Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors Summary and Qualification

ABB Combustion Engineering Nuclear Operations

ABB

CENPD-284-NP-A
(RPA 89-112-NP-A)
(RPA 89-053-NP-A)

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LEGAL NOTICE

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July 1996

ABB Combustion Engineering Nuclear Operations

CENPD-284-NP-A, RPA 89-112-NP-A, and RPA 89-053-NP-A REPORT CONTENTS

CENPD-284-NP-A, RPA 89-112-NP-A, and RPA 89-053-NP-A REPORT

Part I

NRC Acceptance Letter, Safety Evaluation Report (SER), and Technical Evaluation Report (TER)

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

April 12, 1996

Mr. Derek Ebeling-Koning, Manager
Licensing and Safety Analysis
ABB Combustion Engineering Nuclear Operations
P. O. Box 500
Windsor, CT 06095-0500

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORTS; CENPD-284-P, "CONTROL ROD DROP ACCIDENT ANALYSIS METHODOLOGY FOR BOILING WATER REACTORS: SUMMARY AND QUALIFICATION" (TAC NO. M88025); RPA-89-112, "ABB ATOM CONTROL ROD DROP ACCIDENT ANALYSIS METHODOLOGY FOR BOILING WATER REACTORS: THE RAMONA-3B COMPUTER CODE" (TAC No. M75965); AND RPA-89-053, "ABB ATOM HIGH WORTH CONTROL RODS FOR US BWRs: ROD DROP ACCIDENT ANALYSIS" (TAC NO. M75966)

Dear Mr. Ebeling-Koning:

The staff has reviewed the above topical reports submitted by ABB Combustion Engineering by letters dated October 1, 1993 and January 31, 1990. As described in the enclosed safety evaluation report (SER), these reports are acceptable for referencing in license applications to the extent specified and under the limitations stated in the enclosed Brookhaven reports and U.S. Nuclear Regulatory Commission (NRC) safety evaluation. The evaluation defines the basis for the staff's acceptance of the report.

The staff will not repeat its review of the matters described in the topical reports and found acceptable when they appear as a reference in license applications, except to assure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the topical reports and associated responses to questions. In accordance with procedures established in NUREG-0390, the NRC requests that ABB Combustion Engineering publish accepted versions of the above reports, proprietary and non-proprietary, within 3 months of receipt of this letter. These should include the information supplied to the NRC in response to requests for additional information. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract, and an -A (designating accepted) should follow the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion that the report is acceptable is invalidated, ABB Combustion Engineering and any applicant referencing the topical report will be expected to revise and resubmit the respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Robert Jones

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure:
Evaluation for CENPD-284-P, RPA-89-112, and RPA-89-053

SAFETY EVALUATION FOR TOPICAL REPORTS

CENPD-284-P, "CONTROL ROD DROP ACCIDENT ANALYSIS METHODOLOGY FOR BOILING WATER REACTORS, SUMMARY AND QUALIFICATION," OCTOBER 1993.

RPA-89-112, "ABB ATOM CONTROL ROD DROP ACCIDENT ANALYSIS METHODOLOGY FOR BOILING WATER REACTORS," NOVEMBER 1989.

RPA-89-053, "ABB ATOM HIGH-WORTH CONTROL RODS FOR US BWRs ROD DROP ACCIDENT ANALYSIS," AUGUST 1989.

1.0 INTRODUCTION

By letters dated October 1, 1993 (Ref. 1) and January 31, 1990 (Ref. 2) ABB Combustion Engineering Nuclear Operations (ABB-CE), or the preceding organization, as indicated in the references, submitted the above topical reports for review. These reports describe and justify the methodology proposed to be used by ABB-CE to select and analyze the control rod drop (CRD) events required to be examined for a boiling water reactor reload safety review. These reports are closely related and cover the various aspects of the methodology, results and criteria for calculating the limiting CRD events. These reports have been, for the most part, reviewed together by the staff and consultants. Requests for additional information (RAI) and the responses by ABB-CE have covered all three reports as a group. This safety evaluation and the attached consultants' reports will also address the three reports together.

The NRC contractor, Brookhaven National Laboratory (BNL) helped the staff review these topical reports. BNL has written two technical evaluation reports (TERs) that are included in this report as Attachments 1 and 2. They are addressed to the two submittals indicated in References 1 and 2. The TERs provide a detailed discussion of the significant elements of the methodology presented by ABB-CE, and an evaluation of this methodology. The details will not be further discussed here. The staff has reviewed the TERs and adopts their analyses and conclusions.

2.0 EVALUATION

As described in the attachment, the ABB-CE methodology uses well known and widely used computer codes for the calculations. Report CENPD-284-P describes the ABB-CE modeling of the CRD event and the computer codes and their benchmarking. Report RPA-89-112 describes the methodology for performing design basis rod drop analysis for BWR 4-6 plants using the Banked Position Withdrawal Sequence. This methodology is applied to the case of high-worth control rods in the Report RPA-89-053.

The transient analyses are done with the RAMONA-3B-SCP2 version of the widely used (including NRC and BNL) RAMONA code. The various input and auxiliary calculations are done with the equally widely used PHOENIX lattice physics and POLCA steady state simulator codes.

As described in the attachments, the staff and consultants review has determined that this proposed methodology is, with several exceptions or reservations, acceptable for BWR CRD analysis. The problem areas are discussed in Section 3 of the two TERs and the exceptions and reservations are listed and summarized in these TERs in Sections 4, "Technical Position." The reservations are primarily restrictions or requirements to evaluate or justify procedures or parameters of some parts of the analysis if certain specified choices are made. The technical positions indicate that, if certain conditions exist or approaches are used, the analyses are subject to conditions discussed in TER Section 3 and an evaluation or demonstration is required. These conditions and approaches have each been discussed with ABB-CE via RAIs by the staff and consultants and responses by ABB-CE (see Ref. 9 of Attachment 1). The positions stated in the TERs are compatible with the ABB-CE responses, including the indicated actions which would be taken. The documentation of safety analyses using the ABB-CE methodology should clearly indicate conformance to the conditions of the NRC approval.

The use of the STAV code (Ref. 3) to determine the fuel rod gap conductance must be in accordance with conditions of the Safety Evaluation Report approving the STAV methodology. It should be noted that the NRC review of STAV has been performed concurrently with the review of the ABB-CE control rod drop methodology and that the approval restricts use of the STAV code to the analysis of fuel with burnup no greater than 50 GWD/MTU.

The most significant restriction is the requirement that, at this time, because of the uncertainty in the rate of production of voids in this rapid CRD transient, the analysis should be conservatively calculated without moderator voids. The basis for this decision and the problems associated with the analysis of void formation are provided in the TER. As noted in the TER, this restriction does not preclude future exchange of information on this subject and the possibility of a future NRC approval of a void production model if sufficient justification is provided.

3.0 CONCLUSION

The staff and BNL consultants have reviewed the ABB-CE system of computer codes and processes for analyzing BWR control rod drop events as submitted in References 1 and 2. As discussed in the attached TERs (Section 3.0, "Summary of the Technical Evaluation," and 4.0, "Technical Position"), the review has concluded that the methodology presented in the three topical reports that are the subject of this review, and in the responses to staff RAI, is acceptable for use in BWR reload analysis, with, however, restrictions on the methodology as listed, described and evaluated in TER Sections 3 and 4. It is noted, in particular, that at this time, the use of void formation in the analysis is not acceptable, but this area will be considered for further review and possible acceptance if future submittals provide sufficient justification.

4.0 REFERENCES

1. "Transmittal for NRC Staff Review of CENPD-284-P, 'Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification,'" D. B. Ebeling-Koning (ABB-CE) to R. C. Jones (NRC), ATOF-93-105, dated October 1, 1993.

2. "Submittal of Topical Reports Describing ABB Atom Methods for Analyzing Control Rod Drop Accidents in BWRs," Letter, ABB-90-015, N. O. Jonsson to USNRC, dated January 31, 1990.

3. CENPD-285-P, "Fuel Rod Design Methods for Boiling Water Reactors," May 1994, and CENPD-287-P, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors."

Technical Evaluation Report

Topical Report Title: Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification

Topical Report Number: CENPD 284-P

Report Issue Date: October 1993

Originating Organization: ABB Combustion Engineering Nuclear Fuel

1.0 INTRODUCTION

By letter dated October 1, 1993 (Reference-1), ABB-CE has submitted the BWR Control Rod Drop Accident (RDA) Methodology Topical Report CENPD 284-P for NRC review and approval. The Topical Report describes the ABB-CE RDA methodology including the modeling and analysis procedures, the computer codes employed, and the analysis acceptance criteria. The methodology described in CENPD 284-P is based, in part, on the results provided in the ABB-Atom Topical Reports RPA 089-053 and RPA 089-112, and the limitations of the SER approvals for these Topical Reports apply, as appropriate, to the applications of the CENPD 284-P methodology. The RAMONA-3B-SCP2 code (References 2 and 3) used to perform the dynamic RDA analysis is described in RPA 089-112, and the application of RAMONA-3B-SCP2 to the case of high-worth control rods is described in RPA 089-053. The proposed CENPD 284-P methodology is intended for application to U.S. BWRs.

In the ABB-CE methodology, the PHOENIX lattice physics code (Reference-4) is used to calculate the nuclear data input for RAMONA-3B-SCP2 and the POLCA (Reference-1) three-dimensional simulator is used to determine the core fuel burnup and void-history distributions. The three-dimensional coupled neutronic/thermal-hydraulic analysis of the core transient is performed with RAMONA-3B-SCP2. In the CENPD 284-P methodology, a limiting control rod drop accident is defined which is intended to bound the actual cycle-specific RDA. A step-wise procedure is provided for evaluating the cycle-specific event. This procedure requires the evaluation of the predicted transient results against the 280 cal/gm fuel enthalpy limit for reactivity transients and the 170 cal/gm fuel failure threshold.

As qualification for the RAMONA-3B-SCP2 rod drop capability, ABB-CE has calculated six selected SPERT-III E-Core rod drop transients. These comparisons provide a quantitative demonstration of the consistency of the RAMONA-3B-SCP2 rod drop predictions.

The purpose of this review was to evaluate the acceptability of the proposed CENPD 284-P methodology for performing BWR control rod drop accident licensing analyses. This involved the evaluation of both the RDA methodology and the adequacy of the RAMONA-3B-SCP2 benchmarking. The ABB-CE methodology and benchmarking are summarized in Section-2, and the evaluation of the important technical issues raised during this review is presented in Section-3. The technical position is given in Section-4.

2.0 SUMMARY OF TOPICAL REPORT

2.1 Control Rod Drop Accident Calculational Model

In the ABB-CE RDA methodology, the detailed time-dependent core power distribution and local thermal-hydraulic feedback are calculated with the RAMONA-3B-SCP2 code. The RAMONA-3B-SCP2 code employs a standard one-and-a-half group neutronics scheme with six delayed neutron groups. The RAMONA-3B-SCP2 thermal-hydraulics solution is based on conservation equations for vapor mass, and mixture mass, energy and momentum. The core is represented by a set of representative parallel flow channels which are calculated using a closed contour momentum equation. The fuel rod heat conduction equations are solved using a radial finite difference model and the fuel-type dependent gap conductance includes a second order fuel temperature dependence. The RAMONA-3B-SCP2 systems model includes a recirculation loop (with a jet pump) and a steam line.

The RAMONA-3B-SCP2 fuel bundle dependent nuclear cross section input data is determined using the PHOENIX lattice physics code. PHOENIX uses a standard two-dimensional multigroup transport method to determine the rod-wise bundle power distribution, fuel isotopics and bundle reactivity. PHOENIX treats each rod in the fuel bundle explicitly and models the BWR cruciform control rods with cylindrical absorbers. The POLCA steady-state core simulator is used to determine the statepoint fuel burnup and void history distributions. The POLCA three-dimensional calculation uses a modified one-group diffusion theory solution to determine the nodal power distribution. The POLCA spatial model allows one node per bundle radially and twenty-five nodes axially. The POLCA model includes corrections to account for the inter-nodal coupling and the presence of adjacent fuel bundles. The dependence of the neutronics data on the local fuel burnup, fuel temperature, void fraction and void history is included in the POLCA model.

The control rod drop in a BWR results in a large increase in local reactivity and a substantial redistribution of the core power distribution during the course of the transient. In the ABB-CE RDA methodology, the dropped rod is modelled in detail with each fuel bundle typically represented by a single neutronics node and 25 axial nodes. The nuclear cross sections are provided for each node as a function of fuel burnup, fuel temperature, coolant density and void history. POLGEN provides the cross section fitting coefficients used to model the dependence on these local variables. The void feedback model allows for the dependence on the control state of the fuel bundle. The bundle rod-wise power distributions are precalculated by PHOENIX.

2.2 Control Rod Drop Accident Methodology

The RDA core power excursion and peak fuel enthalpy are determined, to a large extent, by the inserted control rod reactivity and the core Doppler feedback coefficient. In the ABB-CE methodology, a detailed evaluation is performed to identify the control rod which will result in the most severe consequences for the RDA. The POLCA three-dimensional core simulator is used to evaluate the reactivity worth of candidate rods throughout the cycle, in both the startup and power range. The expected control rod withdrawal patterns are evaluated observing the restrictions imposed by the Technical Specifications, Bank Position Withdrawal Sequence (BPWS) or group notch withdrawal sequence.

The RAMONA-3B-SCP2 dynamic analysis is performed for the control rods that are expected to have the most severe RDA consequences. A full core model is typically used to model off-center rods where asymmetric effects are important. In RAMONA-3B-SCP2 the dropped rod is modeled as a material boundary which moves at a constant velocity. All control rods, except for the dropped rod, are assumed to insert when a scram is initiated. The peak fuel enthalpy is calculated using the nodal power determined by RAMONA-3B-SCP2 and the precalculated local peaking factor. The fuel rod gap conductance and thermal conductivity are determined by the STAV ABB-CE fuel performance code (Reference-5).

As an example of the application of the methodology, ABB-CE presents a RDA calculation for a reload core operating with the BPWS. A conservative base case RDA transient together with a series of sensitivity calculations are presented. The sensitivity of the peak fuel enthalpy to the most important modeling and input assumptions is calculated for the base case transient. It is concluded that the base case results and calculated sensitivities are consistent with the analyses reported in References 6 and 7.

ABB-CE performs a cycle-specific evaluation for the RDA. This evaluation includes a systematic review of existing RDA results, and makes use of precalculated sensitivities and bounding calculations to envelope the worst-case consequences of the cycle-specific RDA.

2.3 Methods Qualification

As qualification for the RAMONA-3B-SCP2 RDA methodology, ABB-CE has performed a detailed analysis of the SPERT-III CORE-E rod drop transients with RAMONA-3B-SCP2. These tests provide measurements of the power excursion and increase in energy that result from a control rod drop transient for a UO_2 fueled core.

CORE-E consists of forty-eight 25-rod and twelve 16-rod fuel assemblies arranged in an approximately cylindrical pattern having an effective diameter of ~26 in. The core active fuel height is ~38 in. and the fuel enrichment is 4.8 w/o. The centrally located transient-rod is cruciform in shape with a square poison section made of stainless steel containing 1.35 w/o B-10. The power excursion was initiated by dropping the central transient control rod. The tests calculated by ABB-CE were at cold and hot startup conditions.

