to the enthalpy change of specific fuel assembly. However, for simplification, only the largest enthalpy change is calculated for comparison to the most restrictive enthalpy acceptance criteria. Figure B-7 shows the change in enthalpy of the peak pin (radial averaged enthalpy at the hot spot) with time for the Reference Case. As this figure illustrates, the peak enthalpy change of 29.1 cal/gm essentially is reached two seconds into the transient.

Figure B-8 compares the peak powers for the various cases listed in Table B-3. As this figure illustrates, only the case with increased ejected rod worth (and the "Conservative" case with includes the increased ERW) exhibit a significant increase in peak power relative to the reference case. The "Conservative" case is seen to reach a peak power of 879% of Full Power after 229 msec. The pulse width is approximately 32 msec. Figure B-9 compares the change in enthalpy for each of the cases. Again, only those cases with increased ejected rod worth exhibit a significant increase in enthalpy relative to the reference case. The change in peak pin enthalpy for the "Conservative" case is 45.3 cal/gm.

Table B-3
Demonstration Analysis Results

Case	Peak Power (%FP)	Time of Peak Power (sec)	ERW (% Δρ)	ERW ($)	F_q	Max Δcal/gm (Node)	Max Δcal/gm (Pin)	Pulse Width (msec)
Reference	404	0.295	0.638	1.45	14.5	27.70	29.09	45
+10% ERW	732	0.233	0.710	1.61	14.8	33.70	35.38	33
-10% DTC	456	0.294	0.638	1.45	14.6	30.72	32.26	45
No Xenon	450	0.286	0.638	1.45	14.6	28.87	30.32	43
+10% Peaking	404	0.295	0.638	1.45	15.2	30.47	32.00	45
Conservative	879	0.229	0.710	1.61	16.7	39.18	45.26	32

The statistical approach combines the change in enthalpy (Δ/Δcal/gm), relative to the reference case result of 30 Δcal/gm, with the sensitivity established for each of the key parameters identified in Table B-3 using the SRSS (square Root of the Sum of the Squares) combination of uncertainties methodology. The components of the uncertainty analysis are summarized below:

Reference Case	29.1 Δcal/gm	
+10% ERW	35.4 Δcal/gm	(Δ/Δcal/gm = 6.3)
-10% DTC	32.3 Δcal/gm	(Δ/Δcal/gm = 3.2)
No xenon	30.3 Δcal/gm	(Δ/Δcal/gm = 1.2)
+10% Peaking	32.0 Δcal/gm	(Δ/Δcal/gm = 2.9)

$$\sigma = \sqrt{6.3^2 + 3.2^2 + 1.2^2 + 2.9^2} = 7.7\, cal/gm$$

Statistical result = Reference Case result + uncertainty

= 29.2 + 7.7 = 36.8 Δcal/gm

The results of the statistical analysis indicate that a reduction in the conservative REA result from 45.3 to 36.8 Δcal/gm (61 to 56 cal/gm) can be demonstrated.

					J* 1 .5921	K 2 .5597	J 3 .4841	K 4 .3502	J* 5 .2593					
			J* 6 1.8700	K 7 2.7698	K* 8 1.6213	J* 9 1.2792	K* 10 .8673	J* 11 .7563	K* 12 .6267	K 13 .7725	J* 14 .4411			
		J* 15 1.4639	K/ 16 4.1748	J 17 3.9788	J* 18 2.6612	K* 19 2.3345	I 20 1.1733	K* 21 1.3325	J* 22 .9884	J 23 1.0334	K/ 24 .8360	J* 25 .2638		
	J* 26 1.8715	K/ 27 4.1767	I* 28 3.9803	K* 29 4.9071	I* 30 2.6781	I* 31 1.9663	J 32 1.2015	I* 33 1.1113	I* 34 .9901	K* 35 1.2066	I* 36 .5713	K/ 37 .7212	J* 38 .3522	
	K 39 2.7730	J 40 3.9828	K* 41 4.9097	I/ 42 3.3434	J/ 43 3.1775	K* 44 2.7274	I 45 1.5180	K* 46 1.5794	J/ 47 1.2340	I/ 48 .9188	K* 49 1.0260	J 50 .7942	K 51 .5610	
J* 52 .5934	K* 53 1.6239	J* 54 2.6660	I* 55 2.6809	J/ 56 3.1792	K* 57 2.9919	I 58 1.3688	K* 59 1.3694	I 60 .7901	K* 61 1.1699	J/ 62 .9874	I* 63 .6901	J* 64 .6249	K* 65 .3713	J* 66 .1392
K 67 .5603	J* 68 1.2812	K* 69 2.3372	I* 70 1.9685	K* 71 2.7282	I 72 1.3690	J* 73 .5833	J* 74 .3790	J* 75 .3172	I 76 .5229	K* 77 .9116	I* 78 .5806	K* 79 .6331	J* 80 .3365	K 81 .1464
J 82 .4843	K* 83 .8681	I 84 1.1736	J 85 1.2014	I 86 1.5180	K* 87 1.3702	J* 88 .3792	I/ 89 .1533	J* 90 .1924	K* 91 .5499	I 92 .5612	J 93 .4070	I 94 .3700	K* 95 .2641	J 96 .1448
K 97 .3508	J* 98 .7570	K* 99 1.3344	I* 100 1.1129	K* 101 1.5826	I 102 .7917	J* 103 .3180	J* 104 .1927	J* 105 .1865	I 106 .3620	K* 107 .6691	I* 108 .4407	K* 109 .4965	J* 110 .2708	K 111 .1210
J* 112 .2595	K* 113 .6274	J* 114 .9893	I* 115 .9919	J/ 116 1.2370	K* 117 1.1736	I 118 .5246	K* 119 .5514	I 120 .3626	K* 121 .6333	J/ 122 .5941	I* 123 .4421	J* 124 .4178	K* 125 .2553	J* 126 .1006
	K 127 .7737	J 128 1.0353	K* 129 1.2098	I/ 130 .9224	J/ 131 .9924	K* 132 .9155	I 133 .5628	K* 134 .6706	J/ 135 .5944	I/ 136 .4885	K* 137 .5886	J 138 .4782	K 139 .3469	
	J* 140 .4420	K/ 141 .8382	I* 142 .5736	K* 143 1.0320	I* 144 .6959	I* 145 .5842	J 146 .4085	I* 147 .4416	I* 148 .4424	K* 149 .5886	I* 150 .3007	K/ 151 .4082	J* 152 .2074	
		J* 153 .2648	K/ 154 .7250	J 155 .7994	J* 156 .6299	K* 157 .6371	I 158 .3715	K* 159 .4978	J* 160 .4180	J 161 .4782	K/ 162 .4082	J* 163 .1384		
			J* 164 .3543	K 165 .5647	K* 166 .3739	J* 167 .3387	K* 168 .2653	J* 169 .2714	K* 170 .2555	K 171 .3470	J* 172 .2074			
					J* 173 .1403	K 174 .1472	J 175 .1454	K 176 .1214	J* 177 .1006					

Ejected Rod

Hot Box

Figure B-6
Post-Ejected Power Distribution

C8

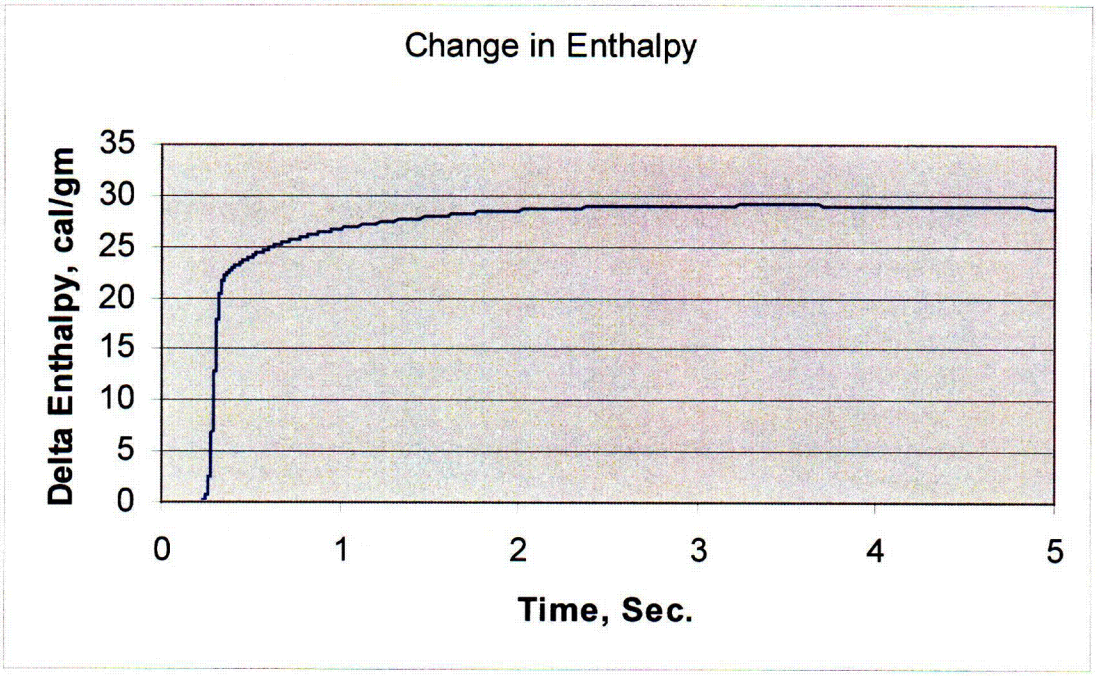


Figure B-7
Change in Enthalpy - Reference Case

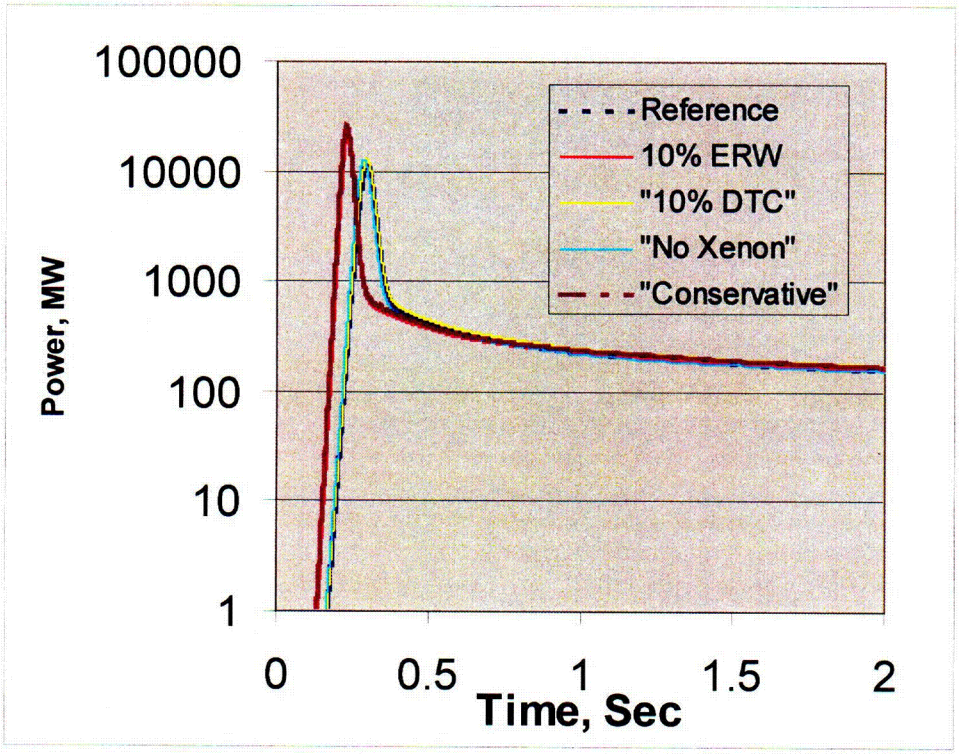


Figure B-8
Core Average Neutron Power

C9

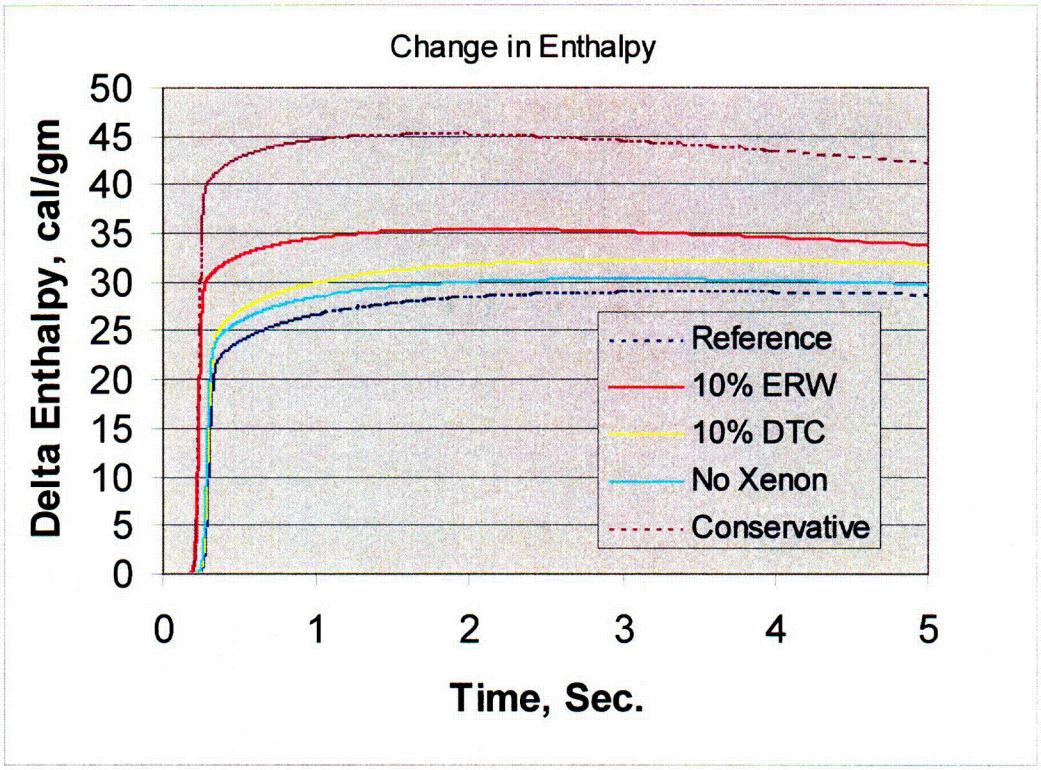


Figure B-9
Maximum 3D Δ cal/gm

B.4 References

1. "HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients, March 1976, CENPD-188-A.

2. Letter, Olan D. Parr (NRC) to A. E. Scherer (CE), "Staff Evaluation of CENPD-188," June 1976.

3. "C-E Method for Control Element Assembly Ejection Analysis." CENPD-190-A, January 1976.

4. "The ROCS & DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983.