The RAMONA-3B-SCP2 analysis included neutronic, thermal-hydraulic and fuel rod models that were constructed in a manner analogous to that used for BWR calculations. Comparisons of the RAMONA-3B-SCP2 predictions and the SPERT tests are presented for the transient peak power, energy up to the time of peak power, and time to peak power. These comparisons indicate that for transients having peak powers consistent with measurement, the prediction of integrated energy, transient power shape, and time-to-peak power is within the measurement uncertainty.

3.0 SUMMARY OF THE TECHNICAL EVALUATION

The Topical Report CENPD 284-P provides a detailed description of the ABB-CE methodology for performing the Chapter-15 design basis control rod drop analysis for BWR plants. The review focused on the applicability and conservatism of the methods used for modeling the reactor transient and determining the peak fuel enthalpy. This review does not, however, include those issues related to the recent measurements of fuel rod behavior at high burnup. Several important technical issues were identified during the initial review which required additional information and clarification from ABB-CE. This information was requested in Reference-8 and was provided in the ABB-CE response included in Reference-9. This evaluation is based on the description and examples presented in the topical report and the supporting information provided in Reference-9. The evaluation of the major issues raised during this review are summarized in the following.

3.1 Control Rod Drop Accident Calculational Model

The BWR RDA power transient is limited by the Doppler and void reactivity feedbacks. In RDA licensing transients involving large control rod reactivity worths, the peak fuel enthalpy is sensitive to the void reactivity feedback and the void dependence of the heat transfer from the fuel rod. The effect of increased voids on both the reactivity feedback and the fuel rod heat transfer decreases the accident peak fuel rod enthalpy. In fact, the comparisons provided in the ABB-CE Topical Reports CENPD-284-P and RPA-89-053 demonstrate the substantial reduction in the calculated peak fuel enthalpy that results from the presence of moderator voids in the RDA. The moderator voids in the RDA result from direct moderator heating and fuel rod heat conduction, and the subsequent production of voids under highly transient conditions. In a typical prompt-critical RDA the core power increases by more than a decade every ~25 msec, and there is a substantial degree of uncertainty in the magnitude and timing of the transient void production. In fact, previously accepted RDA methodologies take credit for the Doppler feedback but conservatively calculate the core power transient assuming no moderator voids.

The RAMONA-3B-SCP2 models that are used to calculate the void generation rate assume steady-state conditions, and have been adjusted and validated using comparisons to steady-state conditions. The applicability of these models and their validation are of concern since: (1) any delay in the generation of voids (resulting, for example, from period-dependence of the void generation rate, moderator superheating or heat transfer) will result in a substantial increase in the peak fuel enthalpy and (2) the steady-state void generation is dominated by the contribution from the voids produced at the wall, while the voids produced in the bulk coolant away from the wall are expected to make a substantial contribution to the void generation rate during the RDA.

In response to these concerns, in Reference-9 ABB-CE has updated the RAMONA-3B-SCP2 fuel rod heat transfer model to account for the very rapid time dependence of the RDA power excursion. In addition, ABB-CE has incorporated a bubble growth model in RAMONA-3B-SCP2 to account for the time dependence of the coolant void generation. In this revised void generation model, a minimum time delay has been incorporated to ensure that voids are not produced until the RDA transient power has decreased to a preselected fraction of the transient peak power.

Based on a review of the RAMONA-3B-SCP2 modeling changes described in Reference-9, it is concluded that the information that has been provided is not sufficient to justify taking credit for the effects of coolant voids in the RDA analysis. The specific areas of concern include the following.

1) Fuel Rod Transient Boiling Heat Transfer The fuel rod heat transfer is extremely sensitive to the amount of voids being produced at the fuel rod surface. The void generation rate depends on the specific surface conditions, the number of initially available unflooded nucleation sites and the subcooling history. The information provided in Reference-9 is not sufficient to demonstrate the applicability of (1) the proposed transient heat transfer model and (2) the cited experimental data to the conditions present during the RDA.

2) Void Production The void production rate depends on the number of initially available unflooded nucleation sites, the rate of activation of the flooded nucleation sites during the transient and the subcooling history. The description of the void production model provided in Reference-9 does not indicate how this dependence is included.

3) RAMONA-3B-SCP2 Model A more detailed description of the revised void production model and its implementation is required. For example, the Reference-9 description does not indicate how the differences in the wall and bulk coolant (1) temperatures and (2) available nucleation sites will be included in the void growth model.

4) Delay in Void Production The model described in Reference-9 includes a time delay in the production of the voids produced during the RDA transient. The basis for the method used to determine this time delay has not been provided.

While the description of the proposed RAMONA-3B-SCP2 void production model provided in Reference-9 is not considered sufficient to provide the basis for applying RAMONA-3B-SCP2 in RDA licensing calculations, this does not preclude a future NRC approval of these models if the necessary justification is provided by ABB-CE.

In Response-C4 (Reference-9), ABB-CE indicates that the reactor scram will be modeled using bounding values for the scram velocity and time delay. The scram worth will include an explicit allowance for uncertainty (at the 95% probability level), or a conservative bounding scram worth will be determined.

The RDA analysis of CENPD-284-P assumes a linear scram insertion which overestimates the magnitude of the initial negative reactivity insertion. In Response-B7 (Reference-9), ABB-CE has indicated that typical licensing calculations initiated with rod-worths based on startup rod patterns are not sensitive to the scram insertion. These transients are terminated by Doppler feedback prior to the scram insertion. However, in the case of very high rod-worths and large inlet subcooling, the Doppler feedback is not generally sufficient by itself to provide a prompt and complete reversal of the RDA transient. In this case, the RDA peak fuel enthalpy is sensitive to the scram reactivity and the assumption of a linear scram insertion. Consequently, in RDA licensing analyses involving very high rod-worths and large inlet subcoolings, the nonconservatism introduced by the assumption of a linear scram should be evaluated and, if necessary, accounted for in the determination of the peak fuel rod enthalpy.

3.2 Applications of the Control Rod Drop Methodology

The increase in the peak fuel rod enthalpy and Doppler reactivity feedback following the rod drop are determined, in part, by the pellet-to-clad gap conductance. ABB-CE has indicated that it may use the STAV fuel performance code to determine the gap conductance. Since STAV is presently being reviewed by the NRC, the use of STAV in RAMONA-3B-SCP2 RDA licensing analyses is contingent on the approval of STAV.

In the CENPD-284-P RDA methodology, the PHOENIX lattice physics code is used to determine the fuel assembly cross sections and kinetics data for input to both POLCA and RAMONA-3B-SCP2. POLCA is used for identifying the highest-worth rod. In Response-C9 (Reference-9), ABB-CE has indicated that both PHOENIX and POLCA have been reviewed and approved by the NRC for application to ABB-CE fuel designs (Reference-4). ABB-CE has indicated that for non-ABB fuel, the accuracy of the PHOENIX/POLCA code system will be demonstrated by comparisons to previous cycle core reactivity and power distribution measurements.

The RDA is a highly localized transient involving large time dependent bundle-to-bundle variations in the inserted reactivity, Doppler and moderator feedback and fuel enthalpy. The reliable prediction of the core power transient and peak fuel enthalpy requires a detailed assignment of the thermal-hydraulic channels to the individual fuel bundles. The steady-state

POLCA calculations are performed with a unique thermal-hydraulic channel assigned to each fuel bundle. ABB-CE has indicated in Response-C5 (Reference-9) that, in RAMONA-3B-SCP2 RDA licensing calculations, each fuel bundle will be represented by a unique thermal-hydraulic channel, or the predictions made by combining channels will be shown to be conservative or insensitive to this approximation.

The rate of reactivity insertion in the RDA is determined by the rod drop speed. The RDA analyses provided in CENPD-284-P assume a maximum rod drop velocity of 3 ft/sec, which results in a conservatively bounding RDA analysis. In Response-C12 (Reference-9), ABB-CE has indicated that a lower value of the rod drop speed will only be used when adequate justification can be provided.

The RDA is a prompt critical transient resulting in a strong exponential increase in the core power and complex thermal-hydraulic feedbacks, and is sensitive to the various modeling and input uncertainties. ABB-CE has indicated, in Response-A19 (Reference-9), that the RDA licensing analysis will include a detailed uncertainty analysis. This analysis will account for uncertainties (at the 95% level) in the power distribution, feedback reactivity, gap conductance, scram reactivity and kinetics parameters. (In certain cases, ABB-CE will use a conservative bounding gap conductance rather than include it in the uncertainty analysis.) The uncertainty in peak fuel enthalpy will be determined using calculated RAMONA-3B-SCP2 sensitivities together with estimated modeling and input uncertainties.

The RDA power transient and fuel enthalpy increase are also sensitive to the inserted dropped rod reactivity. The dropped rod reactivity is determined, in part, by the control rod pattern for the initiating RDA statepoint. Control rod patterns which increase the local power at the dropped rod location increase the reactivity worth of the dropped rod. In Response-C2 and B6 (Reference-9), ABB-CE indicates that the selection of the highest-worth control rod will account for the worst-case single equipment malfunction, operator error, and the maximum number of bypassed rods allowed by the plant Technical Specifications and licensing basis.

The results of the rod drop analysis have a substantial dependence on the initial reactor statepoint. The inserted rod-worths are larger for the low-power rod patterns, and the relative magnitudes of the Doppler and void feedbacks depend on the initial power level. In Responses-B5 and C17 (Reference-9), ABB-CE has indicated that extensive sensitivity studies have been performed to evaluate the statepoint sensitivity of the RDA peak fuel enthalpy. For critical control rod patterns, the low-power/high-inlet-subcooling statepoints tend to result in the maximum peak fuel enthalpies. Based on these sensitivity calculations, ABB-CE has identified a conservative worst-case initial RDA reactor statepoint. ABB-CE has also indicated that, if in licensing analyses it is not apparent that this statepoint is limiting, additional calculations will be performed to determine the worst-case statepoint.

In the CENPD-284-P methodology, the need for an actual RAMONA-3B-SCP2 dynamic calculation is determined by comparison of the total rod-worth and nodal peaking factor for the cycle-specific and precalculated "bounding" RDA. It is indicated in Response-C19 (Reference-9) that, if these comparisons indicate that the bounding RDA is more severe than the cycle-specific RDA and neither the cycle-specific fuel design or plant conditions have changed in a nonconservative direction relative to the bounding analysis, the bounding analysis applies and a RAMONA-3B-SCP2 dynamic analysis is not required. However, since the rod-worth and nodal peaking comparisons do not always ensure that the bounding RDA is in fact limiting, it should also be verified that changes in other parameters having a significant effect on the RDA (given in Response-A19 and Attachment A-19-1) have not made the cycle-specific RDA more limiting than the precalculated bounding RDA.

In the CENPD-284-P methodology, the cycle-specific test used to determine the limiting or bounding RDA employs a comparison of total rod-worths. However, the consequences of the RDA depend on both the total inserted reactivity worth and the rate of reactivity insertion. For example, an RDA resulting from a control rod drop from full-out to the position of the drive mechanism located at the core midplane results in a more severe RDA than when the rod drops from full-out to full-in, assuming the same rod drop speed and total inserted rod-worth. In order to ensure equal rod-worths and reactivity insertion rates, when determining the limiting cycle-specific RDA, the RDA comparisons should be made for cases in which the rod drops at the same speed and over the same axial span.

4.0 TECHNICAL POSITION

The ABB-CE control rod drop analysis Topical Report CENPD 284-P and supporting documentation provided in Reference-9 have been reviewed in detail. Based on this review, it is concluded that the ABB-CE control rod drop methodology is acceptable for performing BWR reload licensing analyses, subject to the conditions stated in Section-3 of this evaluation and summarized in the following.

1) Effect of Moderator Voids

Because of the present uncertainty in the rate of void production during the initial power transient, RDA licensing analyses should be conservatively calculated without moderator voids. While this submittal does not provide a sufficient basis for applying RAMONA-3B-SCP2 in RDA licensing calculations, this does not preclude a future NRC approval of these models if the necessary justification is provided by ABB-CE (Section-3.1).

2) Linear Scram Insertion

In RDA licensing analyses involving very high rod-worths and large inlet subcoolings, the nonconservatism introduced by the assumption of a linear scram should be evaluated and, if necessary, accounted for in the determination of the peak fuel rod enthalpy (Section-3.1).

3) STAV Fuel Performance Code

ABB-Atom determines the fuel rod gap conductance using the STAV fuel performance code. STAV must receive NRC approval prior to its use in RAMONA-3B-SCP2 RDA licensing analyses (Section-3.2).

4) Application to Non-ABB Fuel

The accuracy of the PHOENIX/POLCA code system in applications involving non-ABB fuel must be demonstrated by comparisons to previous cycle measurements of the core reactivity and power distribution (Section-3.2).

5) Selection of Thermal-Hydraulic Channels

In RDA licensing calculations, each fuel bundle should be represented by a unique thermal-hydraulic channel, or the predictions made by combining channels should be shown to be conservative or insensitive to this approximation (Section-3.2).

6) Reduced Rod Drop Velocity

The RDA analyses described in CENPD-284-P assume a rod drop velocity of 3 ft/sec. The use of a lower (less conservative) rod drop speed in RDA licensing analyses will require additional justification (Section-3.2).

7) Calculation Uncertainty Allowance

In order to account for RAMONA-3B-SCP2 modeling and input uncertainties, RDA licensing evaluations should include a detailed uncertainty analysis (Section-3.2).

8) Determination of Highest-Worth Rod

In RDA licensing analyses, the selection of the highest-worth control rod must account for the worst-case single equipment malfunction and operator error allowed by the plant Technical Specifications and licensing basis (Section-3.2).

9) Determination of Bounding Analysis

Since the rod-worth and nodal peaking comparisons do not always ensure that the bounding RDA is limiting, it should be verified that changes in other parameters having a significant effect on the RDA have not made the cycle-specific RDA more limiting than the precalculated bounding RDA (Section-3.2)

10) Limiting Cycle-Specific RDA

When determining the limiting cycle-specific RDA, in order to ensure equal rod-worths and reactivity insertion rates, the RDA comparisons should be made for cases in which the rod drops at the same speed and over the same axial span (Section-3.2).

References

1. "Transmittal for NRC Staff Review of CENPD 284-P, 'Control Rod Drop Accident Analysis Methodology For Boiling Water Reactors: Summary and Qualification,'" D. B. Ebeling-Koning (ABB-CE) to R.C. Jones (NRC), ATOF-93-105, dated October 1, 1993.
2. W. Wulff, et al., "A Description and Assessment of RAMONA-3B Mod. 0 Cycle 4: A Computer Code with Three Dimensional Neutron Kinetics for BWR System Transients," NUREG/CR-3664, 1984.
3. "User Manual for RAMONA-3B FMS Volume II," Scandpower FMS Document.
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TECHNICAL EVALUATION REPORT

Topical Report Titles:	ABB Atom Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors-RPA-89-112 ABB Atom High-Worth Control Rods for US-BWRs-Rod Drop Accident Analysis-RPA-89-053
Topical Report Numbers:	RPA-89-112 and RPA-89-053
Report Issue Dates:	RPA-89-112, November 1989 RPA-89-053, August 1989
Originating Organization:	ABB Atom Corporation

1.0 INTRODUCTION

In Reference-1, ABB Atom has submitted the Topical Reports RPA-89-112 and RPA-89-053 as documentation of their control rod drop accident methods and analyses for application to US boiling water reactors. The detailed ABB Atom rod drop accident (RDA) methodology is described in RPA-89-112 and the methodology is applied to the case of high-worth control rods in RPA-89-053. ABB Atom intends to apply these methods and analyses in determining the consequences of the RDA design basis event for the reload licensing analyses for US BWRs. The methodology described in RPA-089-053 and RPA-089-112 is based, in part, on the results provided in the ABB Atom Topical Report CENPD 284-P, and the limitations of the SER approval for this report apply, as appropriate, to the applications of the RPA-89-053 and RPA-89-112 methodology.

The primary acceptance criteria for the BWR control rod drop accident places a limit on the transient peak fuel rod enthalpy. Because of the large local reactivity effects and resulting strong radial and axial power peaking that occur during the RDA, the ABB Atom RDA methodology calculates the transient increase in fuel rod enthalpy using the RAMONA-3B-SCP2 three-dimensional coupled neutronics/thermal-hydraulics systems transient code. The Ramona-3B-SCP2 code (Reference-2) was initially developed by the Scandinavian Nuclear Research Institutes and by ABB Atom, and more recently by Scandpower International and by BNL. The nuclear cross section and kinetics data required by RAMONA-3B-SCP2 are calculated using the PHOENIX/POLCA code system (Reference-3), and are processed and fitted by POLGEN.

The Topical Reports provide a description of the benchmarking of Ramona-3B-SCP2 against plant measurement and test data, and the RAMONA-3B-SCP2 analysis of the RDA. Sensitivity calculations are presented for the RDA in which important input and modeling parameters are varied and the effects on the peak fuel rod enthalpy are determined. RDA calculations are presented for the case in which moderator feedback is included, and also for the more conservative case in which moderator feedback is neglected and the initial power transient is terminated by doppler feedback alone. RDA calculations are also performed for both standard and high-worth (1.15 $\Delta k/k$) control rods. For both the standard and high-worth control rods, the best-estimate calculations indicate that the peak fuel enthalpy during the RDA is well within the required 280 cal/gm limit.

The Topical Reports are summarized in the following Section-2, and the evaluation of the important technical issues raised during this review is provided in Section-3. The technical position is given in Section-4.

2.0 SUMMARY OF THE TOPICAL REPORTS

2.1 Rod Drop Analysis Methodology Topical Report RPA-89-112

2.1.1 Ramona-3B-SCP2 Methods

In the ABB Atom RDA methodology, the detailed time-dependent core power distribution and local thermal-hydraulic feedback are calculated with the SCP2 version of the Ramona-3B-SCP2 code. The Ramona-3B-SCP2 code employs a standard one-and-a-half group neutronics scheme with six delayed neutron groups. The Ramona-3B-SCP2 thermal-hydraulics solution is based on conservation equations for vapor mass, and mixture mass, energy and momentum. The core is represented by a set of representative parallel flow channels which are calculated using a closed-contour momentum equation. The fuel rod heat conduction equations are solved using a radial finite difference model and the fuel-type dependent gap conductance includes a second order fuel temperature dependence. The Ramona-3B-SCP2 systems model includes a recirculation loop (with a jet pump) and a steam line.

The Ramona-3B-SCP2 fuel bundle-dependent nuclear cross section input data is determined using the PHOENIX lattice physics code. Phoenix uses a standard two-dimensional multigroup transport method to determine the rod-wise bundle power distribution, fuel isotopics and bundle reactivity. PHOENIX treats each rod in the fuel bundle explicitly and models the BWR cruciform control rods with cylindrical absorbers. The POLCA steady-state core simulator is used to determine the statepoint fuel burnup and void history distributions. The POLCA three-dimensional calculation uses a modified one-group diffusion theory solution to determine the nodal power distribution. The POLCA spatial model allows one node per bundle radially and twenty-five nodes axially. The POLCA model includes corrections to account for the inter-nodal coupling and the presence of adjacent fuel bundles. The dependence of the neutronics data on the local fuel burnup, fuel temperature, void fraction and void history is included in the POLCA model.

2.1.2 Ramona-3B-SCP2 Qualification

The qualification of the Ramona-3B-SCP2 methodology includes comparisons to (1) special-effects thermal-hydraulic tests, (2) test-reactor experiments, and (3) plant measurement test data. The special-effects tests were used to validate the thermal-hydraulics and void modeling and the stability limits. The initial comparisons involved loop experiments (References 4 and 5) and the later benchmarking included 6 x 6 and 8 x 8 BWR fuel bundles (References 6-9). In addition to these special-effects benchmarks, Ramona-3B-SCP2 calculations have been compared to the SPERT-III Core-E reactivity accident tests (References 10 and 11). As detailed validation of the Ramona-3B-SCP2 model for calculating strong reactivity transients in a BWR, Ramona-3B-SCP2 calculations were compared to the Peach Bottom-2 turbine trip tests (Reference-12). The Ramona-3B-SCP2 predictions were in good agreement with the transient increase in the LPRM measurements for all three tests. Additional validation of the Ramona-3B-SCP2 models is provided by the comparisons of the predicted and measured power/flow oscillations observed during the CAORSO low-flow/high-power stability tests (Reference-13).

2.1.3 Control Rod Drop Modeling

The control rod drop in a BWR results in a large increase in local reactivity and a substantial redistribution of the core power during the course of the transient. In the ABB Atom RDA methodology, the dropped rod is typically located at the center of a one-eighth core model in which each fuel bundle is represented by a single radial neutronics node and 24 axial nodes. The nuclear cross sections are provided for each node as a function of fuel burnup, fuel temperature, coolant density and void history. POLGEN provides the cross section fitting coefficients used to model the dependence on these local variables. The void feedback model allows for the dependence on the control state of the fuel bundle. The bundle rod-wise power distributions are precalculated by PHOENIX.

The dropped control rod and scram rods may move at constant velocity or constant acceleration. The scram rods are activated by the APRM scram after an appropriate time delay. The fuel loading is selected so that the central control rod in the one-eighth core model represents the highest worth rod and its immediate surroundings. The dropped rod-worth, reactivity insertion rate and local power distribution are modeled conservatively.

The highest worth rod for the RDA analysis is determined by a series of POLCA rod-worth calculations. These calculations assume Banked Position Withdrawal Sequence (BPWS) operation (Reference-14) and each rod in every BPWS group is evaluated. ABB Atom performs the high-worth rod search at both cold and hot standby conditions.

2.1.4 RDA Sensitivity Calculations

In order to establish the accuracy and conservatism of the RDA calculations, ABB Atom has performed a series of sensitivity calculations for the rod drop analysis. The sensitivity calculations performed included: (1) the neglect of moderator feedback, (2) the neglect of heat transfer to the coolant, (3) an increase of rod drop speed from 3.11 to 5.11 ft/s, (4) an increase in control rod density from 25% to 50%, and (5) an increase in rod-worth from 1.174 to 1.426% ($\Delta k/k$). The calculations indicated that the effect of the rod drop velocity and control rod density are small, while the effects of the moderator density feedback, rod-worth and heat transfer are large and must be considered when assessing the accuracy and/or conservatism of specific RDA analyses.

2.2 Rod Drop Analysis for High-Worth Control Rods - Topical Report RPA-89-053

2.2.1 Rod Drop Analysis Model

The analysis model used for the evaluation of the effects of increasing the control rod-worth is essentially the same as described in RPA-89-112 and described in Section-2.1. The specific core used in the high-worth control rod (HWCR) analysis is a 532-bundle D-lattice beginning-of-cycle (BOC) equilibrium core. The core contained 8x8 BWR fuel with water rods, and with axial enrichments ranging from 0.71 to ~3.0 w/o%. The cross section data was generated for a reduced number of representative fuel types. The control rod-worths were determined via a static POLCA calculation without moderator or Doppler feedback. Six thermal-hydraulic channels were used to represent the one-eighth core geometry. The central dropped rod location had a unique thermal-hydraulic channel with the thermal-hydraulic channels becoming coarser closer to the core periphery.

Base cross sections at reference conditions were determined for all fuel types; however, the dependence on fuel temperature, moderator density and control was assumed to be the same for all fuel types. The core-average delayed neutron fraction was determined by importance weighting the exposure-dependent nodal values determined by PHOENIX.

2.2.2 Initial Conditions and Rod-Worths

The RDA initial conditions selected for comparison of the standard control rods (SCRs) and the HWCRs minimized moderator feedback. The calculations were performed at zero power conditions, with a subcooling of 80°C and a vessel pressure of 1.0 bar. The initial radial and axial power distributions are presented for the SCRs and the HWCRs, and indicate slightly more peaking for the HWCRs. The SCR worth is calculated to be very close to the generic value of 1.2% $\Delta k/k$ corresponding to the maximum number of inoperable rods (Reference-14). The high-worth control rod is ~1.4% $\Delta k/k$.

2.2.3 RDA Comparisons for the Standard and High-Worth Control Rods

Detailed comparisons of the RDA calculations for the SCRs and HWCRs are presented in RPA-89-053. The comparisons include the transient reactivity components, channel inlet and outlet flows, void fraction, core power, and peak fuel rod enthalpy. The calculations were performed both with and without moderator density feedback. An additional set of adiabatic calculations were also performed in which there was no heat transfer from the fuel to the coolant.

The calculations with moderator feedback included were carried out past the power excursion, but not to the time at which the peak fuel rod enthalpy occurs. The peak fuel enthalpy for these calculations was determined by an extrapolation based on the calculations without feedback.

As expected, the peak fuel rod enthalpy is significantly higher in the case of the HWCRs. The calculations indicate a substantial decrease in fuel rod enthalpy when moderator density feedback is included, and an even larger decrease in enthalpy when the fuel-to-coolant heat transfer is included. When the fuel-to-coolant heat transfer is included, both the SCR and HWCR RDA calculations indicate a peak fuel rod enthalpy with a large margin to the 280 cal/gm limit.

3.0 SUMMARY OF THE TECHNICAL EVALUATION

The Topical Report RPA-89-112 provides a detailed description of the ABB Atom methodology for performing the Chapter-15 design basis rod drop analysis for BWR/4-6 plants operating with the Banked Position Withdrawal Sequence. RPA-89-053 applies this methodology to the case of high-worth control rods. The review of these reports focused on the applicability and conservatism of the methods used for modeling the reactor transient and determining the peak fuel enthalpy. This review does not, however, include those issues related to the recent measurements of fuel rod behavior at high burnup. Several important technical issues were identified during the initial review which required additional information and clarification from ABB Atom. This information was requested in Reference-15 and was provided in the ABB Atom response included in Reference-16. This evaluation is based on the description and examples presented in the topical reports and the supporting information provided in Reference-16. The evaluation of the major issues raised during this review are summarized in the following.

3.1 Rod Drop Analysis Methodology Topical Report RPA-89-112

3.1.1 Ramona-3B-SCP2 Methods

In the RPA-89-112 rod drop methodology, the ABB Atom PHOENIX/POLCA system is used to perform the neutronics analysis. PHOENIX is used to calculate the nodal cross sections and kinetics data, and POLCA is used to calculate the core power distribution and control rod and scram worths. The same PHOENIX neutronics data is used for both the Ramona-3B-SCP2 and POLCA calculations. In Responses-A1 and C9 (Reference-16), ABB Atom has indicated that both PHOENIX and POLCA have been reviewed and approved by the NRC for application to ABB Atom fuel designs. ABB Atom has also indicated that for non-ABB fuel, the accuracy of the PHOENIX/POLCA code system will be demonstrated by comparisons to previous cycle core reactivity and power distribution measurements.

The Ramona-3B-SCP2 code version used in performing the rod drop analysis includes several methods improvements. The nodal cross section description has been modified to insure agreement with the steady-state calculation. The capability to model: (1) a non-equilibrium xenon distribution and (2) the nodal fuel design and exposure dependence of the delayed neutron fraction has been incorporated. The PRESTO (Reference-17) thermal flux nodal-coupling methodology has also been added to the flux calculation. This PRESTO neutronics model has been approved for steady-state application in Reference-17.

The ENDFB/V data-set resulted in a decrease in the delayed neutron fraction, β, and an increase in the transient power resulting from an RDA. In Response-A3 (Reference-16), ABB Atom indicates that the value of β used in the PHOENIX calculation is less (more conservative) than the ENDFB/V value. In addition, ABB Atom has performed a series of sensitivity calculations which indicate that the RDA is relatively insensitive to β, and that a 10% increase in β results in a small decrease in the RDA peak fuel enthalpy.

The Doppler fuel temperature feedback is the primary feedback in limiting the power excursion in the highly subcooled licensing RDAs. The coolant density and related spectrum moderation have a substantial effect on the strength of the Doppler feedback. In Response-A16 (Reference-16), ABB Atom indicates that the nodal cross section representation used in RAMONA-3B-SCP2 takes explicit account of this Doppler/moderator-temperature dependence.

3.1.2 Ramona-3B-SCP2 Qualification

The BWR rod drop analysis requires a detailed three dimensional spatial kinetics calculation to determine the time-dependent flux at the dropped rod location. The control rod reactivity and insertion rate are extremely sensitive to the neutron flux at the dropped rod location. In addition, the transient peak fuel enthalpy is determined by the power distribution at the dropped rod location. As qualification for the RAMONA-3B-SCP2 three dimensional spatial neutronics calculation, in Response-A17 (Reference-16), ABB Atom has provided detailed comparisons of the RAMONA-3B-SCP2 and POLCA steady-state core calculations. The RAMONA-3B-SCP2 flux solution is based on the PRESTO methodology, while the POLCA flux solution was developed independently and has been approved for steady-state applications. The RAMONA-3B-

SCP2/POLCA comparisons include: (1) axial and radial power distributions (2) total reactivity worth (3) dropped rod and scram reactivity worth, and (4) the hot-to-cold reactivity defect at zero-power conditions. The comparisons indicate generally good agreement consistent with the differences in the two methods.

In order to verify the PHOENIX Doppler feedback calculation, ABB Atom has compared PHOENIX and MCNP-3A Monte Carlo fuel rod Doppler coefficient predictions as a function of U-235 enrichment. The comparisons were made for a 300°C increase in fuel temperature for an infinite array of pin cells. No changes were made in the problem geometry in order to isolate the effects of the neutron transport and the cross section libraries. The comparisons provided in Response-B12 indicate generally good agreement over a full-range of U-235 weight percent and validate the PHOENIX doppler coefficient calculation.

3.1.3 Rod Drop Accident Applications

In RDA licensing applications, the RAMONA-3B-SCP2 modeling of the core will be the same as used in the approved POLCA applications. The same radial and axial nodal representation and fuel-type modeling will be used. Each fuel assembly is modeled as a unique hydraulic channel in POLCA, and will generally be modeled as a single hydraulic channel in RAMONA-3B-SCP2. Several fuel assemblies will be combined into a single hydraulic channel only when it is conservative, or it is shown to have a negligible effect on the RDA.

In a typical BWR reload, the highest worth control rod in the RDA analysis is generally not the central rod but is in an off-center location. However, an off-center rod drop requires a full-core model and results in a substantial increase in the RAMONA-3B-SCP2 computer running times. Consequently, a center rod drop which is conservative relative to the off-center case is used in many RDA licensing applications. However, in Response-A14 (Reference-16), ABB Atom has indicated that: (1) in general, the off-center rod will be located at the actual off-center location and modeled in full-core geometry and (2) an off-center rod will only be modeled as a center-rod when it can be shown to be conservative.

The fuel rod gap conductance affects the RDA peak fuel rod enthalpy directly through the fuel rod temperature and indirectly through the Doppler feedback. ABB Atom has performed sensitivity calculations which indicate that the direct effect on the fuel rod temperature is dominant and a minimum gap conductance is conservative for the RDA. In Response-A19 (Reference-16), ABB Atom has indicated that the uncertainty in the gap conductance will be accounted for in RDA licensing analyses by: (1) using a bounding minimum gap conductance or (2) performing an uncertainty analysis and including an additional margin in the calculated peak fuel rod enthalpy to account for the uncertainty in the gap conductance (at the 95% level).