C10

C
DEMONSTRATION ANALYSIS FOR FOUR-LOOP WESTINGHOUSE REACTOR

C.1 Overview

The Westinghouse four-loop plant REA 3D Δcal/gm demonstration analysis is performed for the Catawba Unit 2 Cycle 11 (C2C11) core design by Duke Power Company. Catawba Unit 2 is a Westinghouse four-loop NSSS plant rated at 3411 MWth. The core consists of 193 17x17 fuel assemblies. The C2C11 core is a transition (mixed) core consisting of 72 Westinghouse RFA fuel assemblies and 121 Framatome ANP Mark-BW fuel assemblies. The C2C11 loading pattern is a ring-of-fire low leakage design as depicted in Figure C-1. It is designed to achieve a maximum cycle burnup of 510 EFPD. The REA demonstration analysis is performed at end-of-cycle (EOC) hot zero power (HZP) conditions. Figure C-1 shows the two-dimensional (2D) assembly average exposures at 510 EFPD. Note that the nominal EOC HZP conditions for C2C11 do not yield sufficient ejected rod worth to achieve the desired prompt critical power response for the demonstration of the REA 3D analysis methodology. The nominal C2C11 EOC HZP ejected rod worth and beta-effective are 413 pcm and 0.00514, respectively. Therefore a multiplier on the ejected control rod cross sections is applied to obtain sufficient ejected rod worth to demonstrate the methodology. For C2C11, a multiplier of 1.18 yields an ejected rod worth of 680 pcm ($1.32) for the REA "reference" case.

The Studsvik-Scandpower SIMULATE-3K (S3K) code (Reference 1) is a 3D transient neutronic version of the SIMULATE-3P code. S3K uses the QPANDA full two-group nodal spatial model developed in SIMULATE-3P, with the addition of six delayed neutron groups. The program employs a fully-implicit time integration of the neutron flux, delayed neutron precursor, and heat conduction models. Beta-effective is fully functionalized similar to other cross sections to provide an accurate value of beta-effective for the time varying neutron flux. The S3K kinetics code is approved by the NRC for REA analyses in Reference 2. The REA analysis is performed using Version 1.26 of the S3K code.

The S3K thermal-hydraulic model includes spatial heat conduction and hydraulic channel models. The heat conduction model solves the conduction equation on a multi-region mesh in cylindrical coordinates. Temperature-dependent values may be employed for the heat capacity, thermal conductivity, and gap conductance. A single characteristic pin conduction calculation is performed consistent with the radial neutronic node geometry, with an optional calculation of the peak pin behavior to represent a hot rod model. A single characteristic hydraulic channel calculation is performed based on the radial neutronic node geometry. The S3K enthalpy calculation is performed by solving for a radial temperature distribution and integrating across the pellet to obtain the average enthalpy. The S3K REA demonstration analysis employs a multiplier on the predicted convection heat transfer coefficient at the cladding surface to appropriately limit heat transfer during the time period of interest.

In addition to calculating the core thermal response and the peak fuel pellet enthalpy within the S3K code, the EPRI VIPRE-01 thermal-hydraulic subchannel analysis code (Reference 3) is also used with transient core power and peaking factor boundary conditions from the S3K analyses. This hot rod analysis uses the Duke Power NRC-approved VIPRE model and methodology for REA fuel rod thermal analyses as described in Reference 4.

	R-	P-	N-	M-	L-	K-	J-	H-	G-	F-	E-	D-	C-	B-	A-
1					41.281	41.574	33.701	46.348	33.682	41.52	41.278				
2			43.101	43.427	19.482	21.861	22.108	35.584	22.102	21.862	19.485	43.432	43.107		
3		43.107	18.98	23.022	42.242	45.373	45.56	23.905	45.551	45.375	42.246	23.025	18.98	43.101	
4		43.432	23.025	41.499	25.004	41.741	24.79	43.929	24.769	41.756	25.027	41.499	23.022	43.427	
5	41.278	19.485	42.246	25.027	44.426	24.895	45.315	24.425	45.309	24.925	44.426	25.004	42.242	19.482	41.281
6	41.52	21.862	45.375	41.756	24.925	45.858	45.029	44.558	45.031	45.858	24.895	41.741	45.373	21.861	41.574
7	33.682	22.102	45.551	24.769	45.309	45.031	22.967	43.878	22.967	45.029	45.315	24.79	45.56	22.108	33.701
8	46.348	35.584	23.905	43.929	24.425	44.558	43.878	50.321	43.878	44.558	24.425	43.929	23.905	35.584	46.348
9	33.701	22.108	45.56	24.79	45.315	45.029	22.967	43.878	22.967	45.031	45.309	24.769	45.551	22.102	33.682
10	41.574	21.861	45.373	41.741	24.895	45.858	45.031	44.558	45.029	45.858	24.925	41.756	45.375	21.862	41.52
11	41.281	19.482	42.242	25.004	44.426	24.925	45.309	24.425	45.315	24.895	44.426	25.027	42.246	19.485	41.278
12		43.427	23.022	41.499	25.027	41.756	24.769	43.929	24.79	41.741	25.004	41.499	23.025	43.432	
13		43.101	18.98	23.025	42.246	45.375	45.551	23.905	45.56	45.373	42.242	23.022	18.98	43.107	
14			43.107	43.432	19.485	21.862	22.102	35.584	22.108	21.861	19.482	43.427	43.101		
15					41.278	41.52	33.682	46.348	33.701	41.574	41.281				

First Cycle

Second Cycle

Third Cycle

Fourth Cycle

Figure C-1
C2C11 EOC Assembly Average Burnup Distribution

C11

C.2 Description of Analysis

Initial conditions for the REA demonstration analyses are described in Table C-1. These conditions along with the previously described ejected rod worth multiplier and convection heat transfer coefficient multiplier define the "reference" case. The initial core radial power distribution for the reference case is provided in Figure C-2.

The control rod is ejected at core location D-12 in 0.1 seconds. The S3K excore power range detector model is used to generate a reactor trip signal on high flux low setpoint. The trip signal is generated when the 3rd highest excore detector indicates 42% full power, which is the 25% technical specification setpoint plus 17% for measurement uncertainty and margin. A 0.5 second time delay between reaching the reactor trip setpoint and the start of control rod insertion is assumed. The duration of the analysis is 3 seconds, which is sufficient to obtain all of the analysis results of interest.

Three "sensitivity" analyses to the reference case are performed for the key physics parameters to obtain the results necessary for the statistical analysis approach. Each of these sensitivity analyses represents a quantification of the effect on the Δcal/gm result due to the uncertainty in the physics parameter. S3K has the capability to apply multiplicative factors to physics parameter cross sections to drive ejected rod worth (ERW), moderator temperature coefficient (MTC), and fuel Doppler temperature coefficient (DTC) to desired values. Each of these physics parameters are changed by 10% in the conservative direction for the three sensitivity cases. The REA sensitivity cases and conditions are listed in Table C-2. In addition, the results of a "conservative" case that combines all three sensitivity factors are given. The Δcal/gm results are converted to peak enthalpy values by adding the initial enthalpy value of 16 cal/gm.

Table C-1
C2C11 Core Initial Conditions

Parameter	Condition	Value
Time-in-cycle	EOC	510 EFPD
Power level	HZP critical	1.0 E-7
Temperature	HZP T-inlet	561°F
Core flow	78% design flow (3 RCPs)	106.31 E6 lbm/hr
Control rods	HZP rod insertion limit	Bank D @ 0 swd, Bank C @ 47 swd
Boron	Critical	356 ppm
Xenon	No xenon	

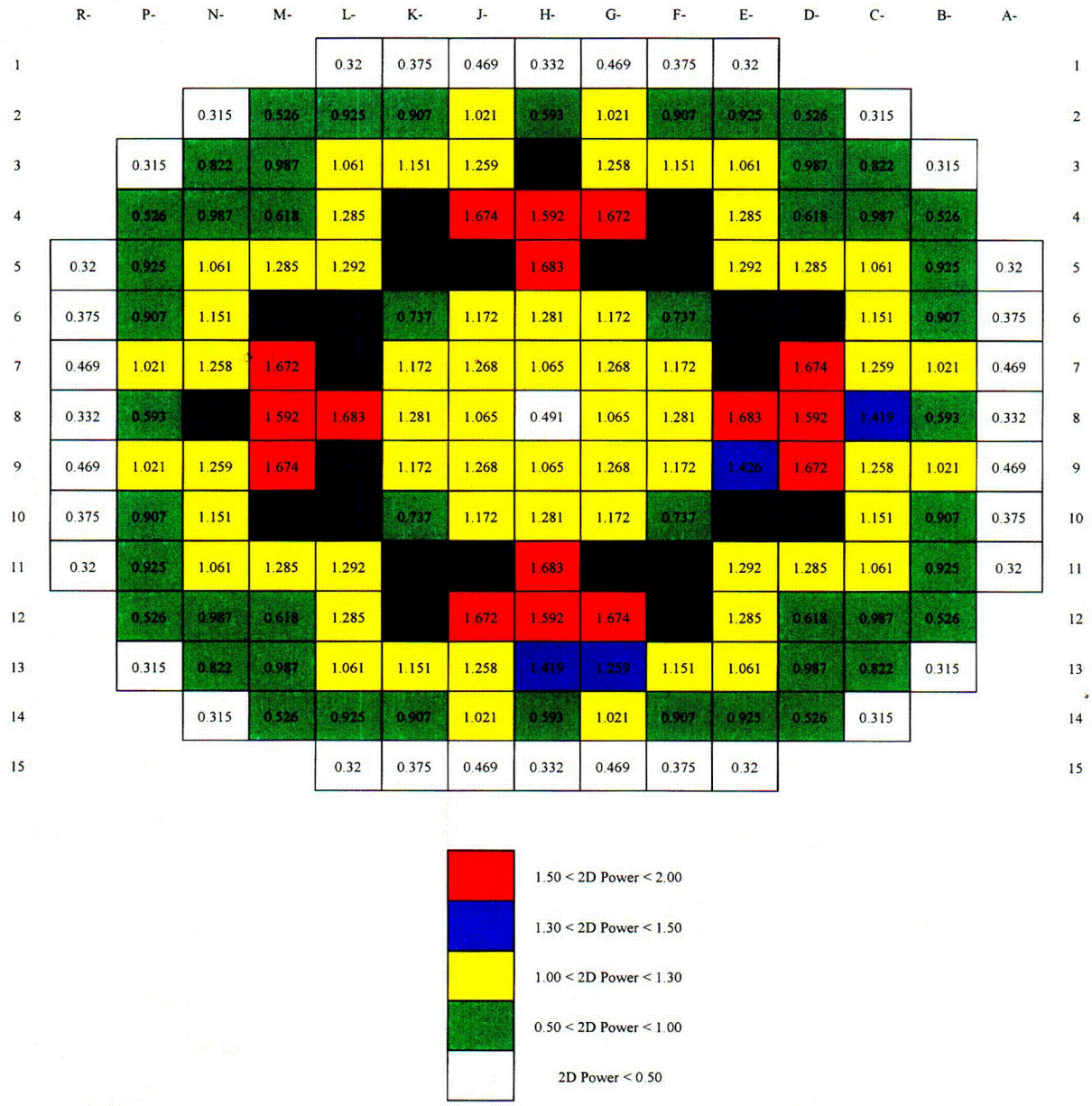


Figure C-2
Initial Conditions Assembly Radial Power Distribution

Table C-2
C2C11 Demonstration Analysis Cases

Description	S3K Cross Section Multiplier
Reference case	RODOUT=1.18
+10% ejected rod worth (ERW) uncertainty sensitivity	RODOUT=1.212
-10% MTC uncertainty sensitivity	RODOUT=1.18, TMO=0.9
-10% DTC uncertainty sensitivity	RODOUT=1.18, TFU=0.9
Conservative case (+10% ERW, -10% MTC, & DTC)	RODOUT=1.212, TMO & TFU =0.9

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The hot rod analysis using the VIPRE-01 code is a conservative model and methodology. A multi-subchannel model of the hot rod and adjacent fuel and non-fuel rods is driven with transient radial peaking factors corresponding to the hot pin from the S3K REA analysis. The axial power shape is assumed to remain at the initial shape for simplicity, which has been shown to be a conservative modeling approach. The gas gap conductivity remains at the initial value to maximize the pellet enthalpy. The VIPRE code then calculates a peak average fuel temperature as a function of time for the hot spot on the hot rod during the REA. This transient temperature result is then converted to enthalpy units using the equation from the MATPRO manual (Reference 5).

The statistical analysis approach applies the SRSS combination of uncertainties methodology to the Δcal/gm results of the three sensitivity analyses plus a fourth value which is a 10% increase on the results of the reference case to account for the uncertainty in the prediction of the power peaking. The result of the statistical calculation is an alternative to the result of the conservative case in which the uncertainties in the key physics parameters are applied simultaneously.

C.3 Results

The results of the REA demonstration analyses are provided in the following tables and figures. Table C-3 provides a summary of the REA demonstration case results for the following parameters:

- Case – description of each analysis (see Table C-2)
- Peak Power Time (sec) – time of peak neutron power
- Trip Signal Time (sec) – time that the trip setpoint is reached
- Peak Power (%FP) – peak neutron power
- RHO ($) – ejected rod worth in dollars
- RHO (ΔK/K) – ejected rod worth in ΔK/K units
- Core Average Moderator Temperature (°F) – maximum core average moderator temperature
- Core Average Fuel Temperature (°F) – maximum core average fuel temperature
- F_DEL-H – maximum radial pin power
- F-Q – maximum total power
- Max Δcal/gm (Pin) – maximum Δcal/g on a pin basis
- Max Δcal/gm (Node) – maximum Δcal/g on a nodal basis
- Peak cal/gm – peak pellet total cal/gm
- Pulse Width (msec) – full-width/half height based on core average power

Table C-3
C2C11 Demonstration Analysis Results Case Summary

Case	Peak Power Time (sec)	Trip Signal Time (sec)	Peak Power (% FP)	RHO ($)	RHO (DK/K)	Core Avg. Moderate Temperature (F)
Reference	0.325	0.291	549	1.32	6.80E-03	564
10% ERW	0.263	0.236	1031	1.45	7.50E-03	565
10% DTC	0.326	0.291	602	1.32	6.80E-03	564
10% MTC	0.324	0.291	557	1.32	6.80E-03	564
Conservative	0.265	0.236	1148	1.45	7.50E-03	565

Case	Core Avg. Fuel Temperature (F)	F_DEL-H	F-Q	Max Δcal/gm (Pin)	Max Δcal/gm (Node)	Peak cal/gm	Pulse Width (msec)
Reference	629	5.26	13.03	23	22	39	39
10% ERW	643	5.72	14.26	30	28	46	28
10% DTC	628	5.26	13.03	26	24	42	39
10% MTC	631	5.26	13.03	24	23	40	39
Conservative	643	5.72	14.26	33	31	49	28

Figure C-3 shows the neutron power transient response for the reference and conservative cases. The peak power for the reference case is 549% and occurs at 0.32 seconds. The peak power for the conservative case is 1148% and occurs at 0.26 seconds. Figure C-4 shows the radial fuel assembly (2D) and total nodal (3D) power distributions at the peak power statepoint for the reference case. Figure C-5 shows the radial fuel assembly and total nodal power distributions for the peak power statepoint for the conservative case. Figure C-6 plots the maximum 3D Δcal/gm transient response on a pin basis for the reference and conservative cases. The maximum 3D pellet average enthalpy increase is 23 Δcal/gm (39 cal/gm) for the reference case and 33 Δcal/gm (49 cal/gm) for the conservative case.

Figure C-7 shows the core average neutron power response for the reference case, the conservative case, and the three sensitivity cases. The peak power results for the sensitivity cases are 1031% for the +10% ERW case, 602% for the -10%DTC case, and 557% for the -10%MTC case (Table C-3). Figure C-8 shows the same results as Figure C-7 but the axes have been reduced so that the REA pulse shape and width can be illustrated. The pulse widths are given in Table C-3 and range from 28 to 39 milliseconds. Figure C-9 shows the 3D maximum delta-enthalpy results for all cases. These results show the ejected rod worth to be the only significant sensitive parameter in the 3D REA analysis.