ABB Atom determines the fuel rod gap conductance using the STAV fuel performance code. STAV is presently being reviewed by the NRC and the approval of the ABB Atom RDA methodology is contingent on the approval of STAV.

The BWR RDA power transient is limited by the Doppler and void reactivity feedbacks. In RDA licensing transients involving large control rod reactivity worths, the peak fuel enthalpy is sensitive to the void reactivity feedback and the void dependence of the heat transfer from the fuel rod. The effect of increased voids on both the reactivity feedback and the fuel rod heat transfer decreases the accident peak fuel rod enthalpy. In fact, the comparisons provided in the ABB-CE Topical Reports CENPD-284-P and RPA-89-053 demonstrate the substantial reduction in the calculated peak fuel enthalpy that results from the presence of moderator voids in the RDA. The moderator voids in the RDA result from direct moderator heating and fuel rod heat conduction, and the subsequent production of voids under highly transient conditions. In a typical prompt-critical RDA the core power increases by more than a decade every ~25 msec, and there is a substantial degree of uncertainty in the magnitude and timing of the transient void production. In fact, previously accepted RDA methodologies take credit for the Doppler feedback but conservatively calculate the core power transient assuming no moderator voids.

The RAMONA-3B-SCP2 models that are used to calculate the void generation rate, assume steady-state conditions, and have been adjusted and validated using comparisons to steady-state conditions. The applicability of these models and their validation are of concern since: (1) any delay in the generation of voids (resulting, for example, from period-dependence of the void generation rate, moderator superheating or heat transfer) will result in a substantial increase in the peak fuel enthalpy and (2) the steady-state void generation is dominated by the contribution from the voids produced at the wall, while the voids produced in the bulk coolant away from the wall are expected to make a substantial contribution to the void generation rate during the RDA.

In response to these concerns, in Reference-16 ABB Atom has updated the RAMONA-3B-SCP2 fuel rod heat transfer model to account for the very rapid time dependence of the RDA power excursion. In addition, ABB Atom has incorporated a bubble growth model in RAMONA-3B-SCP2 to account for the time dependence of the coolant void generation. In this revised void generation model, a minimum time delay has been incorporated to ensure that voids are not produced until the RDA transient power has decreased to a preselected fraction of the transient peak power.

Based on a review of the RAMONA-3B-SCP2 modeling changes described in Reference-16, it is concluded that the information that has been provided is not sufficient to justify taking credit for the effects of coolant voids in the RDA analysis. The specific areas of concern include the following.

1) Fuel Rod Transient Boiling Heat Transfer The fuel rod heat transfer is extremely sensitive to the amount of voids being produced at the fuel rod surface. The void generation rate depends on the specific surface conditions, the number of initially available unflooded nucleation sites and the subcooling history. The information provided in Reference-16 is not sufficient to demonstrate the applicability of (1) the proposed transient heat transfer model and (2) the cited experimental data to the conditions present during the RDA.

2) Void Production The void production rate depends on the number of initially available unflooded nucleation sites, the rate of activation of the flooded nucleation sites during the transient and the subcooling history. The description of the void production model provided in Reference-16 does not indicate how this dependence is included.

3) RAMONA-3B-SCP2 Model A more detailed description of the revised void production model and its implementation is required. For example, the Reference-16 description does not indicate how the differences in the wall and bulk coolant (1) temperatures and (2) available nucleation sites will be included in the void growth model.

4) Delay in Void Production The model described in Reference-16 includes a time delay in the production of the voids produced during the RDA transient. The basis for the method used to determine this time delay has not been provided.

While the description of the proposed RAMONA-3B-SCP2 void production model provided in Reference-16 is not considered sufficient to provide the basis for applying RAMONA-3B-SCP2 in RDA licensing calculations, this does not preclude a future NRC approval of these models if the necessary justification is provided by ABB Atom.

3.2 High Rod-Worth Rod Drop Analysis Topical Report RPA-89-053

3.2.1 High Rod-Worth Rod Drop Analysis Model

The results of the rod drop analysis have a substantial dependence on the initial reactor statepoint. The inserted rod-worths are larger for the low-power rod patterns, and the relative magnitudes of the Doppler and void feedbacks depend on the initial power level. In Response-B5 (Reference-16), ABB Atom has indicated that extensive sensitivity studies have been performed to evaluate the statepoint sensitivity of the RDA peak fuel enthalpy. For critical control rod patterns, the low-power/high-inlet-subcooling statepoints tend to result in the maximum peak fuel enthalpies. Based on these sensitivity calculations, ABB Atom has identified a conservative worst-case initial RDA reactor statepoint. ABB Atom has also indicated that, if in licensing analyses it is not apparent that this statepoint is limiting, additional calculations will be performed to determine the worst-case statepoint.

The RDA analysis of RPA-89-053 assumes a linear scram insertion which overestimates the magnitude of the initial scram reactivity insertion. In Response-B7 (Reference-16), ABB Atom has indicated that typical licensing calculations initiated with rod-worths based on startup rod patterns are not sensitive to the scram insertion. These transients are terminated by Doppler feedback prior to the scram insertion. However, in the case of very high rod-worths and large inlet subcooling, the Doppler feedback is not generally sufficient by itself to provide a prompt and complete reversal of the RDA transient. In this case, the RDA peak fuel enthalpy is sensitive to the scram reactivity and the assumption of a linear scram insertion. Consequently, in RDA licensing analyses involving very high rod-worths and large inlet subcoolings, the nonconservatism introduced by the assumption of a linear scram should be evaluated and, if necessary, accounted for in the determination of the peak fuel rod enthalpy.

The delayed neutron fraction, β, depends on tne fuel isotopics and, consequently, has a significant dependence on fuel type and burnup. RAMONA-3B-SCP2 calculates an importance-weighted core average β, as well as a nodal value of β. However, in Response-B4 (Reference-16), ABB Atom has indicated that in RDA licensing calculations a nodal value of β will be used, rather than an approximate core-average value.

3.2.2 High Rod-Worth Model Qualification

The methodology of RPA-89-053 will be applied to the calculation of rod-worths that are greater than typical BWR rod-worths. In support of this application, ABB Atom has compared PHOENIX rod-worth predictions to critical rod-worth measurements. The measurements were made for rod-worths that are substantially larger than assumed in the rod drop analysis. These comparisons indicate that the predicted and measured rod-worths agreed to within the accuracy of the rod-worth measurements. As further justification, ABB Atom has indicated that their experience using the PHOENIX/POLCA code system for core-follow calculations does not indicate any substantial reactivity bias or dependence of the power distribution uncertainties on fuel burnup.

3.2.3 High-Worth Rod Drop Accident Applications

The high-worth rod drop analysis provided in RPA-89-053 assumes a dropped rod-worth which is 15% larger than a typical BWR rod-worth. The topical report does not indicate how control rod-worths greater than the assumed worth will be evaluated. However, in Response-B1 (Reference-16), ABB Atom indicates that the RDA described in the report is only provided to show the impact of a high-worth rod on the RDA and is not intended as a bounding analysis. ABB Atom has indicated that licensing analyses will be performed with the methodology of CENPD-284, and will account for the core-specific control rod-worth.

The high-worth RDA evaluated in the topical report employed only six thermal-hydraulic channels to represent the core flow distribution. While the RDA evaluated was initiated by a center control rod drop and may be insensitive to this approximation, the error introduced in an off-center licensing analysis may be substantial. However, in Responses-B10 and C5 (Reference-16), ABB Atom has indicated that in RDA licensing calculations involving high-worth control rods, thermal-hydraulic channels will only be combined if the resulting RDA predictions can be shown to be unaffected or conservative.

The RDA power transient and fuel enthalpy increase are sensitive to the inserted dropped rod reactivity. The dropped rod reactivity is determined, in part, by the control rod pattern for the initial RDA statepoint. Control rod patterns which increase the local power at the dropped rod location increase the reactivity worth of the dropped rod. In Response-B6 and C2 (Reference-16), ABB Atom indicates that the selection of the highest worth rod will account for the worst-case single equipment malfunction, operator error, and the maximum number of bypassed rods allowed by the plant Technical Specifications and licensing basis.

The RDA analysis for the high-worth control rods includes calculations with and without moderator reactivity feedback. In cases involving large rod-worths, the RDA power transient is sensitive to the inclusion of moderator feedback. The increase in peak fuel enthalpy is generally terminated earlier and at a reduced value in the case with moderator feedback. In the topical report, a temporal extrapolation of the transient fuel enthalpy is made in the no-feedback case, assuming the time dependence of the fuel enthalpy is the same for the cases with and without feedback. This approximation can introduce a substantial uncertainty into the RDA peak fuel enthalpy. However, in Response-B9 (Reference-16), ABB Atom indicates that in RDA licensing calculations this approximation will not be used.

4.0 TECHNICAL POSITION

The ABB Atom control rod drop analysis Topical Reports RPA-89-112 and RPA-89-053 and supporting documentation provided in Reference-16 have been reviewed in detail. Based on this review, it is concluded that the ABB Atom control rod drop methodology and high rod-worth application are acceptable for performing reload licensing analyses of BWR/4-6 plants using the Banked Position Withdrawal Sequence, subject to the conditions stated in Section-3 of this evaluation and summarized in the following.

1) Application to Non-ABB Fuel

The accuracy of the PHOENIX/POLCA code system in applications involving non-ABB fuel must be demonstrated by comparisons to previous cycle measurements of the core reactivity and power distribution (Section-3.1.1).

2) STAV Fuel Performance Code

ABB Atom determines the fuel rod gap conductance using the STAV fuel performance code. STAV is presently being reviewed by the NRC and the approval of the ABB Atom RDA methodology is contingent on the approval of STAV (Section-3.1.3).

3) Effect of Moderator Voids

Because of the present uncertainty in the rate of void production during the initial power transient, RDA licensing analyses should be conservatively calculated without moderator voids. While this submittal does not provide a sufficient basis for applying RAMONA-3B-SCP2 in RDA licensing calculations, this does not preclude a future NRC approval of these models if the necessary justification is provided by ABB-CE (Section-3.1.3).

4) Selection of Thermal-Hydraulic Channels

In RDA licensing calculations, each fuel bundle should be represented by a unique thermal-hydraulic channel, or the predictions made by combining channels should be shown to be conservative or insensitive to this approximation (Section-3.1.3 and 3.2.3).

5) Linear Scram Assumption

In RDA licensing analyses involving very high rod-worths and large inlet subcoolings, the nonconservatism introduced by the assumption of a linear scram should be evaluated and, if necessary, accounted for in the determination of the peak fuel rod enthalpy (Section-3.2.1).

6) High-Worth Control Rod Reactivity

In RDA licensing analyses involving high-worth control rods, the core-specific control rod-worth must be determined (Section-3.2.3).

7) <u>Determination of Highest-Worth Rod</u>

In RDA licensing analyses, the selection of the highest-worth control rod must account for the worst-case single equipment malfunction and operator error allowed by the plant Technical Specifications and licensing basis (Section-3.2.3).

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CENPD-284-NP-A, RPA 89-112-NP-A, and RPA 89-053-NP-A REPORT

Part II

Body of CENPD-284-NP-A Report

Note that the responses to requests for additional information regarding this part of the report are included in Appendix C of Part II of this Report

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1 SUMMARY AND CONCLUSION

This topical report describes the ABB BWR Control Rod Drop Accident (CRDA) Methodology and provides qualification information demonstrating that the methodology is adequate for ensuring compliance to General Design Criterion (GDC) 28 and Standard Review Plan (NUREG-0800).

This report identifies specific design bases which, if satisfied, assure that all requirements specified in GDC 28 and NUREG-0800 applicable to the CRDA are satisfied.

The ABB methodology for performing CRDA analyses and the systematic cycle-specific strategy utilized by ABB are described in this report.

A complete cycle-specific analysis is fundamentally a two-step approach. [Proprietary Information Deleted]

The second step is simulation of the dynamic response to the identified worst dropped control rod(s) and the subsequent consequences to the fuel. This evaluation is performed with the three dimensional systems transient code RAMONA-3B, described, for example, in References 2 and 3. The candidates for the worst-case condition established in the first step are simulated in the RAMONA-3B core model for the dynamic evaluation. The RAMONA-3B methodology utilizes state-of-the-art phenomenological models including moderator feedback to describe the overall transient response of the plant and core in conjunction with the local thermal behavior of the fuel.

The ABB strategy for a cycle-specific evaluation includes systematic review of existing results and the use of bounding calculations to envelope worst case consequences of the CRDA for the subject cycle.

The qualification basis of the ABB CRDA methodology is presented in this report. It is shown that the PHOENIX/POLCA system of codes is qualified for steady-state control rod worth determinations by reference to the ABB Nuclear Design Methodology in Reference 1. Reference 1 has been reviewed and approved by the NRC. The methodology for the dynamic evaluation using RAMONA-3B is applied to the SPERT-IIIE power excursion tests to demonstrate the adequacy of the methodology for establishing the reactivity and power response resulting from a dropped control rod. Separate supporting data are also presented or cited which establish the adequacy of RAMONA-3B code for dynamic transient responses such as a CRDA, including the RAMONA-3B neutronic, thermal-hydraulic, and fuel-rod enthalpy models.

The ABB CRDA methodology can be used to analyze control rod drop accidents for both standard and high worth control rods. References 4 and 5 provide examples of the application of the ABB methodology for standard control rods and those with high reactivity worths, respectively. These examples demonstrate that the methodology described and justified in this report is practical and can be conveniently and accurately utilized for the CRDA evaluation on a cycle-specific or generic basis.

Based on the evaluation in this report, and the supporting information in References 4 and 5, it can be concluded that:

(1) The design bases identified are sufficient to assure that all requirements and guidelines identified in the GDCs and NUREG-0800 for the CRDA will be satisfied.

(2) The methodology and strategies described are acceptable for design and licensing purposes. Specifically, they are acceptable for identifying the limiting event and evaluating BWR plant response and subsequent consequences to the fuel systems resulting from a postulated CRDA relative to the design bases for design and licensing purposes.

(3) The methodology described in this report can be used to analyze control rod drop accidents for both standard and high worth control rods in BWR reactors.

2 DESCRIPTION OF ACCIDENT

The control rod drop accident assumes the decoupling of an inserted rod drive from the control blade. It is postulated that the drive mechanism is withdrawn while the control blade sticks in position and that the blade subsequently falls at its maximum speed to the position of the drive. Since it is assumed that the event can occur in any reactor operating state, consideration must be given to all the control rod configurations which can occur in normal operation as well as those which can occur as a result of equipment malfunction or operator error (e.g. the most severe single operator selection of an out of sequence control rod).

The accident is most severe when it is assumed to occur at low or zero power conditions when the control rod patterns required to establish criticality provide the highest values of incremental (dropped) single control rod worth. Furthermore, the presence of voids in the core at any significant power level will decrease the consequences of the accident through the negative moderator density reactivity (void) coefficient and the enhanced heat conductivity to the coolant relative to the cold case. Consequently, large subcooled conditions, such as a start-up from cold shut down, which do not result in significant boiling, usually provide the most severe initial states for the transient.

For a particular plant, consideration must be given to the hardware employed for rod sequence control and the technical specifications concerning inoperable rods in order to determine the limiting incremental rod worth.