The VIPRE-01 hot rod analysis result for the conservative case is shown in Figure C-10. The maximum value is 35 Δcal/gm (51 cal/gm). This compares well with the S3K result of

33 Δcal/gm (49 cal/gm) for the conservative case. The trend of the temperature prediction also compares well with the S3K result. It can be concluded that the fuel rod and pellet thermal models in S3K are an acceptable method for calculating the Δcal/gm result for the REA. However, if the cal/gm result predicted by S3K approaches the regulatory acceptance limit, then the more detailed VIPRE-01 hot rod model should be used to confirm that the analysis results meet the acceptance limit.

The statistical approach combines the change (i.e. Δ/Δcal/gm), relative to the reference case result of 24 Δcal/gm, in the Δcal/gm results for the three sensitivity analyses, along with the +10% peaking uncertainty value based on the reference case result, using the SRSS combination of uncertainties methodology. This statistical result is then compared to the results of the conservative case (33 Δcal/gm) to determine the potential margin gain with the statistical approach.

Result of +10% ERW sensitivity case = 30 Δcal/gm (Δ/Δcal/gm = 6)

Result of –10% DTC sensitivity case = 26 Δcal/gm (Δ/Δcal/gm = 2)

Result of –10% MTC sensitivity case = 25 Δcal/gm (Δ/Δcal/gm = 1)

Result of +10% increase in reference case peaking = (Δ/Δcal/gm = 24 x 1.1 = 2.4)

SRSS (Δ/Δcal/gm) = SQRT ($6^2 + 2^2 + 1^2 + 2.4^2$) = 7 Δ/Δcal/gm

Statistical result = reference case result + SRSS uncertainty = 24 + 7 = 31 Δcal/gm (47 cal/gm)

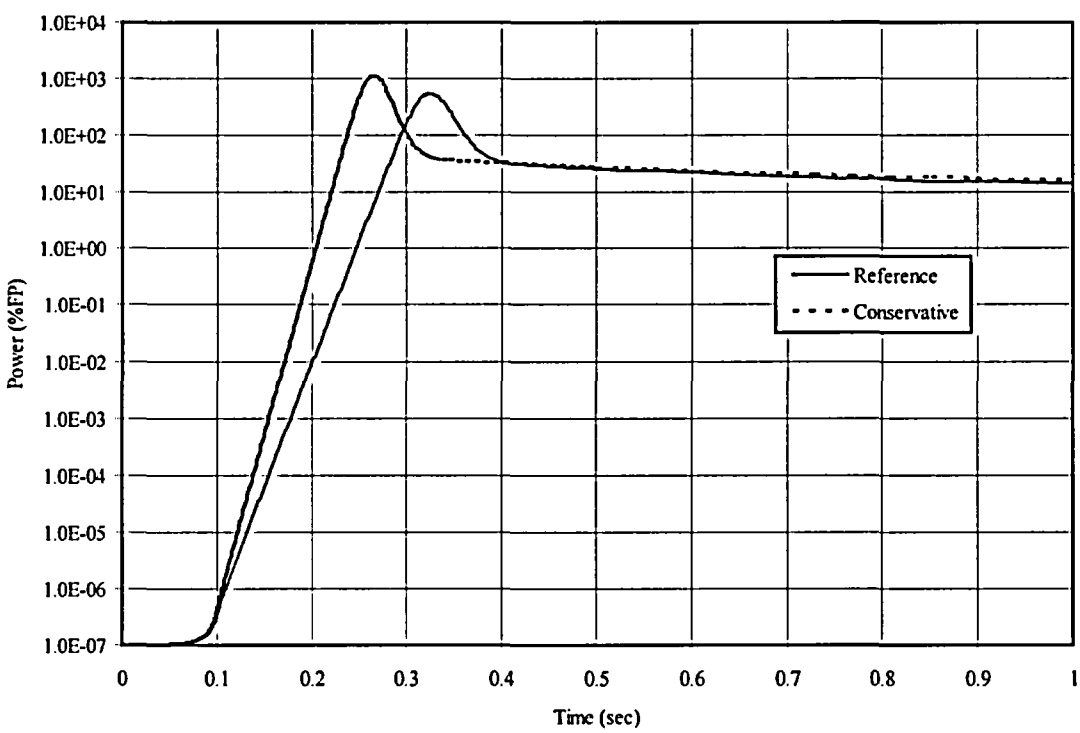

Figure C-3
C2C11 Core Average Neutron Power

	R-	P-	N-	M-	L-	K-	J-	H-	G-	F-	E-	D-	C-	B-	A-	
1					0.109 0.953	0.132 1.22	0.173 1.731	0.131 2.004	0.199 2.659	0.168 3.008	0.151 4.124					1
2			0.103 0.216	0.175 0.353	0.316 0.557	0.321 0.414	0.383 0.681	0.238 0.427	0.442 0.778	0.416 0.535	0.455 0.813	0.279 0.565	0.18 0.375			2
3		0.103 0.216	0.269 0.552	0.331 0.681	0.371 0.773	0.423 0.873	0.492 1.05	0.594 1.241	0.566 1.203	0.559 1.152	0.555 1.158	0.556 1.134	0.499 1.004	0.202 0.414		3
4		0.175 0.353	0.331 0.681	0.215 0.468	0.465 1.012	0.536 1.21	0.682 1.51	0.699 1.602	0.797 1.756	0.738 1.653	0.742 1.598	0.394 0.843	0.672 1.34	0.374 0.726		4
5	0.109 0.185	0.316 0.557	0.371 0.773	0.465 1.012	0.486 1.116	0.571 1.274	0.619 1.445	0.795 1.805	0.742 1.72	0.842 1.855	0.869 1.961	0.952 2.021	0.822 1.654	0.738 1.237	0.263 0.428	5
6	0.132 0.2	0.321 0.413	0.423 0.873	0.536 1.21	0.571 1.274	0.318 0.7	0.566 1.331	0.686 1.623	0.716 1.669	0.528 1.145	1.168 2.558	1.221 2.684	1.033 2.062	0.826 1.02	0.346 0.509	6
7	0.172 0.295	0.383 0.681	0.491 1.049	0.681 1.507	0.618 1.445	0.566 1.331	0.688 1.566	0.669 1.568	0.964 2.177	1.052 2.444	1.411 3.23		1.338 2.779	1.083 1.88	0.501 0.841	7
8	0.131 0.229	0.238 0.427	0.594 1.241	0.699 1.602	0.795 1.805	0.686 1.623	0.669 1.568	0.382 0.862	1.004 2.338	1.378 3.213				0.784 1.369	0.449 0.768	8
9	0.199 0.336	0.442 0.778	0.567 1.204	0.798 1.758	0.742 1.721	0.716 1.669	0.964 2.177	1.004 2.338	1.407 3.152			2.657 5.749			0.808 1.337	9
10	0.168 0.254	0.416 0.535	0.559 1.152	0.738 1.652	0.842 1.854	0.528 1.145	1.052 2.444	1.378 3.213		1.196 2.592	2.807 6.187	3.009 6.725	2.539 5.245		0.792 1.229	10
11	0.151 0.262	0.455 0.813	0.555 1.158	0.742 1.598	0.869 1.962	1.168 2.559	1.41 3.228			2.805 6.182	3.341 7.648	4.126 9.062	3.308 7.158	2.592 4.916	0.825 1.518	11
12		0.279 0.565	0.556 1.134	0.394 0.843	0.953 2.022	1.22 2.683			2.659 5.755	3.008 6.723	4.124 9.058	4.602 10.423	4.191 9.067			12
13		0.18 0.375	0.499 1.004	0.672 1.34	0.822 1.654	1.032 2.061	1.337 2.777			2.539 5.245	3.308 7.157	4.191 9.066	3.514 7.577	1.252 2.778		13
14			0.202 0.414	0.374 0.726	0.738 1.237	0.826 1.02	1.082 1.879	0.784 1.369			2.592 4.915		1.252 2.778			14
15					0.263 0.428	0.346 0.509	0.501 0.84	0.45 0.769	0.809 1.338	0.792 1.228	0.824 1.518				**2D Peak** **3D Peak**	15

2D Peak > 2.50
1.50 < 2D Peak < 2.50
1.30 < 2D Peak < 1.50
1.00 < 2D Peak < 1.30
2D Peak < 1.00

Figure C-4
C2C11 Reference Case Power Distribution at Peak Power Statepoint

C13

	R-	P-	N-	M-	L-	K-	J-	H-	G-	F-	E-	D-	C-	B-	A-	
1					0.105 0.938	0.128 1.204	0.168 1.715	0.128 1.993	0.194 2.663	0.164 3.038	0.148 4.224					1
2			0.099 0.208	0.169 0.339	0.306 0.535	0.312 0.398	0.372 0.656	0.232 0.411	0.431 0.75	0.406 0.517	0.444 0.786	0.273 0.547	0.176 0.363			2
3		0.099 0.208	0.26 0.53	0.32 0.654	0.358 0.743	0.41 0.84	0.477 1.01	0.577 1.195	0.551 1.161	0.546 1.113	0.542 1.121	0.544 1.098	0.49 0.974	0.199 0.402		3
4		0.169 0.339	0.32 0.654	0.208 0.45	0.45 0.972	0.52 1.163	0.662 1.453	0.68 1.543	0.777 1.694	0.721 1.597	0.727 1.547	0.386 0.817	0.661 1.301	0.369 0.706		4
5	0.105 0.178	0.306 0.535	0.358 0.743	0.45 0.973	0.471 1.073	0.554 1.226	0.601 1.393	0.775 1.742	0.724 1.662	0.825 1.796	0.853 1.901	0.938 1.962	0.811 1.608	0.73 1.204	0.261 0.417	5
6	0.128 0.192	0.312 0.398	0.409 0.839	0.52 1.163	0.554 1.226	0.31 0.675	0.551 1.285	0.671 1.569	0.703 1.617	0.521 1.111	1.151 2.485	1.205 2.609	1.022 2.007	0.822 0.995	0.344 0.497	6
7	0.167 0.284	0.371 0.655	0.476 1.01	0.661 1.451	0.601 1.392	0.552 1.285	0.673 1.515	0.657 1.52	0.95 2.115	1.04 2.378	1.395 3.143		1.328 2.71	1.079 1.837	0.5 0.823	7
8	0.128 0.221	0.232 0.411	0.577 1.195	0.68 1.544	0.775 1.742	0.671 1.569	0.657 1.52	0.377 0.838	0.994 2.278	1.367 3.133				0.788 1.343	0.451 0.755	8
9	0.194 0.324	0.431 0.751	0.552 1.162	0.778 1.696	0.725 1.663	0.703 1.617	0.95 2.115	0.994 2.278	1.398 3.078			2.66 5.643			0.817 1.321	9
10	0.164 0.246	0.406 0.517	0.546 1.113	0.721 1.597	0.825 1.795	0.521 1.111	1.04 2.378	1.367 3.133		1.204 2.551	2.83 6.108	3.04 6.655	2.571 5.2		0.805 1.222	10
11	0.148 0.254	0.444 0.786	0.542 1.12	0.726 1.547	0.854 1.902	1.151 2.485	1.395 3.141			2.827 6.104	3.397 7.613	4.227 9.085	3.38 7.163	2.644 4.91	0.841 1.516	11
12		0.273 0.547	0.544 1.098	0.386 0.817	0.938 1.963	1.204 2.608			2.663 5.649	3.038 6.653	4.224 9.081	4.821 10.689	4.318 9.141			12
13		0.176 0.363	0.49 0.974	0.661 1.301	0.811 1.607	1.021 2.006	1.327 2.708			2.571 5.2	3.38 7.162	4.318 9.141	3.614 7.626	1.285 2.794		13
14			0.199 0.402	0.369 0.706	0.73 1.204	0.822 0.994	1.079 1.836	0.788 1.343			2.644 4.909		1.283 2.795			14
15					0.261 0.417	0.345 0.497	0.5 0.822	0.452 0.756	0.818 1.323	0.804 1.221	0.84 1.516				**2D Peak** **3D Peak**	15

	2D Peak > 2.50
	1.50 < 2D Peak < 2.50
	1.30 < 2D Peak < 1.50
	1.00 < 2D Peak < 1.30
	2D Peak < 1.00