For some Banked Position Withdrawal Sequence (BPWS) plants (Reference 6) the Rod Worth Minimizer (RWM) is used below a specified power (typically 5 to 20 %) to enforce the rod withdrawal sequence. To limit the worth of the rod which could be dropped in the Group Notch class of plants a group notch Rod Sequence Control System (RSCS) is installed to control the sequence of rod withdrawal. In GE-built BWR/6 plants a Rod Pattern Control System (RPCS) is used to enforce BPWS rules.

The sequence of the accident is as follows:

(a) At some time a fully inserted rod becomes decoupled from its drive and sticks in the fully inserted position.

(b) During the startup sequence, rod patterns are employed which are permitted by the constraints on rod movement imposed by the plant Technical Specification and hardware including the maximum allowable number of bypassed rods. At some time, under critical reactor conditions, a rod pattern exists for which the decoupled rod has the maximum incremental worth from fully inserted to the position of its drive. The rod is assumed to drop at this time.

(c) The reactor goes on a positive period, and the initial power burst is terminated by the fuel temperature reactivity feedback.

(d) The 120% APRM power signal scram occurs (no credit is taken for the Intermediate Range Monitor or set-down APRM scram).

(f) All withdrawn rods, except the decoupled rod, scram at the technical specification rate.

(g) A scram terminates the accident.

3 DESIGN BASES

The ABB design bases for the CRDA have been selected to be in compliance with the requirements in GDC 28 (10CFR 50, Appendix A, General Design Criteria) and the Standard Review Plan (NUREG-0800).

The criteria against which the consequences of the CRDA are evaluated are based on meeting the requirement of General Design Criterion 28 stating that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capacity to cool the core.

These criteria are given in NUREG-0800, the Standard Review Plan, as:

(1) Reactivity excursions should not result in a radially averaged fuel rod enthalpy greater than 280 calories/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 Code.

(3) The number of fuel rods predicted to reach assumed thresholds and associated parameters, such as the mass of fuel reaching melting conditions, will be input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

4 ABB BWR CONTROL ROD ACCIDENT ANALYSIS METHODOLOGY

4.1 Introduction

The control rod drop accident is analyzed for commercial Boiling Water Reactors as a design basis accident which is bounding for all postulated accidents involving additions of prompt reactivity. The method of analysis chosen must be capable of treating the effects of rapidly changing power distributions which are caused by the rapid control rod movement.

The purpose of this section is to describe the methodology used by ABB to determine the most limiting dropped control rod configuration and to evaluate the consequences of a CRDA in BWRs containing fuel or control rods of the ABB design as well as other vendors' designs. The methodology is illustrated with typical results including an assessment of the sensitivity to major analysis options and important parameters.

4.2 Overview

The consequences of the CRDA are addressed on a cycle-specific basis. The strategy for the cycle-specific evaluation is provided in Section 4.7. The computer codes used by ABB to evaluate the CRDA are summarized in Section 4.3, and the ABB methodology for a CRDA evaluation is discussed in Sections 4.4 through 4.6. As discussed in Section 4.7, options for the cycle-specific evaluation include a complete analysis using the methodology described in this report, demonstration that a previous analysis utilizing the methodology described in this report is bounding for the cycle of interest, or a partial evaluation used in conjunction with applicable previous results.

The consequences of the accident relative to the design bases are evaluated for the most limiting time in the cycle and the most limiting reactor conditions. Existing sensitivities and the use of bounding parameters which effect the fuel performance are utilized to the greatest extent possible to limit the number of reactor conditions and burnups for which specific calculations are required. Many reports are available in the literature (References 6 through 9) which contain discussion of the mechanics of the accident, and parametric studies of the consequences as function of control rod patterns, fuel type, and exposure. Other publications (References 10 through 14) have examined various aspects of the analytical models which can be applied to this event. The reports mentioned are generically applicable and cover a large number of input variables including different fuel types and core designs at different exposures and initial conditions. We have augmented these existing sensitivity studies with our own calculations utilizing RAMONA-3B. Some of these ABB sensitivity results have been previously submitted to the NRC in References 4 and 5.

These sensitivities have established that the accident is most severe when it is assumed to occur at low or zero power conditions when the control rod patterns required to establish criticality provide the highest values of incremental (dropped) single control rod worth. Furthermore, the presence of voids in the core at any significant power level will decrease the consequences of the accident through the negative moderator density reactivity (void) coefficient and the relatively low heat conductivity associated with subcooled conditions. Consequently, the evaluation of the accident usually can be limited to highly subcooled conditions and dropped control rod configurations providing relatively large integrated reactivity and high final nodal peaking. These sensitivities are illustrated in Section 4.5.2.2, and utilized in establishing the cycle-specific analysis strategy discussed in Section 4.7.

Based on the present and previous sensitivity evaluations, the following parameters have the greatest impact on peak fuel enthalpy in the fuel rods:

(1) The reactivity inserted as a function of distance the rod travels, or reactivity shape function. This parameter depends on the axial shape of the neutron flux which is absorbed by the control rod.

(2) The total reactivity worth of the dropped control rod.

(3) The local fuel rod power.

(4) The delayed neutron fractions of the various fuel types.

(5) The negative reactivity inserted as a function of scram control rod insertion distance.

(6) The Doppler reactivity feedback.

(7) The moderator temperature and subcooling.

(8) The initial power level (i.e. initial fuel temperature).

These parameters depend on such variables as the control rod pattern, the core hydraulic conditions, the core burnup and burnup distribution, and type of fuel in the core. Therefore, analysis of a cycle for the most limiting situation requires, in principal, a large matrix of core conditions and burnups. However, as noted above, the range of evaluation of the accident can usually be limited to a range from cold critical to hot standby. Furthermore, as described in Section 4.7, the cycle-specific strategy utilizes a systematic approach based on existing sensitivities to reduce the scope of a cycle-specific evaluation.

The ABB methodology for a complete analysis of the CRDA is fundamentally a two-step approach. The first step involves determination of possible candidates for the control rod which would cause the most severe consequences resulting from a CRDA. [Proprietary Information Deleted] The methodology for establishing candidates for the most limiting dropped control rod is contained in Section 4.4.1.

Having established the candidates for the most limiting dropped control rods within the cycle, the second step is analysis of the dynamic response to those dropped control rods and the subsequent consequences to the fuel. This evaluation is performed with the systems transient code RAMONA-3B described, for example, in References 2 and 3. A summary of the characteristics and capabilities of the RAMONA-3B code is also provided in Sections 4.3.1 and 4.3.2 for convenience.

The candidates which provide the most limiting reactivity insertions established in the first step are simulated in the RAMONA-3B core model for the dynamic evaluation. In the absence of data which would justify the use of a less conservative value, the control rod is assumed to drop at the maximum drop velocity of 0.948 m/sec (3.11 ft/sec) established in the Appendix to Reference 7. Other parameters which effect the severity of the accident, such as scram reactivity, Doppler feedback, delayed neutron fraction, initial fuel temperature, moderator temperature, and moderator subcooling are treated in a manner which insures that the most limiting case is bounded as discussed in Section 4.4.2. Bounding values of some of these variables are utilized to reduce the number of cases which must be evaluated.

Finally, the results are compared with the design bases to confirm that adequate margin is available for the CRDA.

4.3 Computer Codes Used for the Evaluation of the CRDA

The dropped control rod causes a large local increase in reactivity and a substantial change in the power distribution during the course of the accident. The method of analysis must represent this power shape change properly to account for the effect and to calculate the energy deposited in the fuel rods.

The computer codes utilized for the ABB CRDA include the RAMONA-3B code, which is a systems transient code for prediction of the dynamic behavior of a BWR (References 2 and 3), the POLCA code (Reference 1) which is used to provide the core history (burnup and void history distributions), and to determine the most limiting dropped control rod configurations, and the PHOENIX code (Reference 1) which provides the homogenized nuclear constants and local peaking factors to both RAMONA-3B and POLCA including kinetics parameter data for RAMONA-3B.

4.3.1 RAMONA-3B Code

RAMONA-3B is a systems transient code for prediction of the dynamic behavior of a BWR. It is specifically designed to simulate normal and abnormal operational plant transients, as well as accidents such as the control rod drop accident and ATWS transients. Because of its unique feature of combining full 3-D modeling of the reactor core and transient plant response, it is particularly suited for transients showing large local effects in the core.

This section presents a summary of modeling characteristics in RAMONA-3B for neutron kinetics, thermal conduction, and thermal-hydraulics. A detailed description of the code is given in Reference 2.

A 1-1/2 energy group, coarse mesh diffusion model in a three dimensional rectangular coordinate system is used to predict transient three-dimensional fission power distributions in the core. Six delayed neutron groups are accounted for. Decay heat from fission products is computed in RAMONA-3B from ANS Standard 5.1 (1979). All feedback mechanisms between neutron kinetics and thermal-hydraulics are modeled.

The neutron kinetics equations are solved using a box integration procedure to treat the space variables and an implicit time differencing scheme to treat the time variable. The core symmetry can be octant, quarter, half, or full-core and can model both rotational and mirror symmetry.

Thermal energy storage and conduction in the pellet, pellet-clad gap, and fuel cladding is computed using spatial discretization in the radial direction in a finite difference form. Axial conduction and the temperature dependencies of thermal conductivity in the cladding are ignored. The gap conductance and fuel conductivity are defined specifically for each fuel type as a function of average pellet temperature but independent of burn-up. Therefore, different polynomial expressions are utilized as required to capture the impact of burnup. Implicit iterative time integration is used to solve the conduction equations.

The RAMONA-3B models allow two phase flow with unequal phase velocities described by a slip correlation and treat subcooled or superheated liquid phases. Transient boron concentrations can also be treated. The Bankoff-Malnes slip correlation is used. Four equations treating vapor mass, mixture mass, and momentum, and energy conservation describe the coolant dynamics in the vessel.

Two equations of vapor mass and momentum conservation describe the acoustic effects from valve closures in the (adiabatic) steam lines. One boron mass conservation equation is used to predict the transport of boron.

A single pressure is used in the entire system to compute all phasic properties. This technique eliminates the local effects from phasic material properties, neglects acoustic effects in the vessel, and contributes significantly to the computing economy in RAMONA-3B. One closed-contour momentum equation is used to predict the individual axial velocities in each of the parallel core flow channels in the problem. This method increases significantly the computing speed. The partial differential equation for the mass conservation of each phase is integrated by a simple quadrature in space. This method also significantly increases computing speed without loss in accuracy. These three advanced modeling features provide RAMONA-3B with the capability to compute three-dimensional neutron kinetics and thermal hydraulics for multichannel core geometries in the context of a systems code and produce sufficiently accurate results at acceptable costs.

RAMONA-3B accounts for non-equilibrium vapor generation, unequal phase velocities, wall shear and heat transfer for single-phase and two-phase flow conditions.

RAMONA-3B has individual component modeling to accommodate BWR systems of U.S. design. All recirculation loops and all steam lines are represented in RAMONA-3B by a single recirculation loop with a single jet pump and a single steam line, respectively.

The core can be spatially described in the same detail as that in the three-dimensional nodal simulator. Each assembly can represent a separate hydraulic channel and radial node with the same axial nodalization (e.g. 25 nodes) as the three-dimensional nodal simulator.

Due to the wide spread use and acceptance of RAMONA-3B for reactor analysis, further definition of the version utilized by ABB for the BWR

CRDA is probably appropriate. The code utilized by ABB is an extension of the version used, documented, and released by Brookhaven National Laboratory (BNL) in 1983 in Reference 2. Documentation of basic methods, code features and limitations are found in Reference 2. Reference 2 also describes some results from applications and provides a complete documentation of that code version.

The present code version, referred to as the "Scandpower version of RAMONA-3B", includes the features in BNL Level 10 and can be considered to be upgraded to BNL version "Level 10". Unless otherwise indicated, the term "RAMONA-3B" in the discussion below refers to the Scandpower version of RAMONA-3B.

The features of the Scandpower version of RAMONA-3B are summarized in the User's Manual (Reference 3). The most important extensions relative to Reference 2 can be summarized as follows:

(1) The nuclear cross-section data representation is made compatible with Scandpower's static 3-D core analysis methods (FMS) and the ABB CORE MASTER system which includes POLCA.

(2) The option to input a non-equilibrium xenon distribution which provides the capability to initiate the transient from a non-equilibrium xenon state.

(3) Effective delayed neutron fractions are treated as nodal variables as a function of fuel design and burnup.

(4) The nodal coupling method dealing with the thermal flux diffusion has been upgraded to that of the static three-dimensional nodal simulator, PRESTO (Reference 15).

(5) The modeling of reverse flow conditions has been improved.

(6) The time integration of the hydraulics has been improved to optionally allow for higher order explicit methods as well as implicit integration of some of the equations.

4.3.2 Major Computer Codes Supporting RAMONA-3B in the ABB CRDA

The CRDA is studied with RAMONA-3B using an extended code version (Scandpower version of RAMONA-3B) that has been linked to PHOENIX and POLCA. PHOENIX and POLCA are the standard ABB codes for static core design. Figures 4.3.1 and 4.3.2 show the relationship and interaction between computer codes used by ABB in the CRDA analysis.

As documented in Reference 1, the PHOENIX and POLCA codes have been reviewed and accepted for ABB design and licensing by the NRC. These are the standard ABB BWR codes used for neutronic design and licensing calculations.

The PHOENIX code is a two-dimensional, multi-group transport theory code which is used to calculate the lattice physics constants for fuel assemblies having varying complexities. The POLCA code is a modified one-group nodal code which is used for the three-dimensional simulation of the nuclear and thermal-hydraulic conditions typical of boiling water reactor cores. Auxiliary codes used with PHOENIX and POLCA include FOBUS, PHOEBE, and PHIPO.

The PHOEBE code is used to prepare the nuclear data library for PHOENIX. The FOBUS code generates the self-shielded multigroup microscopic absorption cross-sections for the gadolinium burnable absorber isotopes for use in PHOENIX. The PHIPO code serves as the linking code between PHOENIX and POLCA.

4.3.3 PHOENIX Code

PHOENIX is a two-dimensional, multi-group transport theory code which is used for the calculation of eigenvalue, spatial flux, reaction rate distributions, and depletion of rod cells for BWR fuel assemblies. The code is described in detail in Reference 1 which also contains documentation of its NRC review and acceptance. A brief description of code is provided here for convenience.

The code can simulate BWR cruciform control blades containing cylindrical absorber elements, water gaps, burnable absorber rods, burnable absorbers that are integral with the fuel, water rods, and the presence of objects in the water gaps such as neutron detectors.

PHOENIX is supported by the burnable absorber program FOBUS and by the PHOENIX library service program PHOEBE. PHOENIX is the standard ABB depletion program for BWR fuel assembly and rod cell calculations. Each of the fuel rods is individually treated, and there is no limitation on the number of different rod types that can be represented in the PHOENIX problem. The code can accommodate a variety of geometric configurations including fuel rods with different radii, plutonium fuel, burnable absorber rods, and water holes.

In the water gaps, any number of objects may be specified, such as detectors, control blades, and control blade tips. These are either treated homogeneously or, in the case of a control blade with absorbing rods, heterogeneously. In addition to rod cell and fuel assembly calculations, quadruple assembly problems can be run, consisting of four assemblies in 2x2 array. This option permits the detailed calculation of rodwise power distribution, reaction rates, reactivities, and detector constants for a 2x2 array containing different fuel assembly types. The principal output of PHOENIX is fuel assembly reactivity, isotopic concentrations as a function of burnup, rod-by-rod power distributions, two group homogenized controlled and uncontrolled cross-sections, tables of detector signals, local peaking factor, factors related to the rodwise power distribution used in the critical power ratio correlation, xenon, samarium and boron microscopic cross-sections, and kinetics parameters such as delayed neutron fractions and inverse velocities.