Figure C-5
C2C11 Conservative Case Power Distribution at Peak Power Statepoint

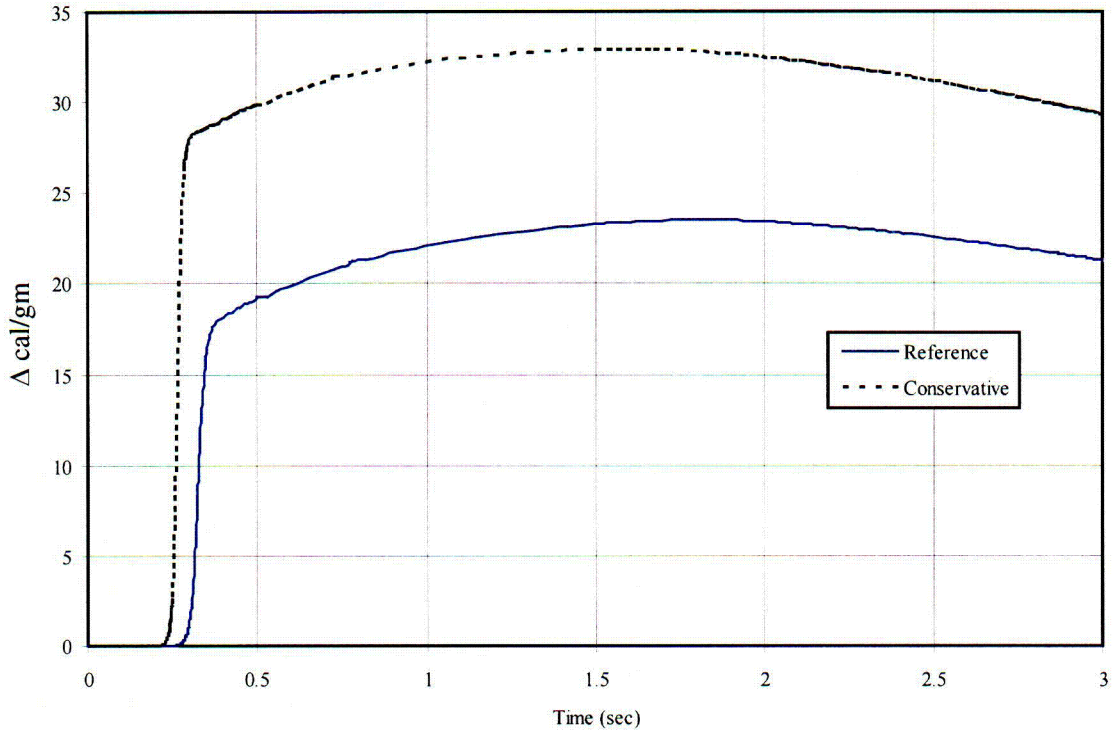

Figure C-6
C2C11 Maximum 3D Δ cal/gm

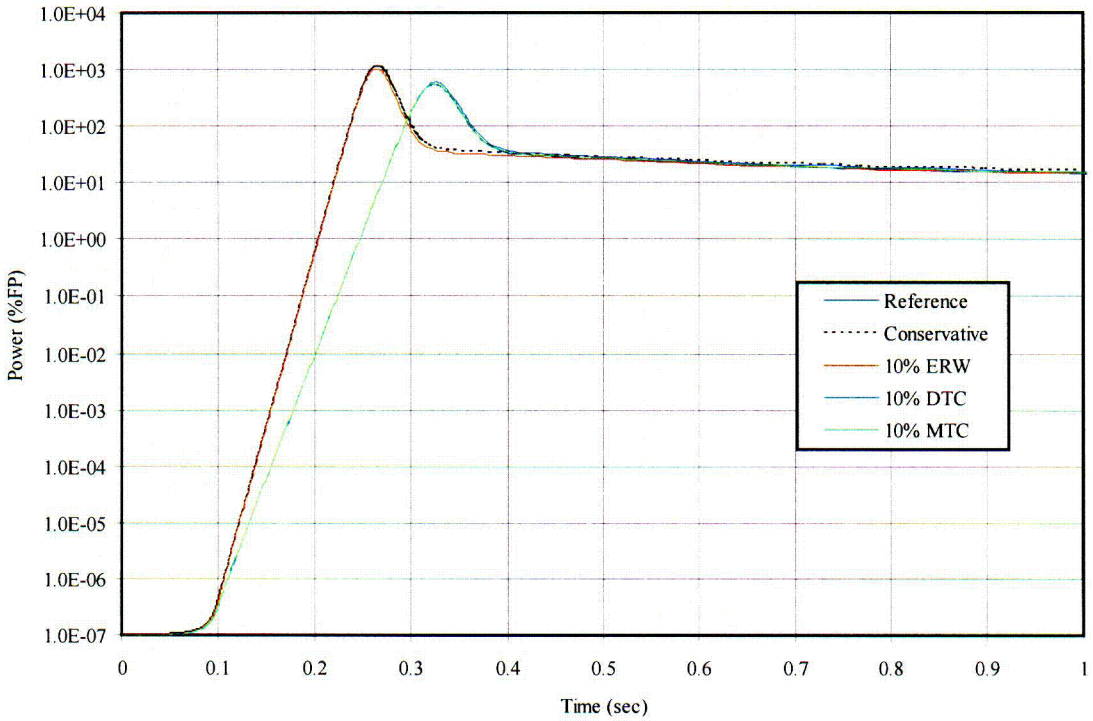

Figure C-7
C2C11 Core Average Neutron Power

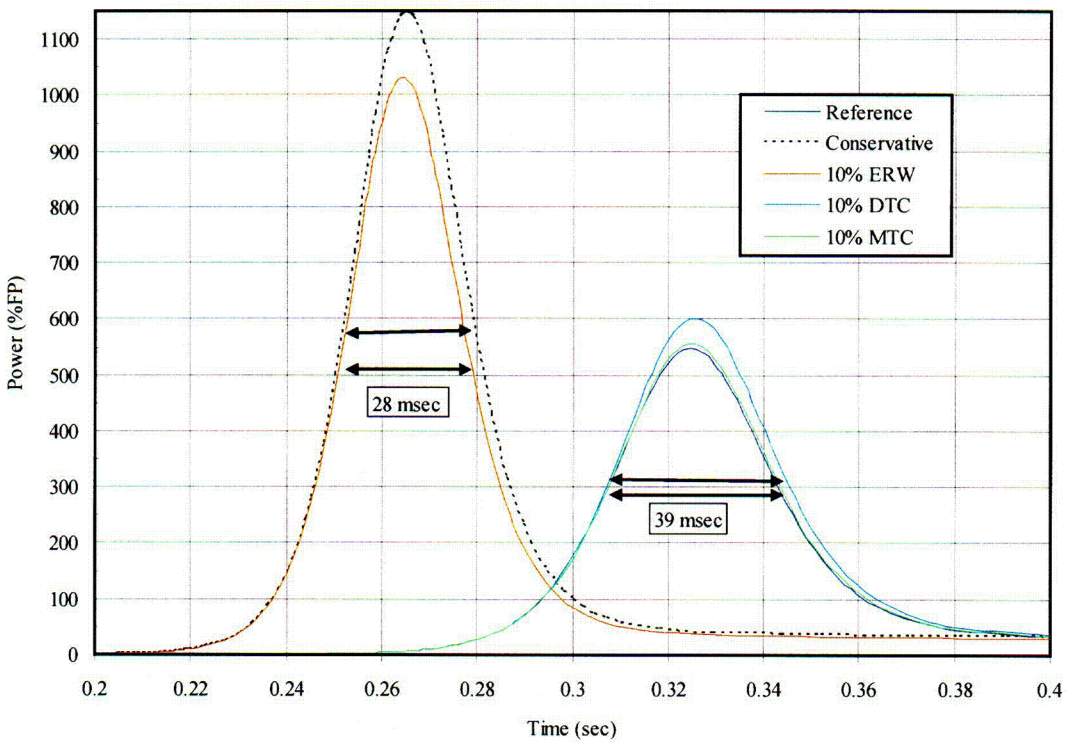

Figure C-8
C2C11 Pulse Width

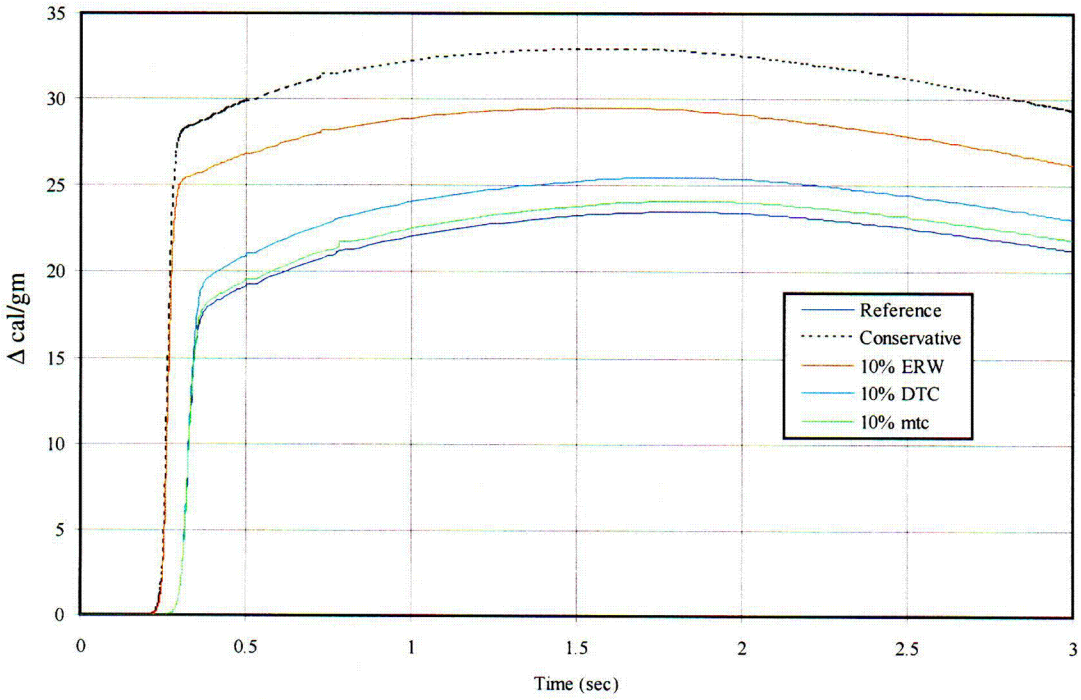

Figure C-9
Maximum 3D Δ cal/gm

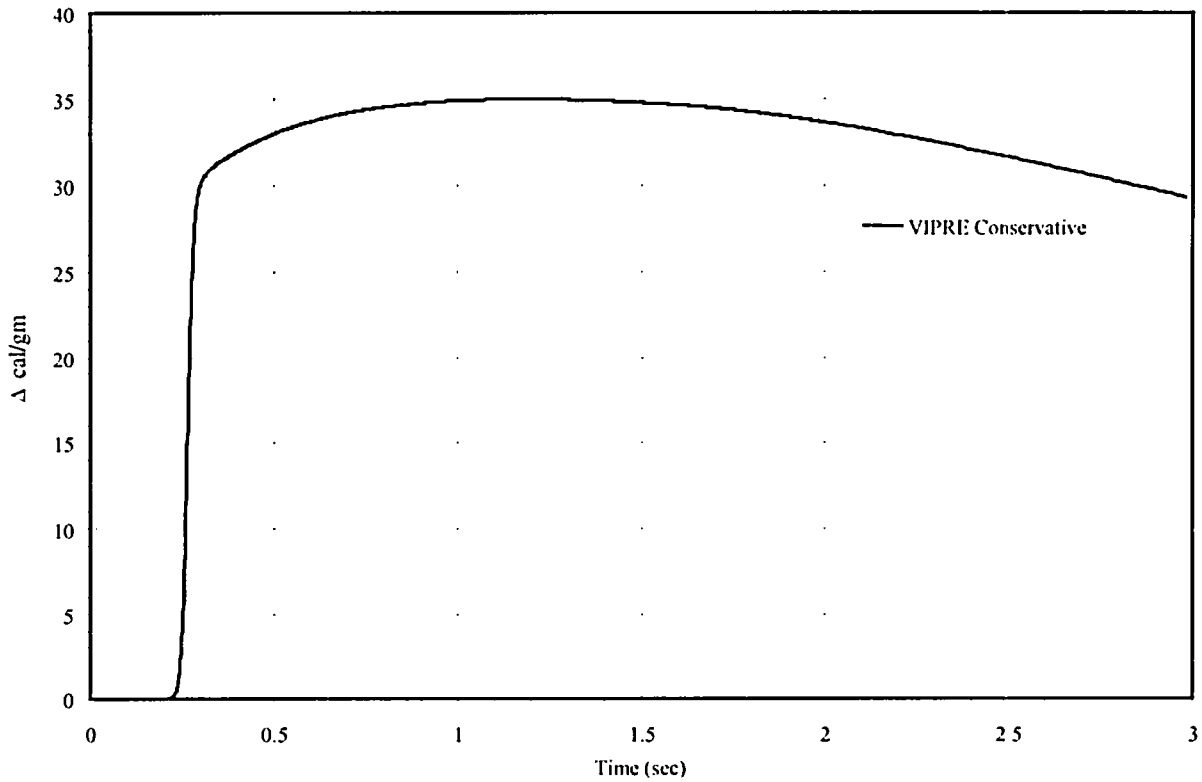

Figure C-10
VIPRE-01 Hot Rod Analysis

The results of the statistical analysis approach indicate that a reduction in the conservative REA result from 33 to 31 Δcal/gm (49 to 47 cal/gm) can be demonstrated. This is not a significant margin gain for this particular demonstration of the REA 3D methodology. The statistical approach can also be applied to the VIPRE hot rod model analysis results.

C.4 References

1. "SIMULATE-3 Kinetics Theory and Model Description," SOA-96/26, Studsvik of America, April 1996

2. "Duke Power Company Westinghouse Fuel Transition Report," DPC-NE-2009P-A, December 1999

3. "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," EPRI NP-2511-CCM, Revision 3, EPRI, August 1989

4. "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," DPC-NE-3001P-A, December 2000.

5. NUREG/CR-0479, MATPRO Version 11 (Revision 2), A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, August 1981

D
DEMONSTRATION ANALYSIS FOR TWO-LOOP B&W REACTOR

D.1 Overview

The B&W design REA 3D Δcal/gm demonstration analysis is performed for the Oconee Unit 1 Cycle 19 (O1C19) core design by Duke Power Company. Oconee Unit 1 is rated at 2568 MWth. The core consists of 177 Framatome ANP Mk-B10 15x15 fuel assemblies.. The O1C19 loading pattern is an in-in-out, low leakage design as shown in Figure D-1. It is designed to achieve a cycle burnup of 500 EFPD. The REA demonstration analysis is performed at end-of-cycle (EOC) hot zero power (HZP) conditions. Figure D-1 shows the two-dimensional (2D) assembly average exposures at 500 EFPD. Note that the nominal EOC HZP conditions for O1C19 did not yield sufficient ejected rod worth to achieve the desired prompt critical power response for the demonstration of the REA 3D analysis methodology. The nominal O1C19 EOC HZP ejected rod worth and beta-effective are 297 pcm and 0.00516, respectively. Therefore a multiplier on the ejected control rod cross sections is applied to obtain sufficient ejected rod worth to demonstrate the methodology. For O1C19, a multiplier of 1.46 was applied to the ejected control rod cross sections to yield an ejected rod worth of 774 pcm ($1.50) for the REA "reference" case.

The Studsvik-Scandpower SIMULATE-3K (S3K) code (Reference 1) is a 3D transient neutronic version of the SIMULATE-3P code. S3K uses the QPANDA full two-group nodal spatial model developed in SIMULATE-3P, with the addition of six delayed neutron groups. The program employs a fully-implicit time integration of the neutron flux, delayed neutron precursor, and heat conduction models. Beta-effective is fully functionalized similar to other cross sections to provide an accurate value of beta-effective for the time varying neutron flux. The S3K kinetics code is approved by the NRC for REA analyses in Reference 2. The REA analysis is performed using version 1.26 of the SIMULATE-3 Kinetics (S3K) code.

The S3K thermal-hydraulic model includes spatial heat conduction and hydraulic channel models. The heat conduction model solves the conduction equation on a multi-region mesh in cylindrical coordinates. Temperature-dependent values may be employed for the heat capacity, thermal conductivity, and gap conductance. A single characteristic pin conduction calculation is performed consistent with the radial neutronic node geometry, with an optional calculation of the peak pin behavior to represent a hot rod model. A single characteristic hydraulic channel calculation is performed based on the radial neutronic node geometry. The S3K enthalpy calculation is performed by solving for a radial temperature distribution and integrating across the pellet to obtain the average enthalpy. The S3K REA demonstration analysis employs a multiplier on the predicted convection heat transfer coefficient at the cladding surface to appropriately limit heat transfer during the time period of interest.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A							38.963	35.798	38.948						
B					42.315	29.537	16.734	42.225	16.768	29.547	42.300				
C			39.859	15.140	29.204	19.109	34.265	19.738	34.286	19.109	29.195	15.115	39.859		
D			15.115	34.541	20.311	36.998	21.024	47.521	21.026	37.004	20.289	34.541	15.140		
E		42.300	29.195	20.289	45.433	21.217	37.534	20.715	37.534	21.219	45.433	20.311	29.204	42.315	
F		29.547	19.109	37.004	21.219	37.089	31.215	44.531	31.201	37.089	21.217	36.998	19.109	29.537	
G	38.948	16.768	34.286	21.026	37.534	31.201	36.506	20.195	36.506	31.215	37.534	21.024	34.265	16.734	38.963
H	35.799	42.225	19.738	47.521	20.715	44.531	20.195		20.195	44.531	20.715	47.521	19.738	42.225	35.798
K	38.963	16.734	34.265	21.024	37.534	31.215	36.506	20.195	36.506	31.201	37.534	21.026	34.286	16.768	38.948
L		29.537	19.109	36.998	21.217	37.089	31.201	44.531	31.215	37.089	21.219	37.004	19.109	29.547	
M		42.315	29.204	20.311	45.433	21.219	37.534	20.715	37.534	21.217	45.433	20.289	29.195	42.300	
N			15.140	34.541	20.289	37.004	21.026	47.521	21.024	36.998	20.311	34.541	15.115		
O			39.859	15.115	29.195	19.109	34.286	19.738	34.265	19.109	29.204	15.140	39.859		
P					42.300	29.547	16.768	42.225	16.734	29.537	42.315				
R							38.948	35.798	38.963						

First Cycle

Second Cycle

Third Cycle

Fourth Cycle

Figure D-1
O1C19 EOC Assembly Average Burnup Distribution

In addition to calculating the core thermal response and the peak fuel pellet enthalpy within the S3K code, the EPRI VIPRE-01 thermal-hydraulic subchannel analysis code (Reference 3) is also used with transient core power and peaking factor boundary conditions from the S3K analyses. This hot rod analysis uses the Duke Power NRC-approved VIPRE-01 model and methodology for REA fuel rod thermal analyses as described in Reference 2.