4.3.4 POLCA Code

POLCA is a modified one-group nodal code designed to provide realistic three-dimensional simulation of the nuclear, thermal and hydraulic conditions in boiling water reactors. The code is described in detail in Reference 1 which also contains documentation of its NRC review and acceptance. A brief description of code is provided here for convenience.

The three-dimensional neutronics of the reactor core are described by a modified one-group nodal model. The nodal equations are the result of a specially adapted coarse-mesh diffusion approximation. A set of coupling coefficients are evaluated from two-group data which are stored as a number of three-dimensional tables. The table entries are burn-up, void, and void history. The void history affects the isotopic composition per node. The neutronics equations are solved by Gauss-Seidel inner iterations with a Chebyshev iteration of the fission source. A thermal coupling correction, based on the asymptotic thermal fluxes of the direct neighbors, is made by modifying the removal cross-sections prior to the iteration process.

The hydraulic calculations are performed by a special version of the CONDOR thermal-hydraulic design code. A description of the CONDOR code, its qualification, and the NRC Safety Evaluation Report are provided in Reference 16.

In addition to the computation of the linear heat generation rate and CPR edits, POLCA also edits bundle wise, core average axial, and three-dimensional node wise distributions of power, burn-up, void, xenon, and iodine concentrations, inlet flow distribution, local power range monitor (LPRM), and traversing incore probe (TIP) predicted signals. POLCA can be run in quarter-, half-, or full-core configurations.

Typical modeling of the fuel assembly utilizes one radial node per assembly and 25 axial nodes.

4.4 Determination of Candidates for the Limiting Control Rod

4.4.1 Analysis Methodology

[Proprietary Information Deleted]

Since it must be assumed that the reactor can be shutdown and restarted at any time during the cycle, the possibility of a control rod drop in this operating range must be considered throughout the cycle. The parameters to which the severity of the accident is sensitive can change throughout the cycle. For example, the fuel temperature Doppler coefficient typically tends to get somewhat more negative with burnup, which would tend to make the accident less severe with increasing burnup with all other parameters to which the CRDA is sensitive held constant. However, the delayed neutron fraction typically tends to decrease with increasing burnup, which tends to make the accident more severe. The axial fission source shape also changes throughout the cycle which can affect the reactivity shape function.

[Proprietary Information Deleted]

The POLCA 3-D core simulator is used to simulate the control rod withdrawals observing the restrictions imposed by the plant Technical Specifications. The two most common rod withdrawal sequences specified for U.S. Plants are the Banked Position Withdrawal Sequence (BPWS) and the Group Notch Sequence. These control rod programs are used to withdraw the control rods in a manner which will mitigate the severity of the CRDA.

[Proprietary Information Deleted]

The POLCA calculations in the start-up and power ranges are performed using the methods described in Sections 4.3.3 and 4.3.4 as well as in Reference 1. Cross section data required for the POLCA calculations at the burnups considered are calculated with the PHOENIX code.

4.4.2 Example of a Scoping Calculation

[Proprietary Information Deleted]

The illustrative calculations were performed for a 764-assembly BWR/5 core. The rated core thermal power and flow rate are 3323 MW_{th} and 108.5 Mlb/hour, respectively. An equilibrium, reload core of ABB SVEA-96 assemblies designed for an 18-month cycle application was utilized for the these illustrations.

[Proprietary Information Deleted]

4.5 Dynamic Analysis To Determine Energy Deposition in Fuel

4.5.1 Analysis Methodology

The dynamic analysis is performed with the 3-D plant transient analysis code, RAMONA-3B, for each of the potentially limiting cases identified in the POLCA scoping evaluation. The potentially limiting cases are those for which the failure threshold of 170 calories/gm could credibly be achieved during a CRDA. Typically, each fuel bundle is individually modeled and divided into 25 axial nodes. Full-, half-, quarter-, or eighth-core calculations can be performed as required. Full-core calculations are generally required to simulate the dropping of a single control rod and account for asymmetric effects. Appropriate files from POLCA provide the nodal burnups and void histories for the specific case considered as shown in Figure 4.3.1.

Figure 4.3.1 also shows that the two-group cross sections and local peaking factors from PHOENIX are put into the polynomial forms required by RAMONA-3B. The cross section dependence on burnup and void history is converted to the standard RAMONA-3B format. The void and fuel temperature dependence at discrete burnup and void history state points are also translated into the standard RAMONA-3B representation.

The kinetics parameters required by RAMONA-3B (delayed neutron fractions and inverse velocities) are obtained from PHOENIX and assigned on a nodal basis.

[Proprietary Information Deleted]

In the absence of data which would justify the use of a less conservative value, the control rod is assumed to drop at the maximum drop velocity of 0.948 m/sec (3.11 ft/second) established in Reference 7. [Proprietary Information Deleted]

The fuel pellet enthalpies are calculated in each node using the nodal powers from the dynamic RAMONA-3B calculation and the local peaking factor assigned to that node. The local peaking factors are assumed to be constant throughout the transient. [Proprietary

Information Deleted] The pellet and clad are nodalized into concentric rings, and heat conduction in the axial direction is neglected. The constants required for the fuel thermal conductivity and gap conductance are obtained from a licensed fuel performance code. ABB intends to utilize the ABB fuel rod thermal-mechanical performance code, STAV (Reference 22). Constants required for the RAMONA-3B fuel heat capacity treatment are obtained from the most recent MATPRO compilation.

The RAMONA-3B calculations explicitly treat the feedback mechanisms which are most important for a postulated CRDA. [Proprietary Information Deleted]

Moving control rods are represented in RAMONA-3B as fast and thermal group controlled cross-sections added to the cross sections for unrodded fuel. Therefore, the dropped rod is modeled as a boundary which moves at a constant speed of, for example, 0.948 m/second (3.11 feet/second). A weighted average of controlled and uncontrolled cross sections is used for axial locations interior to an axial node. Similarly, the partially or fully inserted scram rods are represented as controlled cross sections which move into the core at constant speed after a set delay when activated by a scram signal. Control rods which are fully inserted and stationary throughout the transient are modeled as separate fuel types. They are assigned a separate set of cross sections for controlled fuel. The ABB methodology assumes that all control rods except the dropped rod are fully inserted into the core as the result of a scram.

The ABB methodology utilizes state-of-the-art methods. Unnecessary conservatisms in the methodology have been avoided to allow the accurate prediction of margin to design bases. Conservatisms are included by assuming bounding input parameters. Bounding calculations are performed to reduce the number of analyses required.

4.5.2 Example of a Dynamic Calculation to Determine the Energy Deposition in the Fuel

This section provides illustrations of the ABB methodology for performing the dynamic CRDA evaluation. The illustrative results are shown for the same 764-assembly BWR/5 equilibrium SVEA-96 core for which the steady-state calculational results were presented in Section 4.4.2. The bundle averaged cross-sections, peaking factors, and kinetic parameters were obtained from PHOENIX, and nodal burnups and void-histories were obtained from POLCA following the method described in Section 4.5.1. Full-core RAMONA-3B calculations are generally required for cycle-specific applications to simulate the dropping of a single control rod and account for asymmetric effects.

Octant core (one-eighth core with reflecting boundary conditions) calculations are, however, useful for performing sensitivity studies since less computer running time is required, and a larger range of conditions can be evaluated. Therefore, calculations were performed for both full-core and octant core configurations.

Results are presented for a conservative base case in Section 4.5.2.1, and sensitivities to the parameters to which the peak fuel enthalpy is most sensitive following a CRDA are discussed in Section 4.5.2.2.

References 4 and 5 provided similar results utilizing the ABB methodology described in this document applied to a European D-lattice BWR/4 containing 532 assemblies. Results from the calculations in References 4 and 5 are compared to those from the present calculations as appropriate.

4.5.2.1 Base Case

A set of "Base Case" conditions were selected to provide an example of a dynamic CRDA calculation which would be expected to be limiting for the 764-assembly BWR/5 SVEA-96 equilibrium core provided as an illustration and to provide a basis for the sensitivities in Section 4.5.2.2.

The Base Case conditions are listed in Table 4.5.1. The BPWS configuration selected for the Base Case from the POLCA survey calculations described in Section 4.4.2 provides a total reactivity worth of [Proprietary Information Deleted] pcm. It is expected that this total reactivity worth would be limiting or bounding for a realistic case.

[Proprietary Information Deleted]

Insight into the transient is provided by examining the core power response, reactivity contributions, and power shapes as a function of time for the Base Case calculation summarized in Table 4.5.1. Figure 4.5.1 shows the core power as a function of time. [Proprietary Information Deleted]

Further insight into this transient power behavior is provided by the estimated contributions to the reactivity inserted into the core as shown in Figure 4.5.2. [Proprietary Information Deleted]

The top-skewed axial power shape leads to the reactivity shape function as the control rod is withdrawn shown in Figure 4.5.4.

The top-skewed axial power shape in this cold condition is typical of reload cores. A reload core was selected, rather than an initial core, to provide a relatively top-skewed power distribution which would tend to provide a more limiting shape reactivity function and reduce the effectiveness of the scram.

Figure 4.5.5 shows the peak instantaneous fuel enthalpy and integrated fuel enthalpy for the Base Case. As shown in Figure 4.5.5, [Proprietary Information Deleted]

Figure 4.5.6 shows the peak fuel temperature for the Base Case. The peak fuel temperature behaves qualitatively in the same manner as the peak fuel enthalpy.

[Proprietary Information Deleted]

4.5.2.2 Sensitivities

Perturbation calculations on the Base Case discussed in Section 4.5.2.1 were performed to evaluate the effects of parameters to which the peak fuel enthalpy is most sensitive during the accident. These sensitivity calculations not only provide further understanding of the various physical phenomena contributing to the response of the core to the dropped rod, but also allow a comparison of the important sensitivities predicted by RAMONA-3B with previous work.

Sensitivities of the peak fuel enthalpy reached during the accident to the following parameters are discussed in this section:

[Proprietary Information Deleted]

Unless otherwise specified, the Base Case parameters listed in Table 4.5.1 were utilized in the sensitivity calculations.

[Proprietary Information Deleted]

<u>Summary</u>

Conclusions from the Base Case and Sensitivity evaluations can be summarized as follows:

(1) The RAMONA-3B results presented in Section 4.5.2 are consistent with conclusions and sensitivities provided in previous work. The numerical results are in good agreement with the results presented in References 4 and 5 for a 532-assembly European reactor.

[Proprietary Information Deleted]

4.6 Evaluation of Peak System Pressure During the Transient

An evaluation was performed to ensure compliance with the reactor pressure vessel design bases during a control rod drop accident if the fuel enthalpy limit of 280 calories/gm is satisfied. [Proprietary Information Deleted]

4.7 Strategy for Cycle-Specific Evaluations

Sections 4.4 and 4.5 described the ABB methodology for performing CRDA analyses and provided an application of those methods to a 764-assembly BWR/5 reactor. This section describes the type of strategy which ABB intends to use for applying these methods to cycle-specific licensing evaluations of the CRDA. The sensitivities and results discussed in Section 4, as well as the results of previous work, were utilized to establish this strategy.

[Proprietary Information Deleted]

4.8 Comparison of Analysis Results with Evaluation Criteria

A reload design is acceptable only if it conforms to the design criteria in Section 3. As discussed in Section 4.6, satisfaction of the 280 calories/gram limit will assure that the vessel "Service Class C" pressure ASME limit will be satisfied. Therefore, a cycle-specific or plant-specific evaluation against the ASME pressure limit criterion is not required.

Peak fuel enthalpies are confirmed to be less than 280 cal/gm.

If the peak enthalpy exceeds 170 calories/gram, the number of rods exceeding 170 calories/gram is calculated from the pin power distributions for those bundles whose peak enthalpy exceeds 170 calories/gram at any axial level. The number of rods exceeding 170 calories/gram must be confirmed to be less than or equal to the number of failed rods demonstrated to be acceptable in the FSAR radiological evaluation.

5 QUALIFICATION OF ABB CRDA ANALYSIS METHODOLOGY

5.1 Introduction and Summary

This section contains information to verify that the ABB methodology described in Section 4 for evaluation of the CRDA is sufficiently accurate and conservative for licensing applications. The verification is provided by systematically addressing the significant components of the methodology which affect the predicted peak fuel enthalpy which is compared to the design bases. Specifically, the following areas are addressed:

(1) The capability of the supporting PHOENIX/POLCA system of codes to provide adequate local pin power distributions, cross sections, and burnup and void histories for RAMONA-3B is discussed in Section 5.2. [Proprietary Information Deleted]

(2) The capability of the RAMONA-3B code to predict physical phenomena important for the determination of peak fuel enthalpies is addressed in Section 5.3. Specifically, the capability of the RAMONA-3B code to predict the time variation of core power, Doppler feedback, moderator density feedback, heat transfer from the pellet to the coolant, and fuel pellet enthalpy are discussed.

(3) The capability of RAMONA-3B to simulate integral tests of a CRDA. Specifically, simulations of six of the SPERT-IIIE power excursion tests are provided in Section 5.4. To our knowledge these SPERT tests provide the best data for directly testing the RAMONA-3B capability to describe a CRDA.

The results in Sections 5.2 through 5.4 support the following conclusions:

(1) The qualification of the PHOENIX/POLCA system of codes in Reference 1 is sufficient to support their application in the ABB CRDA methodology described in Section 4. Specifically, the local pin power distributions, cross sections, and burnup and void histories provided for RAMONA-3B are calculated with sufficient accuracy to support demonstration by RAMONA-3B that the CRDA design bases are satisfied. Furthermore, power distributions and void distributions are predicted by POLCA with sufficient accuracy to provide an adequate reference point for the corresponding power and void distributions predicted by RAMONA-3B just prior to the control rod drop. [Proprietary Information Deleted]

It is concluded that the available benchmark data base for the PHOENIX/POLCA code system fully qualifies it for the manner in which it is applied in the ABB CRDA methodology described in Section 4.

(2) The RAMONA-3B nuclear, kinetic, thermal-hydraulic, and fuel rod performance models predict the time variation of core power, Doppler feedback, moderator density feedback, heat transfer from the pellet to the coolant, and fuel pellet enthalpy with sufficient accuracy to provide reliable predictions of peak fuel enthalpy during the CRDA to confidently demonstrate that the CRDA design bases are satisfied using the ABB methodology described in Section 4.

(3) Comparison of RAMONA-3B simulation predictions with SPERT-IIIE power excursion test results shows that the Scandpower version of RAMONA-3B simulations using the ABB CRDA methodology show good agreement with the tests for which the nominal initial conditions quoted appear to reflect the actual situation. The comparisons demonstrate that for a peak power consistent with the experimental data, the ABB methodology using RAMONA-3B predicts resulting values of inserted reactivity, power shape, integrated energy, and time-to-peak power which agree with the experimental values to within the experimental uncertainties. Therefore, it is concluded that the ABB methodology predicts the results of the SPERT-III E-Core tests for which the comparisons were made to within the uncertainties in the tests and the uncertainties associated with the information available regarding those tests.

(4) In summary, the ABB methodology described in Section 4 for evaluating the CRDA using RAMONA-3B simulations can predict calculated peak pellet enthalpies during a postulated CRDA which are sufficiently accurate to demonstrate that the design bases provided in Section 3 are satisfied.