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D.2 Description of Analysis

Initial conditions for the REA demonstration analyses are described in Table D-1. These conditions along with the previously described ejected rod worth multiplier and convection heat transfer coefficient multiplier define the "reference" case. The initial core radial power distribution for the reference case is provided in Figure D-2.

The control rod is ejected from core location L-10 in 0.15 seconds. The reactor is assumed to trip on high flux at the time of peak neutron power, although the actual trip would occur a few milliseconds earlier. A 0.4 second time delay between trip actuation and the start of control rod insertion is assumed. The duration of the analysis is 4 seconds, which is sufficient to obtain all of the analysis results of interest.

Three "sensitivity" analyses to the reference case are performed for the key physics parameters to obtain the results necessary for the statistical analysis approach. Each of these sensitivity analyses represents a quantification of the effect on the Δcal/gm result due to the uncertainty in the physics parameter. S3K has the capability to apply multiplicative factors to physics parameter cross sections to drive ejected rod worth (ERW), moderator temperature coefficient (MTC), and fuel Doppler temperature coefficient (DTC) to desired values. Each of these physics parameters are changed by 10% in the conservative direction for the three sensitivity cases. The REA sensitivity cases and conditions are listed in Table D-2. In addition, the results of a "conservative" case that combines all three sensitivity factors are given. The Δcal/gm results are converted to peak enthalpy values by adding the initial enthalpy value of 15 cal/gm.

Table D-1
O1C19 Core Initial Conditions

Parameter	Condition	Value
Burnup	EOC	500 EFPD
Power	HZP critical	1.0 E-7 %
Temperature	HZP T-inlet	532°F
Core flow	77% design flow (3 RCPs)	101.78 E6 lbm/hr
Control rods	HZP rod insertion limit	Banks 5-7 fully inserted
Boron	Critical	236 ppm
Xenon	No xenon	

Table D-2
O1C19 Demonstration Analysis Cases

Description	S3K Cross Section Multiplier
Reference case	RODOUT=1.4582
+10% ejected rod worth (ERW) uncertainty sensitivity	RODOUT=1.4997
-10% MTC uncertainty sensitivity	RODOUT=1.4582, TMO=0.9
-10% DTC uncertainty sensitivity	RODOUT=1.4582, TFU=0.9
Conservative case (+10%ERW, -10% MTC & DTC)	RODOUT=1.4997, TMO & TFU =0.9

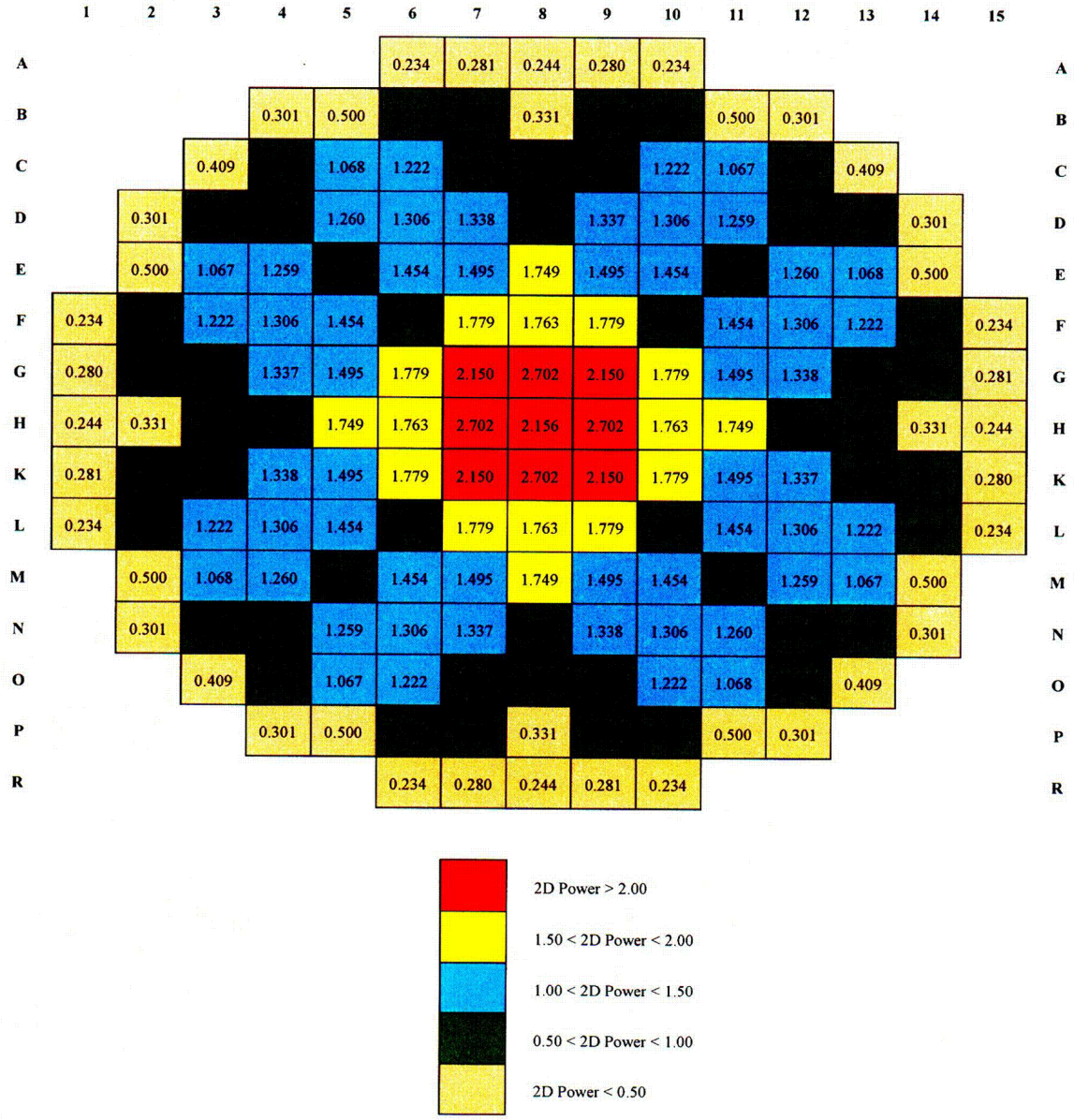

Figure D-2
O1C19 Initial conditions Assembly Radial Power Distribution

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The hot rod analysis using the VIPRE-01 code is a conservative model and methodology. The hot rod is driven with transient radial peaking factors corresponding to the hot pin from the S3K REA analysis. The axial power shape is assumed to remain at the initial shape for simplicity, which has been shown to be a conservative modeling approach. The gas gap conductivity remains at the initial value to maximize the pellet enthalpy, and is a conservative assumption. The VIPRE code then calculates a peak average fuel temperature as a function of time for the hot spot on the hot rod during the REA. This transient temperature result is then converted to enthalpy units using the equation from the MATPRO manual (Reference 4).

The statistical analysis approach applies the SRSS combination of uncertainties methodology to the Δcal/gm results of the three sensitivity analyses plus a fourth value which is a 10% increase on the results of the reference case to account for the uncertainty in the prediction of the power peaking. The result of the statistical calculation is an alternative to the result of the conservative case in which the uncertainties in the key physics parameters are applied simultaneously.

D.3 Results

The results of the REA demonstration analyses are provided in the following tables and figures. Table D-3 provides a summary of the REA demonstration case results for the following parameters:

- Case – description of each analysis (see Table D-2)
- Peak Power Time (sec) – time of peak neutron power
- Trip Signal Time (sec) – time that the trip setpoint is reached
- Peak Power (%FP) – peak neutron power
- RHO ($) – ejected rod worth in dollars
- RHO (ΔK/K) – ejected rod worth in ΔK/K units
- Core Average Moderator Temperature (°F) – maximum core average moderator temperature
- Core Average Fuel Temperature (°F) – maximum core average fuel temperature
- F_DEL-H – maximum radial pin power
- F-Q – maximum total power
- Max Δcal/gm (Pin) – maximum Δcal/g on a pin basis
- Max Δcal/gm (Node) – maximum Δcal/g on a nodal basis
- Peak cal/gm – peak pellet total cal/gm
- Pulse Width (msec) – full-width/half height based on core average power

Figure D-3 shows the neutron power transient response for the reference and conservative cases. The peak power for the reference case is 1168% and occurs at 0.33 seconds. The peak power for the conservative case is 2113% and occurs at 0.29 seconds. Figure D-4 shows the radial fuel assembly (2D) and total nodal (3D) power distributions at the peak power statepoint for the reference case. Figure D-5 shows the radial fuel assembly and total nodal power distributions for

the peak power statepoint for the conservative case. Figure D-6 plots the maximum 3D Δcal/gm transient response on a pin basis for the reference and conservative cases. The maximum 3D pellet average enthalpy increase is 31 Δcal/gm (46 cal/gm) for the reference case and 42 Δcal/gm (57 cal/gm) for the conservative case.

Table D-3
O1C19 Demonstration Analysis Results Case Summary

Case	Peak Power Time (sec)	Trip Signal Time (sec)	Peak Power (% FP)	RHO ($)	RHO (ΔK/K)	Core Avg. Moderator Temperature (F)
Reference	0.332	0.332	1168	1.39	7.20E-03	535
10% ERW	0.290	0.290	1905	1.50	7.80E-03	536
10% DTC	0.332	0.332	1282	1.39	7.20E-03	536
10% MTC	0.332	0.332	1184	1.39	7.20E-03	535
Conservative	0.291	0.291	2113	1.50	7.80E-03	536

Case	Core Avg. Fuel Temperature (F)	F_DEL-H	F-Q	Max Δcal/gm (Pin)	Max Δcal/gm (Node)	Peak cal/gm	Pulse Width (msec)
Reference	617	5.18	12.12	31	28	46	33
10% ERW	632	5.49	13.01	38	35	53	25
10% DTC	615	5.15	12.12	34	31	49	34
10% MTC	617	5.17	12.12	31	29	46	34
Conservative	632	5.46	13.01	42	39	57 / 65	25

Figure D-7 shows the core average neutron power response for the reference case, the conservative case, and the three sensitivity cases. The peak power results for the sensitivity cases are 1905% for the +10% ERW case, 1282% for the -10%DTC case, and 1184% for the -10%MTC case (Table D-3). Figure D-8 shows the same results as Figure D-7 but the axes have been reduced so that the REA pulse shape and width can be illustrated. The pulse widths are given in Table D-3 and range from 25 to 31 milliseconds. Figure D-9 shows the 3D maximum delta-enthalpy results for all cases. These results show the ejected rod worth to be the most significant sensitive parameter in the 3D REA analysis.

The VIPRE-01 hot rod analysis result for the conservative case is shown in Figure D-10. The maximum value is 50 Δcal/gm (65 cal/gm). This exceeds the S3K result of 42 Δcal/gm (57 cal/gm) for the conservative case. The trend of the temperature prediction compares well with the S3K result. It can be concluded that the fuel rod and pellet thermal models in S3K may not be sufficiently conservative for calculating the Δcal/gm result for the REA. If the Δcal/gm result predicted by S3K approaches the regulatory acceptance limit, then the more detailed VIPRE-01 hot rod model must be used to confirm that the analysis results meet the acceptance limit.

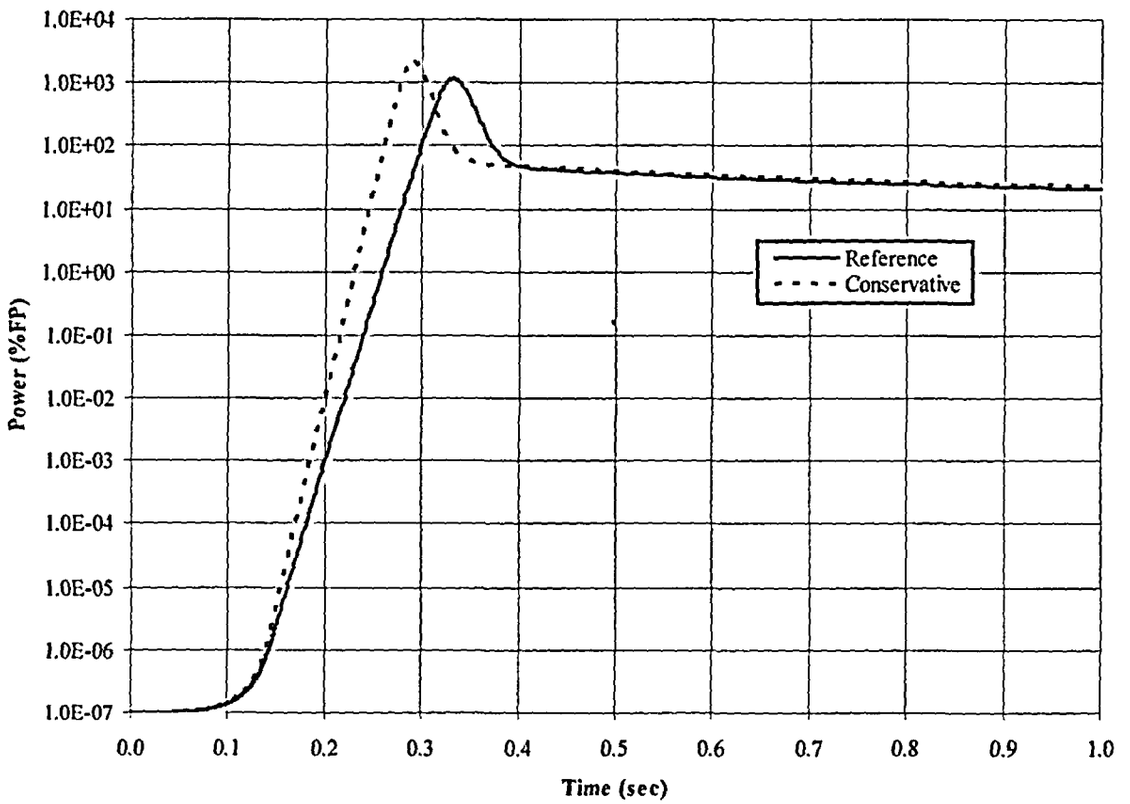

Figure D-3
O1C19 Core Average Neutron Power

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A						0.121 0.278	0.149 0.346	0.136 0.314	0.163 0.378	0.140 0.319						A
B				0.151 0.344	0.256 0.585	0.374 0.865	0.409 0.932	0.189 0.436	0.453 1.030	0.438 1.007	0.314 0.713	0.193 0.439				B
C			0.203 0.472	0.486 1.114	0.556 1.279	0.658 1.498	0.345 0.798	0.479 1.096	0.383 0.883	0.780 1.769	0.691 1.583	0.636 1.451	0.280 0.647			C
D		0.151 0.344	0.485 1.113	0.345 0.795	0.680 1.549	0.746 1.687	0.817 1.883	0.446 1.036	0.906 2.081	0.889 2.001	0.865 1.961	0.475 1.087	0.703 1.602	0.224 0.508		D
E		0.256 0.585	0.555 1.278	0.679 1.548	0.405 0.937	0.900 2.074	0.999 2.360				0.550 1.265		0.846 1.930	0.399 0.904		E
F	0.121 0.278	0.374 0.864	0.658 1.498	0.746 1.686	0.900 2.074	0.656 1.544			1.501 3.538	0.855 2.000				0.617 1.413	0.206 0.468	F
G	0.148 0.345	0.409 0.932	0.345 0.798	0.817 1.883	0.999 2.360		1.762 4.138	2.405 5.542	2.102 4.926	1.889 4.440	1.621 3.796		0.606 1.386	0.728 1.643	0.270 0.620	G
H	0.136 0.314	0.189 0.435	0.479 1.096	0.446 1.036			2.405 5.542	2.175 4.972	3.117 7.169	2.306 5.424	2.378 5.497	0.898 2.059	0.978 2.211	0.387 0.880	0.277 0.634	H
K	0.163 0.379	0.453 1.030	0.383 0.883	0.906 2.082		1.501 3.539	2.102 4.927	3.118 7.169	3.152 7.378	3.373 7.891	2.717 6.321	2.171 4.914	0.893 2.027		0.367 0.840	K
L	0.140 0.319	0.438 1.008	0.780 1.769	0.890 2.001		0.855 2.001	1.889 4.440	2.307 5.425	3.374 7.894	4.137 9.628	3.349 7.602	2.299 5.105	1.904 4.251		0.327 0.735	L
M		0.314 0.713	0.692 1.585	0.865 1.963	0.550 1.266		1.621 3.796	2.378 5.499	2.718 6.323	3.349 7.603		2.163 4.835	1.677 3.786	0.751 1.684		M
N		0.193 0.439	0.636 1.452	0.474 1.087				0.898 2.059	2.172 4.915	2.300 5.106	2.164 4.837			0.455 1.019		N
O			0.280 0.647	0.703 1.600	0.845 1.929		0.606 1.386	0.978 2.211	0.893 2.027	1.904 4.253	1.678 3.789		0.622 1.416			O
P				0.224 0.508	0.399 0.904	0.617 1.413	0.728 1.645	0.387 0.879			0.751 1.684	0.456 1.020				P
R						0.206 0.467	0.269 0.619	0.277 0.635	0.368 0.843	0.327 0.736						R

2D Peak
3D Peak

2D Peak > 3.00

2.00 < 2D Peak < 3.00

1.50 < 2D Peak < 2.00

1.00 < 2D Peak < 1.50

2D Peak < 1.00

Figure D-4
Reference Case Power Distribution at Peak Power Statepoint

C19

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A						0.117 0.267	0.144 0.333	0.131 0.303	0.158 0.365	0.136 0.308						A
B				0.146 0.330	0.247 0.561	0.362 0.830	0.395 0.896	0.184 0.420	0.440 0.995	0.426 0.974	0.306 0.690	0.189 0.425				B
C			0.196 0.452	0.468 1.068	0.537 1.227	0.637 1.440	0.335 0.768	0.466 1.058	0.374 0.854	0.761 1.712	0.675 1.534	0.622 1.408	0.275 0.629			C
D		0.146 0.330	0.468 1.067	0.333 0.763	0.657 1.488	0.724 1.624	0.795 1.817	0.435 1.002	0.886 2.017	0.870 1.940	0.847 1.905	0.466 1.058	0.691 1.561	0.221 0.496		D
E		0.247 0.561	0.536 1.226	0.657 1.487	0.393 0.903	0.876 2.001	0.976 2.283				0.542 1.235	0.987 2.212	0.834 1.886	0.394 0.884		E
F	0.117 0.267	0.362 0.830	0.636 1.439	0.724 1.623	0.876 2.001	0.642 1.495				0.847 1.960				0.611 1.387	0.204 0.459	F
G	0.144 0.332	0.396 0.896	0.335 0.768	0.795 1.817	0.976 2.283		1.737 4.029	2.380 5.411	2.091 4.836	1.887 4.379	1.622 3.751		0.604 1.367	0.725 1.620	0.268 0.612	G
H	0.131 0.303	0.184 0.420	0.466 1.058	0.436 1.002			2.380 5.411	2.167 4.886	3.127 7.086	2.329 5.401	2.404 5.479	0.906 2.052	0.984 2.197	0.388 0.873	0.277 0.629	H
K	0.159 0.366	0.440 0.994	0.374 0.854	0.886 2.017		1.483 3.455	2.091 4.837	3.128 7.086	3.205 7.388	3.471 7.995	2.784 6.381	2.208 4.926	0.903 2.024		0.370 0.836	K
L	0.136 0.308	0.426 0.974	0.761 1.713	0.870 1.941		0.847 1.960	1.887 4.379	2.330 5.403	3.472 7.997	4.374 10.018	3.463 7.738	2.344 5.132	1.927 4.249		0.329 0.733	L
M		0.306 0.691	0.675 1.535	0.847 1.906	0.542 1.235		1.622 3.751	2.405 5.480	2.785 6.384	3.463 7.739	1.507 3.394	2.200 4.851	1.698 3.782	0.759 1.680		M
N		0.189 0.425	0.622 1.409	0.465 1.058	0.987 2.211			0.906 2.052	2.209 4.927	2.345 5.134	2.201 4.854			0.460 1.017		N
O			0.275 0.629	0.691 1.559	0.833 1.885		0.604 1.367	0.984 2.197	0.904 2.024	1.928 4.251	1.699 3.785	1.501 3.332	0.628 1.413			O
P				0.221 0.495	0.394 0.884	0.611 1.387	0.725 1.621	0.388 0.872			0.759 1.681	0.460 1.018				P
R						0.204 0.459	0.268 0.610	0.278 0.630	0.371 0.839	0.329 0.734						R

2D Peak
3D Peak

	2D Peak > 3.00
	2.00 < 2D Peak < 3.00
	1.50 < 2D Peak < 2.00
	1.00 < 2D Peak < 1.50
	2D Peak < 1.00