5.2 PHOENIX and POLCA Qualification

As discussed in Section 4, the PHOENIX code provides cross section data to POLCA and RAMONA-3B as well as local (pin) power distributions and kinetics parameters, such as delayed neutron fractions and inverse velocities, to RAMONA-3B for the CRDA calculations. POLCA provides burnup and void history distributions to RAMONA-3B and is used to identify candidates for the RAMONA-3B control rod drop analyses primarily based on calculated total control rod reactivity worths. A secondary selection criterion for candidates for the most limiting dropped control rod configuration is the peak nodal power predicted by POLCA with the dropped control rod in the withdrawn position.

Confirmation of the capability of PHOENIX and POLCA to calculate these quantities with sufficient accuracy to support demonstration by RAMONA-3B that the CRDA design bases are satisfied is provided in Reference 1. Reference 1 contains detailed qualification bases for the use of the PHOENIX/POLCA code system for steady-state nuclear design and analyses of BWR cores by ABB. The various components of Reference 1 were submitted to the NRC between 1982 and 1987 and accepted by the NRC for BWR reload design and analysis applications in 1988. The specific information in Reference 1 which confirms that the data provided by PHOENIX and POLCA for the ABB CRDA analyses are sufficiently accurate are summarized in this section.

PHOENIX

The benchmarking in Reference 1 included the following comparisons between calculated PHOENIX predictions and measured data or results from higher order methods:

(1) Neutron multiplication factors predicted by PHOENIX for room temperature, uniform lattice critical configurations were compared with experimental data. Comparisons were made with forty cold, clean, uniform UO_2, light water moderated critical configurations. The lattices spanned a wide range of Uranium-to-water ratios, rod dimensions, rod pitch, and U-235 enrichments bounding the conditions encountered in typical BWR fuel bundles. The mean $k_{effective}$ value and variation relative to the mean for the forty cases inferred from the PHOENIX calculations and the measured bucklings demonstrated the PHOENIX capability of accurately describing the reactivity of the various configurations.

(2) Calculated fuel rod power distributions were compared to gamma scan measurements performed at EOC2 of Quad Cities 1. Burnup and void history data were obtained for each bundle elevation from a POLCA simulation of Cycle 2 of Quad Cities 1. Agreement between the PHOENIX calculations and measurements were quite good. The pin power standard deviations for the UO_2 assemblies (The comparisons included assemblies with PuO_2 fuel rods.) were generally within the measurement uncertainties in the gamma-scan data (± 3 %).

(3) Fuel rod power distributions predicted with PHOENIX were compared with results from the KENO-IV Monte Carlo

program for P8x8R and the SVEA-64 fuel assembly design developed for the U.S. market. This assembly is referred to as "QUAD+". The comparison involved unirradiated assemblies at several void fractions. The overall agreement in fuel rod powers is comparable to the statistical uncertainty in the Monte Carlo results (2σ statistical uncertainty of $\pm$ 4-5 %).

(4) Comparisons of PHOENIX predictions of uranium and plutonium isotopic concentrations with measurements from Cycle 5 of Yankee Rowe demonstrated that PHOENIX reliably predicts the relative isotopic concentrations of important fissionable isotopes.

(5) The results of PHOENIX comparisons with small core critical experiments performed at the KRITZ facility and to gadolinia rod depletion data from Oskarshamn-1 showed good agreement for fuel rod fission rates, $k_{effective}$, Gd_2O_3 rod depletion, and Gd^{155} and Gd^{157} concentrations as a function of fluence.

The good agreement between fuel rod powers predicted with PHOENIX and the values measured at Quad Cities and in the Kritz facility, as well as those from KENO-IV calculations, confirm that the fuel rod power distributions predicted by PHOENIX have a relative uncertainty [Proprietary Information Deleted] This accuracy is considered to be sufficient for predicting relative pin powers in the CRDA methodology described in Section 4. The good agreement between neutron multiplication factors based on PHOENIX calculations and critical facility measurements, as well as the comparisons with the Yankee Rowe isotopic data, demonstrate the capability of PHOENIX to accurately predict neutron balance and reactivity as a function of burnup. This provides a direct indication that the delayed neutron fractions and inverse velocities provided to RAMONA-3B are reliable and a good indirect indication that the cross section data provided to POLCA and RAMONA-3B for the CRDA analyses are reliable.

<u>POLCA</u>

The qualification of POLCA in Reference 1 is based on simulations of the first three cycles of a typical U.S. BWR/4 and of the first two cycles of a U.S. BWR/3.

Hot and cold effective neutron multiplication factors ($k_{effective}$) were calculated by POLCA for critical control rod patterns. The accuracy of power distributions predicted by POLCA was confirmed by comparison of calculated results with measured values determined by gamma-scanning bundles as well as with TIP detector readings. Comparisons were also made with the plant process computer predictions. The results of these comparisons can be summarized as follows:

(1) The $k_{effective}$ values predicted by POLCA for the critical BWR/4 state points involved 95 hot reactivity depletion steps and 13 cold critical state points covering three cycles. The cold calculations simulated in-sequence cold criticals as well as asymmetrically withdrawn control rods to simulate stuck rod configurations. The reactivity of both the hot and cold critical configurations was reasonably well predicted by POLCA with no observable biases as a function of exposure. The cold critical $k_{effective}$ values for cases with asymmetric rods compared well with those for symmetric rod patterns.

The hot $k_{effective}$ values predicted by POLCA for the critical BWR/3 state points involved 41 burnup intervals over two cycles.

In general, the standard deviation of hot and cold calculated $k_{effective}$ values relative to the mean predicted by POLCA were [Proprietary Information Deleted]

(2) POLCA predictions were compared with the results of gamma scan data from 73 of the BWR/3 bundles scanned at 12 axial elevations and an additional 16 bundles scanned at 24 elevations.

[Proprietary Information Deleted]

Based on all of these comparisons of calculated powers with measurements, including the PHOENIX pin power measurements, the following uncertainties in calculated powers were established in Reference 2:

CODE	UNCERTAINTY	VALUE
[Proprietary Information Deleted]		
[Proprietary Information Deleted]		
[Proprietary Information Deleted]		

In addition to the results in Reference 1, extensive benchmarking of the PHOENIX/POLCA code system has been performed based on comparisons with ABB Nordic BWR data as well as data from KWU and GE BWRs in Continental Europe. The core follow experience on these plants with the PHOENIX/POLCA system of codes represents more than 70 years of full power reactor operation. The agreement between POLCA predictions and plant measurements for this data base is very similar to that discussed above for the U.S. plants. Specifically, calculated $k_{effective}$ biases as a function of burnup are relatively small, the spread in $k_{effective}$ values calculated for comparable critical conditions is similar to that determined for the U.S. plants, and the magnitude of the uncertainties in nodal and local powers determined from gamma-scan and comparisons with TIP data are similar to those quoted above. It should be noted that the combined U.S. and European data base includes a wide variety of 7x7, 8x8, 9x9, and 10x10 bundle designs involving open lattice and water cross configurations. No significant code bias has been observed for different bundle designs.

This extensive qualification data base for POLCA provides sufficient confirmation that the accuracy of the quantities provided by the code to the CRDA analysis is sufficient to support demonstration by RAMONA-3B that the CRDA design bases are satisfied. Specifically, the lack of substantial bias and relatively small spread in $k_{effective}$ values calculated as a function of burnup, as well as the absence of a burnup dependence on power uncertainties and their relatively small magnitude, indicates that the nodal cross sections and burnup and void history distributions provided to the RAMONA-3B calculations are sufficiently accurate. [Proprietary Information Deleted] Furthermore, the state-of-the-art accuracies in calculated nodal powers confirm that the void and power distributions calculated by POLCA represent a reliable benchmark to which initial steady-state RAMONA-3B predictions can be compared.

Therefore, the overall conclusion is that the available benchmark data base for the PHOENIX/POLCA code system fully qualifies it for the manner in which it is applied in the ABB CRDA methodology described in Section 4.

5.3 RAMONA-3B Qualification

RAMONA-3B originated from a joint development project by the Nuclear Research Institutes of the Scandinavian countries in the early 1970's. Subsequent development of the code by Scandpower International Consultants and Brookhaven National Laboratory has continued to the present. A substantial part of this development work has been supported by the NRC, and the code is utilized by the NRC for reference 3-D BWR systems transient analyses.

The code is specifically designed to simulate normal and abnormal operational plant transients as well as accidents such as the CRDA and ATWS events. Over the years it has been successfully used to simulate a wide variety of BWR transients and accidents. For example, it has been successfully utilized to study Anticipated Transients Without Scram (ATWS) (e.g. Reference 18), the Overpressurization Transient for various applications, stability analyses for various plants including the Ringhals and LaSalle, the Turbine Trip Tests at Mühleberg and Peach Bottom 2 (e.g. References 2 and 19), and the scram tests at Gundremmingen A (Reference 20). Since the code has the capability of treating each assembly in the core, it is particularly well suited for transients characterized by large local effects such as the CRDA.

The following sections contain qualification information which confirms that RAMONA-3B, as applied in the ABB methodology described in Section 4, predicts peak fuel enthalpy during a postulated CRDA with sufficient reliability to confidently demonstrate that the CRDA design bases are satisfied using the ABB methodology described in Section 4. The qualification is addressed in terms of the capability of RAMONA-3B to predict physical effects to which the CRDA is sensitive as well as direct applications of RAMONA-3B to the CRDA. The most important application is simulation of SPERT III E-Core experiments in Section 5.3.2.

5.3.1 Prediction of Physical Effects to which the CRDA is Sensitive

Comparison of calculated results with experimental data is the most convincing method of validating a computer code such as RAMONA-3B. Unfortunately, experimental data for plant transients resulting from the severe reactivity insertion which would be expected for a CRDA are very limited. The most applicable test data available to ABB is from the SPERT-IIIE test series, and RAMONA-3B simulations of six of these tests are compared to the measured data in Section 5.3.2.

Since test data simulating conditions expected in a CRDA are limited, comparison of RAMONA-3B predictions with available data is augmented by separate evaluations of the capability of applicable core modules to predict physical effects to which the CRDA is most sensitive. Specifically, the capability of RAMONA-3B to predict the following effects to which the CRDA is most sensitive is addressed in this section:

(1) Control Rod Worth and Power Distributions,

(2) Hydraulic Conditions,

(3) Doppler Reactivity, and

(4) Thermal Energy in the Pellet and Energy Conduction to the Coolant.

Control Rod Worth and Power Distributions

The capability of a three-dimensional nodal simulator code to predict control rod worths is dependent upon its capability to predict power shapes. Therefore, the two effects can not be treated independently and are discussed together in this section.

[Proprietary Information Deleted]

The very stable hot and cold values of $k_{effective}$ predicted by PRESTO for three reactors over a wide range of burnups and control rod densities and configurations reported in Reference 21 demonstrate the capability of the methodology to predict reliable control rod worths.

Reference 21 also contains extensive comparisons of PRESTO predictions with TIP and gamma scan measurements. Based on their evaluation of the all of the data, the following estimated standard deviations in powers predicted by PRESTO calculations with measurement uncertainties removed from the data is provided in Reference 21:

Quantity	Uncertainty in Power One Standard Deviation Measurement Uncertainty Removed
#	#
#	#
#	#

\# Proprietary Information Deleted

Therefore, Reference 21 demonstrates the state-of-the-art capability of the PRESTO code to provide reliable control rod worths and power shapes under hot and cold conditions. Since the hydraulic and neutronic models in steady-state are equivalent to those in the Scandpower version of RAMONA-3B used in the ABB methodology, the benchmarking in Reference 21 demonstrates that control rod worths and power shapes will be reliably predicted in RAMONA as well. It should also be noted that the benchmarking results are very similar to those reported for POLCA in Section 5.2. This is not surprising since the fundamental neutronics and hydraulic models are very similar for the two codes, and the void model is based on the same FRIGG Loop data base.

Therefore, the steady-state benchmarking data base provides a very convincing indication that control rod worths and power distributions are predicted by the Scandpower version of RAMONA-3B with sufficient accuracy to provide sufficiently reliable predictions of peak fuel enthalpy during the CRDA to confidently demonstrate that the CRDA design bases are satisfied using the ABB methodology described in Section 4.

Hydraulic Conditions

As in the case of the neutronics models, the time-dependent hydraulic models in the Scandpower version of RAMONA-3B are equivalent to those in the PRESTO three dimensional core simulator under steady-state conditions. As discussed below, the capability of PRESTO to predict steady-state hydraulic conditions provides an indication of the capability of the Scandpower version of RAMONA-3B to reliably predict hydraulic conditions during a transient.

Detailed descriptions and qualification of the PRESTO hydraulic models was submitted to the NRC in References 15 and 21. It is demonstrated in these documents that the hydraulic modeling of the BWR two-phase system under steady-state conditions in PRESTO is a state-of-the-art representation. The code solves the standard energy and momentum conservation and mass balance correlations with semi-empirical expressions augmenting the analytical methods where it is required. For example, loss coefficients are based on measured data, and the void correlation and two-phase multiplier are based on hydraulic loop data.

[Proprietary Information Deleted]

It is judged that this steady-state hydraulic benchmarking provides a reasonable indication that, for a given heat flux distribution, hydraulic conditions will be predicted by the Scandpower version of RAMONA-3B with sufficient accuracy to provide sufficiently reliable predictions of peak fuel enthalpy during the CRDA to confidently demonstrate that the CRDA design bases are satisfied using the ABB methodology described in Section 4.

Doppler Effect

The effect of resonance broadening on reactivity, or the Doppler effect, is treated in the Scandpower version of RAMONA-3B by modifying the nodal fast group absorption, fission, and removal cross sections by a term of the form:

$a_f(\sqrt{T_f}-\sqrt{T_{f0}})$, where T_f is the fuel temperature, and T_{f0} is a reference fuel temperature.

The coefficient, a_f, is a polynomial function of moderator density at each burnup for each fuel type and is calculated from the lattice physics code mainline depletion and branch calculational results. Therefore, the accuracy of the Doppler feedback model depends on the adequacy of this polynomial description in RAMONA-3B and the capability of the PHOENIX cross sections to reliably reflect the impact of changes in fuel temperature.

As pointed out in Reference 2, a_f is sensitive to changes in voids. Therefore, the recommendation in Reference 2 that the impact of void fraction be incorporated into RAMONA-3B has been implemented in the Scandpower version of RAMONA-3B by making a_f a polynomial function of moderator density.

The cross sections upon which the coefficients, a_f, are based are computed for uncontrolled assemblies. The sensitivity of the predicted Doppler feedback to the presence or absence of control rods in the cross section data base is sufficiently small that this approximation will not significantly affect the capability of the methodology to demonstrate that the CRDA design bases are satisfied. [Proprietary Information Deleted]

Therefore, it is judged that the modeling of the Doppler effect in Scandpower version of RAMONA-3B is adequate.

[Proprietary Information Deleted]

A more direct check of the capability of the cross section library, in conjunction with PHOENIX, to calculate Doppler reactivity is also available. PHOENIX predictions can be compared with the MCNP-3A benchmark results reported in Reference 23. For example, PHOENIX predictions utilizing the current BWR ENDFB-IV library are compared with the benchmark results in Figure 5.3.1. As shown in Figure 5.3.1, the agreement between the Doppler Coefficients calculated with the current ENDFB-IV library and the benchmark results from Reference 23 is well within the ±10% uncertainty in the Monte Carlo results from Reference 23. Therefore, Figure 5.3.1 demonstrates that the cross sections from this library in conjunction with PHOENIX will provide sufficiently reliable predictions of the Doppler Effect for demonstrating that CRDA fuel rod enthalpy limits are satisfied.