Figure D-5
Conservative Case Power Distribution at Peak Power Statepoint

C20

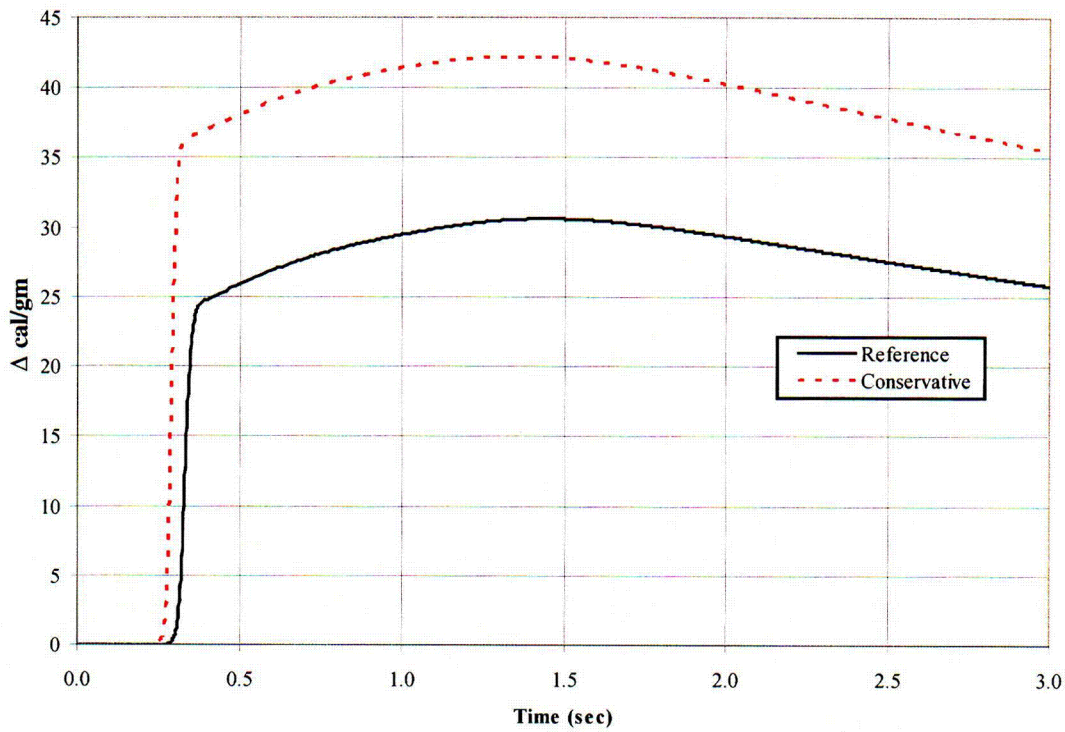

Figure D-6
O1C19 Maximum 3D Δ cal/gm

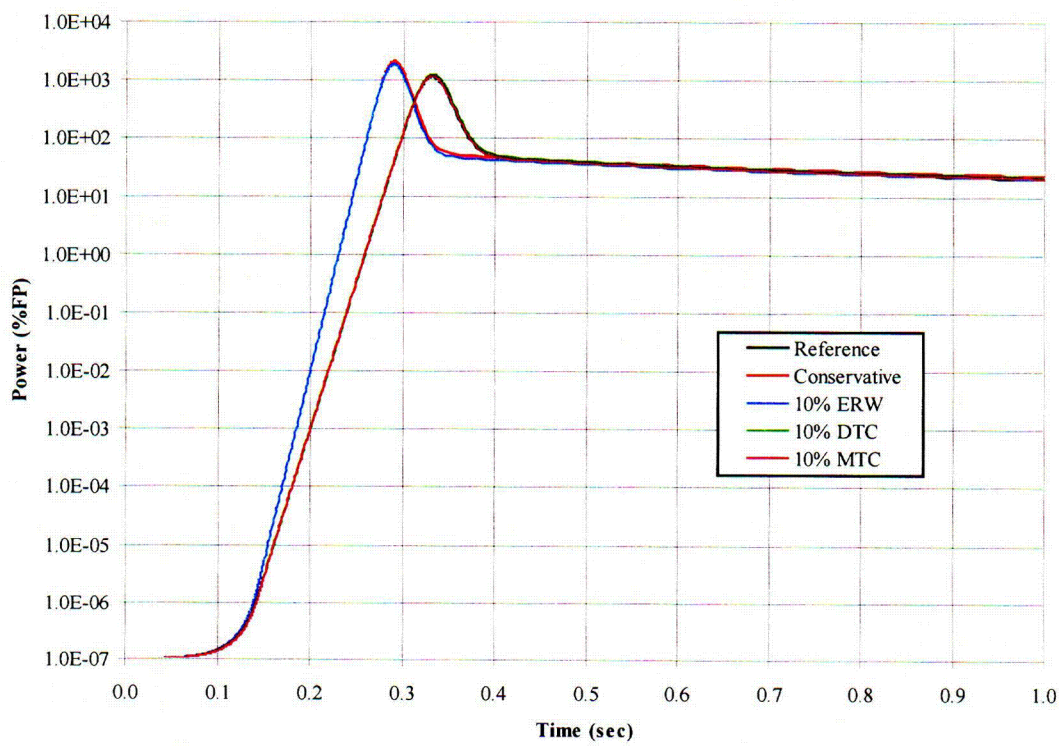

Figure D-7
O1C19 Core Average Neutron Power

C21

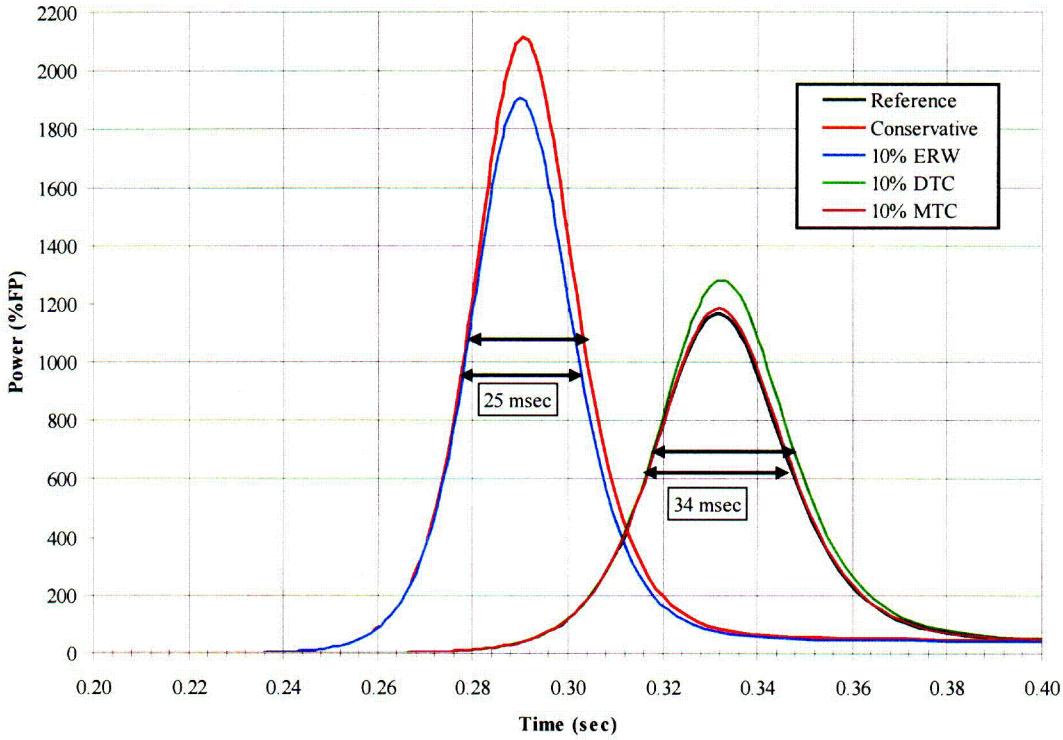

Figure D-8
O1C19 Pulse Width

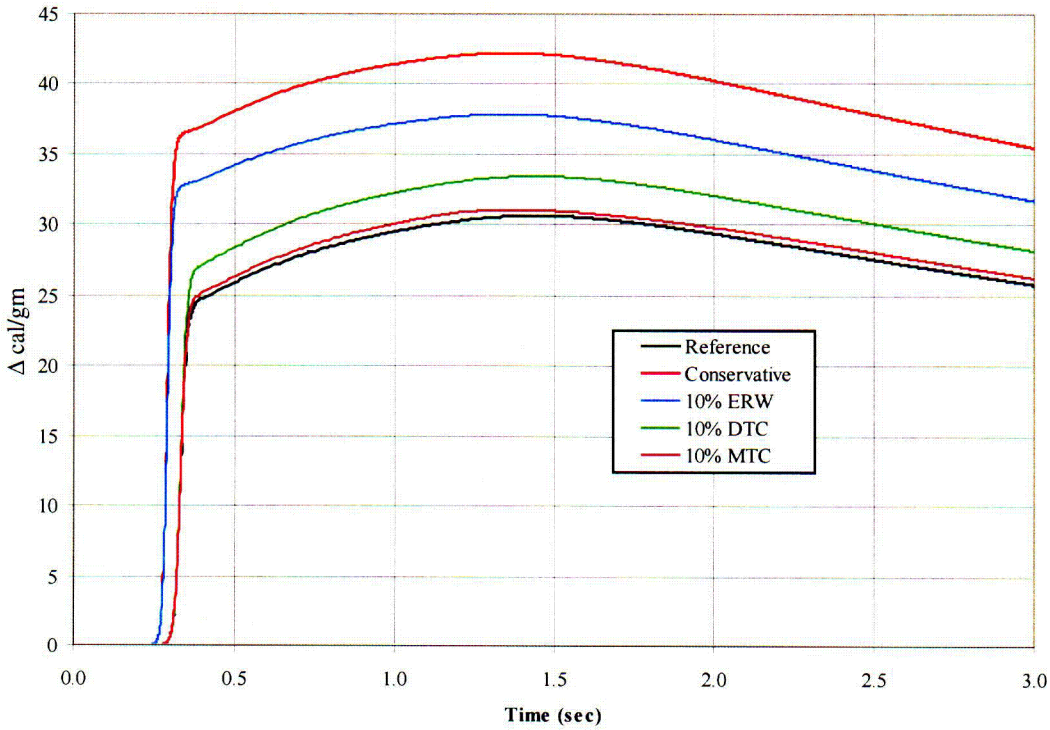

Figure D-9
O1C19 Maximum 3D Δ cal/gm

C22

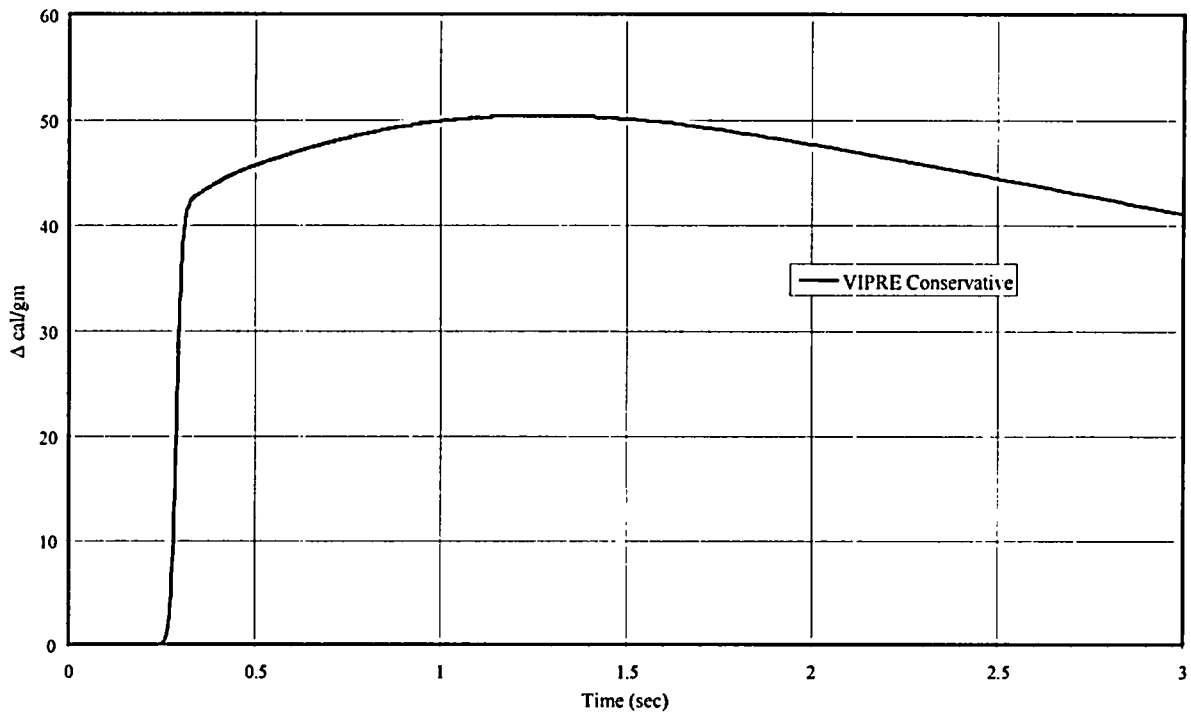

Figure D-10
VIPRE-01 Hot Rod Analysis

The statistical approach combines the change (i.e. Δ/Δcal/gm), relative to the reference case result of 31 Δcal/gm, in the Δcal/gm results for the three sensitivity analyses, along with the +10% peaking uncertainty value based on the reference case result, using the SRSS combination of uncertainties methodology. This statistical result is then compared to the results of the conservative case using the S3K results (42 Δcal/gm) to determine the potential margin gain with the statistical approach.

Result of +10% ERW sensitivity case = 38 Δcal/gm (Δ/Δcal/gm = 7)

Result of –10% DTC sensitivity case = 34 Δcal/gm (Δ/Δcal/gm = 3)

Result of –10% MTC sensitivity case = 31 Δcal/gm (Δ/Δcal/gm = 0)

Result of +10% increase in reference case peaking = (Δ/Δcal/gm = 31 x 1.1 = 3.1)

SRSS (Δ/Δcal/gm) = SQRT $(7^2 + 3^2 + 0^2 + 3.1^2)$ = 8 Δ/Δcal/gm

Statistical result = reference case result + SRSS uncertainty = 31 + 8 = 39 Δcal/gm = 54 cal/gm

The results of the statistical analysis approach indicate that a reduction in the conservative REA result from 42 to 39 Δcal/gm (57 to 54 cal/gm) can be demonstrated using the S3K analysis results. This is not a significant margin gain for this particular demonstration of the REA 3D methodology. The statistical approach can also be applied to the VIPRE hot rod model analysis results.

D.4 References

1. "SIMULATE-3 Kinetics Theory and Model Description," SOA-96/26, Studsvik of America, April 1996

2. "UFSAR Chapter 15 Transient Analysis Methodology", DPC-NE-3005-PA, Revision 1, Duke Power Company, August 1999.

3. "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," EPRI NP-2511-CCM, Revision 3, EPRI, August 1989

4. NUREG/CR-0479, MATPRO Version 11 (Revision 2), A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, August 1981

E
PROPOSED REVISIONS TO REA REGULATORY DOCUMENTS

E.1 Introduction

The current licensing analyses of postulated REA events are governed primarily by two regulatory documents. These are: a) Regulatory Guide 1.77, "Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors", issued May 1974, and b) Standard Review Plan (SRP) 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)", Revision 2, July 1981. Additional revisions have been proposed to the SRP, but these have not yet been adopted. These are: a) SRP 15.4.8, draft Rev. 3, April 1996, and b) SRP 15.4.8 APPENDIX A, draft Rev. 2, April 1996. The proposed revisions to the SRP consist of editorial revisions, format modifications, and changes to names of NRC branches due to reorganization.

These two documents, Regulatory Guide 1.77, and the SRP, identify acceptance criteria for meeting the requirements of General Design Criteria 28. The intent is that effects of postulated reactivity accidents should not result in damage to the reactor coolant pressure boundary greater than limited local yielding, or sufficient damage to impair significantly the capacity to cool the core.

The current licensing analyses of REA events generally conform to the requirements specified in these documents, and essentially follow a deterministic approach. The methodology proposed in this report for REA analysis departs from the conventional licensing analyses in several areas. First, it is recognized that the acceptance criteria for fuel energy deposition in a REA event are in the process of being modified by the NRC in light of the recent experimental results for high burnup fuel. The nuclear industry, through the Nuclear Energy Institute, has proposed new PWR REA cladding failure and core coolability acceptance limits for REA for high burnup fuel. The new proposed limits were developed by the Electric Power Research Institute (EPRI) on behalf of the industry, and are detailed in "Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel and Core Coolability Criteria [16]". This report was submitted to the NRC in June 2002.

Second, the methodology utilizes 3D kinetics models for calculating the core power distribution, and hence the local transient power in the hot fuel rod. Third, probability-based assumptions are used to define the scope of core initial conditions to be analyzed. Finally, a statistical approach is used to address uncertainties associated with key analysis parameters.

This report describes a new methodology for REA 3D fuel enthalpy analysis that is expected to be used by various industry organizations as a part of licensing submittals. The two regulatory documents are reviewed here to assess how they would be impacted if the approach proposed in

the methodology is considered acceptable by the NRC for licensing-basis REA analysis. The results of the review are provided as markups to the regulatory documents, with notes explaining the reason for each markup. The intent of providing this information is to highlight how the features of the new methodology relate to the regulatory documents.

It is noted that Appendix A of Standard Review Plan 15.4.8 provides additional information for radiological analysis. Since the perspective of this report is on the fuel enthalpy calculation, the impact on Appendix A is not addressed here.

E.2 Regulatory Guide 1.77

Regulatory Guide 1.77 identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium oxide-fueled pressurized water reactors (PWRs). It is essentially based on the state-of-the-art in reactor physics and thermal-hydraulic analysis as of 1974.

The regulatory guide discussed that high fuel energy densities could result in prompt rupture of fuel pins and rapid heat transfer to the water from finely dispersed molten UO_2. The conversion to mechanical energy could conceivably disarrange the reactor core or breach the primary system. The regulatory guide states that "a calculated radial average energy density of 280 cal/g at any axial fuel location in any fuel rod provides a conservative maximum limit to ensure that core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired."

The regulatory guide states that initial reactor states should include at least:

1. Zero power (hot standby) – beginning-of-life (BOL) and end-of-life (EOL)

2. Low power – BOL and EOL

3. Full power – BOL and EOL

The regulatory position in Section C of the Regulatory Guide requires that it be shown that:

1. Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location in any fuel rod.

2. Maximum reactor pressure during any portion of the assumed transient will be less than the Emergency Condition stress limits in the ASME Boiler and Pressure Vessel Code.

3. Offsite dose consequences will be within the guidelines of 10 CFR Part 100.

Regulatory Guide 1.77, Appendix A, addresses physics and thermal-hydraulics, and describes assumptions which should be applied in the following areas:

1. Ejected rod worth.

2. Reactivity insertion rate.

3. Calculation of effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (l^*).
4. Initial reactor coolant pressure, core inlet temperature, and flow rate.
5. Fuel thermal properties, such as fuel-clad gap heat transfer coefficient and fuel thermal conductivity.
6. Specific heat of UO_2.
7. Moderator reactivity coefficient.
8. Doppler coefficient.
9. Control rod reactivity insertion during trip versus time.
10. Reactor trip delay time.
11. Computer code capabilities
12. Documentation and conservatism of codes and models
13. Pressure surge calculation.
14. Number of fuel rods experiencing clad failure, assuming clad failure if the DNB heat flux is exceeded.

E.3 Standard Review Plan 15.4.8

The SRP essentially provides guidance for review of FSAR REA analyses and REA methodology topical reports submitted for NRC approval. The SRP adds to the criteria specified in Regulatory Guide 1.77 by requiring that the number of rods used in the radiological evaluation is the number of rods calculated to have departure from nucleate boiling (DNB). The review procedures in Appendix A to the SRP further require consideration of the amount of fuel reaching the fuel melting temperature, as opposed to the 280 cal/g criteria.

The SRP provides procedures for conducting the review of the methodology and analysis, calling out specific features of the analysis for review.

E.4 Proposed Revisions to Regulatory Documents

The regulatory documentation, particularly Regulatory Guide 1.77, contains implicit assumptions in many areas that the methods employed will analyze the problem in less than three dimensions, and will therefore require conservative modeling and assumptions to compensate. The regulatory guide also assumes that deterministic methods employing conservative values of input parameters will be used.

The regulatory documentation, mostly the SRP, emphasizes DNB and fuel melt as failure mechanisms. Based on current knowledge of REA phenomena, neither of these is likely to be a factor in prompt-critical events from zero power. The likely failure mode for prompt-critical REA events from zero power is pellet-clad mechanical interaction, which is not addressed in the current regulatory documents.

The criteria for fuel failure would need to be revised if a lower cal/gm limit is adopted. A lower cal/gm limit would likely make DNBR and fuel melting less limiting consequences of a zero power prompt-critical control rod ejection. DNBR and fuel melting would still be relevant to analysis of control rod ejection events from full power.

Appendix A to Regulatory Guide 1.77 makes reference to conservative values of inputs or changes to account for calculation uncertainties in several subsections. Specifically, Section A (1) calls for increases in calculated rod worths, Section A(3) calls for conservatism in β_{eff} and prompt neutron lifetime, Section A(4) calls for conservatively chosen inlet conditions, Section A(5) calls for conservatively chosen fuel thermal properties, Section A(8) calls for Doppler coefficients which compare conservatively, and Section A(12) calls for conservatism of flux shapes. This guidance is not consistent with an approach using a statistical combination of uncertainties. The 280 cal/gm acceptance criteria will have to be changed to the new value. Also, 3D nodal models use neutron velocities instead of prompt neutron lifetime.

The review procedures in SRP 15.4.8 include reference to the 280 cal/g criteria and to DNBR as the fuel failure criteria. The 280 cal/g limit will need to be changed, and the DNBR criteria is probably not relevant to zero power prompt-critical events.

The following sections provide proposed revisions to the regulatory documents that reflect the above comments. The proposed revisions are indicated by number on marked-up copies of the regulatory documents. Table E-1 summarizes the reasons for each markup, referencing each item in Regulatory Guide 1.77. Table E-2 provides similar information for SRP Section 15.4.8.

Table E-1
Proposed Revisions to Regulatory Guide 1.77

Item	Revision	Comment
1	Delete paragraph and insert new acceptance criteria	New acceptance criteria expected to be implemented
2	Insert paragraph: "As an alternative, a probabilistic approach may be used to justify the set of initial reactor states considered for analysis."	New methodology uses probabilistic analysis, and does not always choose maximum or minimum values for analytical parameters with respect to their expected (most probable) values. Methodology has shown, using probability-based arguments, that it is not necessary to consider all the plant initial states considered in current licensing analysis.
3	Delete paragraph and replace with new acceptance criteria.	New acceptance criteria expected to be implemented
4	Insert sentence: "As an alternative, a probabilistic approach may be used to justify the ejected rod worth considered for analysis."	Methodology uses probability-based assumptions and analysis and assesses uncertainties in rod worth.
5	Insert paragraph: "If three-dimensional kinetics methods are used, the reactivity insertion rate due to an ejected rod is simulated by ejected rod position vs. time input to the analysis."	Three-dimensional kinetics methodology explicitly accounts for rod worth versus rod position. The methodology does not use differential rod worths.
6	Insert paragraph: " If three-dimensional kinetics methods are used, the inverse neutron velocity is equivalent to the prompt neutron lifetime used in point-kinetics methods."	Prompt neutron lifetime, a concept associated with simplified kinetics models, is replaced by inverse neutron velocity in three-dimensional kinetics methods.

Table E-2
Proposed Revisions to SRP 15.4.8

Item	Revision	Comment
1	Delete paragraph. Insert new acceptance criteria	New acceptance criteria expected to be implemented
2	Insert paragraph: "If a three-dimensional space-time calculation is performed, the reactivity feedback resulting from the cross section files must be consistent with the core initial conditions assumed in the analysis. The methodology must include a characterization of the accuracy of the reactivity feedback."	Consistent with the proposed new methodology.
3	Insert sentence: "If three-dimensional kinetics methods are used, differential rod worths are not relevant."	Three-dimensional kinetics methodology explicitly accounts for rod worth versus rod position. The methodology does not use differential rod worths.
4	Insert new acceptance criteria	New acceptance criteria expected to be implemented
5	Insert new acceptance criteria	New acceptance criteria expected to be implemented

May 1974

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.77

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

A. INTRODUCTION

Section 50.34, "Contents of applications: technical information," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each application for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the potential risk to public health and safety resulting from operation of the facility. General Design Criterion 28, "Reactivity Limits," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, requires the reactivity control system to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. General Design Criterion 28 also requires that these postulated reactivity accidents include consideration of the rod ejection accident unless such an accident is prevented by positive means.

This guide identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium oxide-fueled pressurized water reactors (PWRs). In some cases, unusual site characteristics, plant design features, or other factors may require different assumptions which will be considered on an individual basis. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

The rate at which reactivity can be inserted into the core of a uranium oxide-fueled water-cooled power reactor is normally limited by the design of the control rod system to a value well below that which would result in serious damage to the reactor system. However, a postulated failure of the control rod system provides the potential for a relatively high rate of reactivity insertion which, if large enough, could cause a prompt power burst. For UO_2 fuel, a large fraction of this generated nuclear energy is stored momentarily in the fuel and then released to the rest of the system. If the fuel energy densities were high enough, there would exist the potential for prompt rupture of fuel pins and the consequent rapid heat transfer to the water from finely dispersed molten UO_2. Prompt fuel element rupture is defined herein as a rapid increase in internal fuel rod pressure due to extensive fuel melting, followed by rapid fragmentation and dispersal of fuel cladding into the coolant. This is accompanied by the conversion of nuclear energy, deposited as overpower heat in the fuel and in the coolant, to mechanical energy which, in sufficient quantity, could conceivably disarrange the reactor core or breach the primary system.

The Regulatory staff has reviewed the available experimental information concerning fuel failure thresholds. In general, failure consequences for UO_2 have been insignificant below 300 cal/g for both irradiated and unirradiated fuel rods. Therefore, a calculated radial average energy density of 280 cal/g at any axial fuel location in any fuel rod as a result of a postulated rod ejection accident provides a conservative maximum limit to ensure that core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired. 1

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position.

A sufficient number of initial reactor states to completely bracket all possible operational conditions of interest should be analyzed to assure examination of upper bounds on ultimate damage. In areas of uncertainty, the appropriate minimum or maximum parameters relative to nominal or expected values should be used to assure a conservative evaluation. The initial reactor states should include consideration of at least the following:

Zero power (hot standby)	– Beginning of Life (BOL) and End of Life (EOL);
Low power	– BOL and EOL;
Full power	– BOL and EOL.

2

The effects of the loss of primary system integrity as a result of the failed control rod housing should be included in the analysis. It should also be shown that failure of one control rod housing will not lead to failure of other control rod housings.

The approach that should be used in the radiological analysis of a control rod ejection accident is to determine the amount of each gaseous radionuclide released to the primary containment and, with this information in conjunction with the procedures set forth in Appendix B of this guide, to determine the radiological consequences of this accident for a pressurized water reactor.

C. REGULATORY POSITION

Acceptable assumptions and evaluation models for analyzing a rod ejection accident in PWRs are presented in Appendices A (Physics and Thermal-Hydraulics) and B (Radiological Assumptions) of this guide. By use of these appendices, it should be shown that:

1. Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location in any fuel rod. 3

2. Maximum reactor pressure during any portion of the assumed transient will be less than the value that will cause stresses to exceed the Emergency Condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.[1]

3. Offsite dose consequences will be well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria."

[1] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, New York 10017.

APPENDIX A

PHYSICS AND THERMAL-HYDRAULICS

The assumptions described below should be applied in evaluating the physics and thermal-hydraulic behavior of the reactor system for a control rod ejection accident.

1. The ejected rod worth should be calculated based on the maximum worth rod resulting from the following conditions: (a) all control banks at positions corresponding to values for maximum allowable bank insertions at a given power level and (b) additional fully or partially inserted misaligned or inoperable rod or rods if allowed by operating procedures. Sufficient parametric studies should be performed to determine the worth of the most reactive control rod in each rod group for different control rod configurations, both expected and unexpected. The worth of single rods in rod groups should be evaluated during startup physics tests and compared with values used in the rod ejection analysis. The accident should be reanalyzed if the rod worths used in the initial analysis are found to be nonconservative. Calculated rod worths should be increased, if necessary, to account for calculational uncertainties in parameters such as neutron cross sections and power asymmetries due to xenon oscillations.

2. The reactivity insertion rate due to an ejected rod should be determined from differential control rod worth curves and calculated transient rod position versus time curves. If differential rod worth curves are not available for the reactor state of interest, conservatism should be included in the calculation of reactivity insertion through consideration of the nonlinearity in reactivity addition as the rod passes through the active core. The rate of ejection should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction.

3. The calculation of effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (ℓ^*) should be based on the well-known definitions resulting from perturbation theory, such as those described by Henry (Ref. 1), using available experimental delayed neutron data and averaging by the fraction of fission in the various fissionable materials. In cases where the accident is quite sensitive to β_{eff} (where the ejected rod worth $\geq \beta_{eff}$), the minimum calculated value for the given reactor state should be used. For smaller transients, conservatism in the value should include consideration of not only the initial power rise (which increases with decreasing β), but also the power reduction after the trip. Similar considerations should also be applied to determine an appropriately conservative value of ℓ^* to be used.

4. The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen with respect to their influence on the magnitude of the transient. Pressure and temperature are mainly significant with respect to their effect on the amount of reactivity inserted if there exists a positive moderator coefficient.

5. The fuel thermal properties such as fuel-clad gap heat transfer coefficient and fuel thermal conductivity should be conservatively chosen, depending upon the transient phenomenon being investigated. For conditions of a zero or positive moderator coefficient (usually at beginning of life), for example, high heat transfer parameters would reduce the Doppler feedback and increase any positive moderator feedback effects and hence tend to increase the magnitude of the reactivity transient. For a negative moderator coefficient, high heat transfer parameters could cause the magnitude of the transient to decrease if a given quantity of heat produces more feedback in the moderator than in the fuel. In the consideration of pressure pulses which may be generated, high moderator heating rates could cause significant pressure gradients to develop in the moderator channels. In computing the average enthalpy of the hottest fuel pellet during the excursion for power cases, low heat transfer would be conservative.

6. The specific heat of UO_2 has been determined experimentally and is a deterministic factor in the calculated amount of stored energy (enthalpy) in the fuel. Recommended values in the range of 25 to 902°C are the data reported by Moore and Kelly (Ref. 2). In the range of 900 to 2842°C, the data obtained by Hein and Flagella (Ref. 3), Leibowitz, Mishler, and Chasanov (Ref. 4), and Chasanov (Ref. 5) are recommended for the heat capacity of the fuel. These recommended values are for clean core conditions. Possible variation in the specific heat due to burnup should be investigated and appropriate values used, if necessary.

7. The moderator reactivity coefficients due to voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If no three-dimensional space-time kinetics calculation is performed, the reactivity feedback due to these coefficients should be conservatively weighted to account for the variation in their spatial importance in the missing dimension(s). If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed.

8. The Doppler coefficient should be calculated based on the effective resonance integrals and should include corrections for pin shadowing (Dancoff correction). Calculations of the Doppler coefficient of reactivity should be based on and should compare conservatively

with available experimental data such as those of Hellstrand (Ref. 6). Since the Doppler coefficient reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting fuel temperatures at different power levels should be reflected by conservatism in the applied value of the Doppler coefficient. If no three-dimensional space-time kinetics calculation is performed, the reactivity effect of spatially weighting the core average temperature rise in both the axial and radial directions should be calculated.

9. Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. If the rod worth curve (reactivity vs. depth of insertion) is not obtained from a "true" representation (i.e., an x, y, z, t or an r, z, t calculation); the conservatism of the approximate calculation should be shown. The difference in the depth of insertion at zero power and at full power should be accounted for in calculating the available scram reactivity.

10. The reactor trip delay time, or the amount of time which elapses between the instant the sensed parameter (e.g., pressure or neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (a) time required for instrument channel to produce a signal, (b) time for the trip breaker to open, (c) time for the coil to release the rods, and (d) time required before scram rods enter the core if the tips lie above the core-reflector interface.

11. The computer code used for calculating the transient should be a coupled thermal, hydrodynamic, and nuclear model with the following capabilities: (a) incorporation of all major reactivity feedback mechanisms, (b) at least six delayed neutron groups, (c) both axial and radial segmentation of the fuel element, (d) coolant flow provision, and (e) control rod scram initiation on either coolant system pressure or neutron flux.

12. The analytical models and computer codes used should be documented and justified and the conservatism of the models and codes should be evaluated both by comparison with experiment, as available, and with more sophisticated spatial kinetics codes. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes should be investigated, and the 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. Also, sensitivity studies on variations of the Doppler effect, power distribution, fuel element heat transfer parameters, and other relevant parameters should be included.

13. The pressure surge should be calculated on the basis of conventional heat transfer from the fuel, a conservative metal-water reaction threshold, and prompt heat generation in the coolant to determine the variation of heat flux with time and the volume surge. The volume surge should then be used in the calculation of the pressure transient, taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer relief and safety valves. No credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

14. The number of fuel rods experiencing clad failure should be calculated and used to obtain the amount of contained fission product inventory released to the reactor coolant system. It should be assumed that clad failure occurs if the heat flux equals or exceeds the value corresponding to the onset of the transition from nucleate to film boiling (DNB), or for other appropriate causes.

The margin to DNB is expressed in terms of a departure from nucleate boiling ratio (DNBR). The DNBR at any position in the hottest channel is the ratio of the DNB heat flux to the actual heat flux. The DNB heat flux should be evaluated using correlations based on recognized studies and experimental heat transfer DNB data. A minimum DNBR should be determined from the evaluation of the experimental data to ensure a 95% probability with a 95% confidence level that DNB has not occurred for the fuel element being evaluated. One example of a correlation which has been used to date is given by Tong (Ref. 7). The use of this correlation and the above probabilities and confidence level yields a minimum DNBR of 1.30. Other DNB or clad failure correlations may be used if they are adequately justified by analytical methods and supported by sufficient experimental data.

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NUREG-0800
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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Accident Evaluation Branch (AEB)

I. AREAS OF REVIEW

The CPB evaluates the consequences of a control rod ejection accident in the area of physics. The review covers the possible initial conditions, rod patterns and worths, scram worth as a function of time, adequacy of the various reactivity coefficients, adequacy of the calculational methods, and any core parameters which affect the peak reactor pressure or the probability of fuel rod failure.

The relevant thermal-hydraulic analyses are reviewed under SRP Section 4.4.

The AEB reviews, as part of its secondary review responsibility, described in the appendix to this SRP section, the radiological consequences of a rod ejection accident by using a source term for dose calculations based on the amount of failed fuel as obtained by CPB from the reactor core analyses. The evaluation finding provided is as indicated in the attached Appendix.

The applicant's determination of the reactor trip delay time, i.e., the time elapsed between the instant the sensed parameter reaches the level for which protective action is required and the onset of negative reactivity insertion, is reviewed under SRP Sections 7.2 and 7.3.

II. ACCEPTANCE CRITERIA

CPB acceptance criteria are based on meeting the requirements of General Design Criterion 28 (Ref. 1) as it relates to the effects of postulated reactivity accidents neither resulting in damage to the reactor coolant pressure boundary greater than limited local yielding, nor causing sufficient damage to impair significantly the capacity to cool the core.

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Regulatory Positions and specific guidelines necessary to meet the relevant requirements of GDC 28 are in Regulatory Guide 1.77.

Regulatory Guide 1.77 (Ref. 2) identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a control rod ejection accident. Specific criteria used by CPB in evaluating the control rod ejection accident are:

a. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod. (1)

b. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code (Ref. 3).

c. The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling (DNB) condition is an input to the radiological evaluation by AEB. The radiological criteria used in the evaluation of control rod ejection accidents (PWRs) are given in Appendix B of Regulatory Guide 1.77 (Ref. 2).

III. REVIEW PROCEDURES

1. Review of the applicant's analyses, showing that the first of the acceptance criteria above is met, proceeds as follows:

   a. A spectrum of initial conditions is considered, which must include both zero-power and full-power conditions, at beginning and end of a reactor fuel cycle (BOC and EOC), to assure examination of upper bounds on possible fuel damage. Initial full-power conditions should include the uncertainties in the calorimetric measurement of power.

   b. From the initial conditions of (a) and from control rod patterns, the limiting rod worth is determined. Where confirmation is considered necessary the reviewer may calculate, as an audit, the worth of limiting rods.

   c. Reactivity coefficient values corresponding to the limiting initial conditions must be used at the beginning of the transient. The reviewer checks the reactivity coefficient curves used by the applicant with those reviewed by the CPB under SRP Section 4.3. The two coefficients of most interest are the Doppler and moderator coefficients. If no three-dimensional space-time calculation is performed, the reactivity feedback must be conservatively weighted to account for the variation in the missing dimension(s). (2)

   d. The reviewer inspects the control rod insertion assumptions which include: trip parameters, trip delay time, rod velocity curve, and differential rod worth. Trip parameters and delay time are reviewed under SRP Section 7.2. Control rod worth is checked by the reviewer for consistency with the review performed under SRP Section 4.3. (3)

   e. The applicant's analytical methods are reviewed. The reviewer may use the results of previous case work, if the analytical methods have been previously reviewed and approved by the staff. Otherwise

he must perform a de novo review on this case. Alternatively an audit of several calculations, using methods considered acceptable to the staff, may be done by the reviewer (or consultants to the staff). The primary concern of the reviewer is how well the elements of the analytical model represent the true three-dimensional problem. Other items checked by the reviewer include feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.

f. Results of the calculations done by procedures described in steps a-e are expressed as values of the radially-averaged fuel rod enthalpy (in units of cal/gm). The reviewer determines that the maximum value does not exceed 280 cal/gm. (4)

2. Verification of compliance with the second acceptance criterion is accomplished as follows:

   a. The same procedures considered in steps a-f above are followed.

   b. For each accident, the maximum primary system pressure should be calculated by an analytical method acceptable to the staff or, as before, an independent audit calculation is made by the staff. The reviewer checks the results (as obtained by the applicant or the staff) for compliance with the second criterion.

3. The number of fuel rods experiencing clad failure is determined (for use in evaluating the radiological consequences) by the following procedure:

   a. The reviewer determines that an acceptable procedure for calculating a departure from nucleate boiling condition during the reactivity excursion has been used. This may be done by referring to previous cases for the same nuclear steam supply system (NSSS) vendor. If no approved technique is available, as might be the case for the first project using a new or substantially revised model, the reviewer must perform a separate detailed review (which may be documented separately in a topical report).

   b. The reviewer must determine that the number of rods used in the radiological evaluation is the number of rods calculated to have a departure from nucleate boiling. Departure from nucleate boiling must be calculated in accordance with the criteria reviewed and accepted under SRP Section 4.4. Typically, the criterion defines a departure from nucleate boiling ratio (DNBR) less than 1.30 when DNB correlations such as W-3 (Ref. 4) or BAW-2 (Ref. 5) are used.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

The staff concludes that the analysis of the rod ejection accidents is acceptable and meets the requirements of General Design Criterion 28. This conclusion is based on the following:

The applicant met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding, or cause sufficient damage that would significantly impair the capability to cool the core. The requirements have been met by demonstrating that the regulatory positions of Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWR's" are complied with. The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum control rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

V. IMPLEMENTATION

The following section is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP Section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method described herein are contained in the referenced regulatory guide.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 28, "Reactivity Limits."

2. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

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