Therefore, it is concluded that the Scandpower version of RAMONA-3B modeling of Doppler reactivity and the ABB approach for confirming the adequacy of cross section data input to RAMONA is sufficient to assure that Doppler reactivity is predicted with sufficient accuracy during the CRDA to confidently demonstrate that the CRDA design bases are satisfied using the ABB methodology described in Section 4.

Thermal Energy in the Pellet and Energy Conduction to the Coolant

The modeling of the fuel pellet and heat conduction to the coolant is particularly important since:

(1) The design limit is on peak fuel enthalpy which is directly affected by the fuel temperature and heat conduction from the fuel.

(2) The fuel temperature determines the Doppler reactivity feedback which is the primary feedback mechanism in the subcooled range when there is no substantial boiling.

(3) The heat conduction to the coolant will affect the onset and degree of boiling which, in turn, will be important in the power range in establishing the void reactivity feedback.

As discussed in Section 4.5.1, the thermal energy distribution and heat conduction from the pellet to the coolant is performed in RAMONA-3B by solving standard coupled, time-dependent radial heat conduction differential equations in the pellet and clad. The pellet and clad are nodalized into concentric rings, and heat conduction in the axial direction is neglected. Dimensional changes in the pellet and cladding are neglected. [Proprietary Information Deleted]

Therefore, the methodology used to provide the input parameters required by the Scandpower version of RAMONA-3B heat conduction and fuel temperature models assures that physical effects that can effect the pellet temperatures and heat conduction to the coolant are accounted for adequately .

5.3.2 Application of RAMONA-3B to the CRDA

5.3.2.1 Background

The potential suitability of utilizing RAMONA-3B for CRDA analyses has been recognized for some time. The capability of the code to model each assembly and control rod in the core and perform detailed transient calculations make the analysis of a very local reactivity insertion characteristic of the CRDA more straightforward and less ambiguous for RAMONA-3B than for codes requiring simpler geometries. Furthermore, the hydraulic model is sufficiently detailed to allow modeling of the hydraulic feedback effects.

Brookhaven National Laboratory (BNL) has studied the CRDA in detail using both the BNL-TWIGL code and RAMONA-3B. For example, the accident was studied in References 11 and 12 using BNL-TWIGL, and comparative analyses utilizing BNL-TWIGL and RAMONA-3B are documented in Reference 13. The sensitivity of peak power and fuel enthalpy to dropping the same worth rod from the center location relative to a location elsewhere in the core was studied with RAMONA-3B in Reference 14. A sample CRDA for a BWR/4 using RAMONA-3B was discussed in Reference 4.

It was pointed out in Reference 13 that neglect of coolant superheat in BNL-TWIGL introduced a relatively large error for the hot-zero power case considered. The hydraulics model in the Scandpower version of RAMONA-3B provides for coolant superheat. While experimental data on transient superheat in the fluid is not available which will directly support the description of water superheat in the Scandpower version of RAMONA-3B, comparison of code predictions with available transient and steady-state data indicates that the treatment of superheated water in RAMONA-3B is not inconsistent with the available data. Furthermore, it was suggested in Reference 2 that the effect of moderator density on Doppler broadening should be included in RAMONA, and this effect has been included in the Scandpower version of RAMONA-3B used by ABB.

Scandpower also addressed the application of RAMONA to the CRDA in detail and concluded in that the code is particularly well suited to the evaluation of the CRDA.

Therefore, both BNL and Scandpower have evaluated the applicability of RAMONA-3B to the CRDA and concluded that the code is particularly well suited to that application. Furthermore, approximations adversely affecting the results of the CRDA identified by the BNL work have been corrected in the Scandpower version of RAMONA-3B version utilized by ABB.

5.3.2.2 Qualification of RAMONA-3B Against SPERT Experiments

This section contains results of RAMONA-3B simulations of six of the SPERT-III E-core power excursion tests. The purpose of the analyses is to demonstrate the ability of the ABB methodology using the RAMONA-3B code to predict the behavior of light water reactor cores during reactivity-initiated transients.

Reactivity Insertion Tests with the E-core

An objective of the SPERT program was to obtain data for the evaluation of analytical models. One of the goals of this program was to determine the nuclear behavior of UO_2-fueled reactors for the reactivity insertion accident which represents one type of postulated accident. The series of tests described in this document produced experimental reactivity insertion and power excursion data for initial operating conditions that are similar to commercial light-water reactor conditions.

A total of 80 non-fuel-damaging power excursions were performed. Forty of the tests were initiated from cold startup conditions, 32 tests simulated hot startup conditions, five tests simulated hot standby conditions, and three tests simulated initiation from the operating power range. The tests were initiated with rapid reactivity insertions ranging from 0.5 dollars to 1.3 dollars resulting in power excursions with reactor periods ranging from 1000 ms to 10 ms. (Reactivity in dollars is defined as the ratio of (K_{eff}-1) to the fraction of delayed neutrons, ß).

References 24 through 28 contain detailed descriptions of the SPERT tests. For example, the results of the SPERT-III E-core experiments are summarized in Reference 24 in tabular form as well as in a set of 80 diagrams. The curves and diagrams depict transient reactor power, energy, and system reactivity. A few of the figures and tables from References 24 through 28 have been reproduced in this document for convenience.

Design characteristics of the E-core are presented in Table 5.3.2. A cross-section of the SPERT-III E-core is shown in Figure 5.3.2. Location of the cruciform poison rod to be dropped from the core in the geometric center of the reactor required that fuel assemblies with two sizes be used in the reactor to maintain the symmetry of the lattice. The fuel in all assemblies is in the form of UO_2 pellets with 4.8 wt-% U^{235}. The diameter of the fuel rods is 1.07 cm (0.42 inch). Each of the fuel rods contains 38.5 g of U^{235} with an active fuel length of 97.3 cm (38.3 inches). The cladding material is stainless steel.

The core is composed of 60 assembly locations with a pitch of 1.49 cm (0.585 inches) and a core diameter of approximately 66 cm (26 inches). The four central assemblies each contain 16 fuel rods. There are also 48 assemblies containing 25 fuel rods. The remaining eight assembly locations contain control elements with two in each quadrant. Figure 5.3.3 shows sketches of the 25-rod fuel assembly and the control element. The two rods in each quadrant are joined by a yoke and are, therefore, moved as a unit. The control elements contain both fuel and poison sections. The lower section contains fuel and is referred to as the "fuel follower". The upper section contains B^{10}-stainless steel neutron absorber material. The fuel section in the control element is a 16-rod assembly, and the poison section is a square box constructed of stainless steel containing 1.35 wt-% B^{10}.

The central transient cruciform rod used for the reactivity insertions also consists of two sections. The upper section is 142 cm (56 inches) long and is constructed of stainless steel. The lower section is 96.5 cm (38 inches) long and is constructed of 1.35 wt-% boron-10 stainless steel.

Initial criticality was typically achieved with the poison section of the central cruciform control rod extending below the core and the upper stainless-steel section in the core. In preparation for an excursion, the control assemblies were withdrawn to a predetermined position, and the reactor was maintained in a critical condition by inserting the poison section of the transient rod into the lower part of the core. The excursion was initiated by dropping the transient rod poison section from the core. The cruciform control rod acceleration was 5080 cm/s^2 (2000 in/s^2). The control rod attained a speed of 0.948 m/sec (3.11 ft/sec) in about 1.866 seconds with this acceleration.

In the present work, six of the 80 tests were randomly selected to provide a reasonable range of reactivity insertion for analysis with RAMONA-3B. Four of these tests were initiated from cold startup conditions, and two were initiated from hot startup conditions. The identification numbers and initial conditions for the tests simulated with RAMONA-3B in the present work can be summarized as follows:

Test	Pressure (MPa)	Temp. (°C)	Core Flow (kg/s)
Cold start-up cases			
22	0.1	20	0
18	0.1	20	0
49	0.1	24	0
43	0.1	26	0
Hot start-up cases			
32	10.3	126	126
62	10.3	260	756

The reported uncertainties in the SPERT data are 15 % for reactor power, 4 % for reactivity insertion, and 17 % for energy release to time of peak power. These tests were selected to obtain a reasonable range of reactivity insertion due to the ejection of the central transient rod.

RAMONA-3B Model

[Proprietary Information Deleted]

Results

The predictions from the RAMONA-3B simulations are compared to experimental results for the six cases selected in Table 5.3.2. The experimental values of reactivity inserted are compared with the values predicted by the RAMONA-3B simulations for the cases considered in Figure 5.3.8. The ±4% uncertainty is shown for the experimental results. Figure 5.3.9 is a comparison of the energy to peak power as a function of inserted reactivity for the six SPERT cases evaluated. The error bars are those reported in Reference 27. Figures 5.3.10 through 5.3.15 show comparisons of the relative power excursions predicted by the RAMONA calculations with the experimental results reported in Reference 24 as well as the reactivity components predicted by RAMONA for the six tests evaluated.

Figure 5.3.8 is a comparison of inserted reactivity predicted by RAMONA with experimental values. The correlation is very good. The sensitivity shown in Figure 5.3.7 demonstrates that the definition of the inserted reactivity must be quite precise to obtain reasonable predicted power excursions. It should be noted that the RAMONA reactivity shown in Figure 5.3.8 is a relative change in static values of $k_{effective}$ between the fully inserted and the fully withdrawn configurations. Figure 5.3.10 through 5.3.15 show the transient variation of the system reactivity for each simulated SPERT test. The reactivities described in these figures are derived by perturbation theory and are functions of the two-group, three-dimensional neutron flux distributions and local cross-section variations. They are included in the code to provide insight into the reactivity balance and does not control the simulated process. For large changes in system reactivity the approximate nature of this reactivity parameter should be recognized. Therefore, its absolute value does not necessarily coincide exactly with the inserted reactivity strictly defined as the change in static system reactivity.

Figure 5.3.9 is a comparison of the integrated energy between the time the central rod is dropped and the time the peak power occurs as a function of the inserted reactivity predicted by RAMONA with the corresponding experimental values. The agreement is quite good since the predicted values generally agree with the experimental values to within the quoted uncertainties.

Figures 5.3.10 through 5.3.15 contain comparisons of the power excursions measured experimentally with the predictions of the RAMONA calculations as well as the reactivity components predicted by RAMONA. The reactivity components shown are estimates of the reactivity inserted by the dropped central control rod, the Doppler feedback reactivity, and the "Moderator density" feedback reactivity. The 'Doppler' and 'Moderator density' components represent the reactivity contributions due to the fuel temperature increase and moderator density decrease resulting from the power excursion, respectively. ' Total' refers to the sum of the three components. The contribution of these components at the time of peak power are summarized in Table 5.3.3.

Discussion

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Conclusion

It is concluded that the RAMONA-3B simulations using the ABB CRDA methodology show good agreement with the SPERT-III E-Core tests for which the nominal initial conditions quoted appear to reflect the actual situation. The comparisons in this section demonstrate that for a peak power consistent with the experimental data, the ABB methodology using RAMONA-3B predicts resulting values of inserted reactivity, power shape, integrated energy, and time-to-peak power which agree with the experimental values to within the experimental uncertainties. Therefore, it is concluded that the ABB methodology predicts the results of the SPERT-III E-Core tests for which the comparisons were made to within the uncertainties in the tests and the uncertainties associated with the information available regarding those tests.

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TABLE 4.4.1

WITHDRAWAL SEQUENCE

STEP	ARRAY	MOVE	STEP	ARRAY	MOVE
1	1A	0-48	31	7A	0-4
2	1B	0-48	32	6A	0-4
3	2A	0-48	33	6B	0-4
4	2B	0-48	34	7C1,7C2	4-8
5	3A	0-4	35	7B	4-8
6	3B	0-4	36	7A	4-8
7	3A	4-6	37	6A	4-8
8	3B	4-6	38	6B	4-8
9	3A	6-8	39	7C1,7C2	8-12
10	3B	6-8	40	7B	8-12
11	3A	8-10	41	7A	8-12
12	3B	8-10	42	6A	8-12
13	3A	10-12	43	6B	8-12
14	3B	10-12	44	7C1,7C2	12-48
15	3A	12-48	45	7B	12-48
16	3B	12-48	46	7A	12-48
17	4A	0-4	47	6A	12-48
18	4B	0-4	48	6B	12-48
19	4A	4-6	49	8B1,8B2	0-4
20	4B	4-6	50	8A1,8A2	0-4
21	4A	6-8	51	5B	0-4
22	4B	6-8	52	8B1,8B2	4-8
23	4A	8-10	53	8A1,8A2	4-8
24	4B	8-10	54	5B	4-8
25	4A	10-12	55	5B	8-12
26	4B	10-12	56	5A	0-12
27	4A	12-48	57	5B	12-48
28	4B	12-48	58	5A	12-48
29	7C1,7C2	0-4	59	8B1,8B2	8-12
30	7B	0-4	60	8A1,8A2	8-12

TABLE 4.5.1

Proprietary Information Deleted

TABLE 5.3.1
DESIGN CHARACTERISTICS OF THE SPERT-III E-CORE
(FROM REFERENCE 24)

Component	Specification
	Vessel and Primary System
Vessel Type	All-welded multilayer vessel
Vessel Composition	304L stainless steel
Vessel Size	4-ft ID by 23-3/4 ft long
Design Pressure	2500 psig
Design Temperature	700°F
Flow Characteristics	0 to 20,000 gpm upward through core
Heat Removal Capabilities	Up to 60 MW for 1/2-hr duration
	Core
Configuration	Approximately cylindrical, 26-in. diam.
Number and Type of Fuel Assemblies	48 twenty-five-rod assemblies 12 sixteen-rod assemblies
Moderator-Reflector	Light water
Nonmoderator-to-Moderator Ratio	1.03
	Fuel
Type	UO_2 pellets
Length of Fuel Rods	40.8 in.
Active Length	38.3 in.
Pitch	Square, 0.585 in.
Fuel Rod OD	0.466 in.
Clad Thickness	0.020 in.
Enrichment	4.8 percent
UO_2 Density	10.5 g/cm^3
Mass of UO_2 per Fuel Rod	913.5 g
Mass of U-235 per Fuel Rod	766.4 g
Mass of U-238 per Fuel Rod	38.5 g
Cladding	Type 348 stainless steel
	Control Rods
Number and Type	8 total, coupled in units of 2 per quadrant
Composition	Fuel follower and Type 18-8 stainless steel with 1.35 wt% B-10
Dimension of Poison Section	2.496 in. square by 46 in. long
Dimension of Fuel Follower	2.496 in. square by 45-41/64 in. long
	Transient Rod
Type	Cruciform shape
Composition	Upper section: 18-8 stainless Poison section: 1.35 wt% B-10 stainless steel
Length	Poison section: 38 in.

TABLE 5.3.2

COMPARISON OF RAMONA-3B SIMULATIONS WITH EXPERIMENTAL VALUES

Test	Transient rod insertion (inch)		Reactivity insertion ($)		Max. power (MW)		Energy release to time of peak power (MJ)		Time to peak power (s)		Fig
	SPERT	RAMONA	SPERT	RAMONA	SPERT	RAMONA	SPERT	RAMONA	SPERT	RAMONA	
22	3.8	4.06	0.77	0.77	2.1	2.3	6.9	8.7	13.7	15.5	5.3.10
18	4.3	4.49	0.90	0.91	4.3	4.7	6.7	7.2	5.3	5.4	5.3.11
49	4.5	4.79	1.00	1.01	10.6	11.3	2.1	2.6	0.97	1.16	5.3.12
43	5.2	5.32	1.21	1.22	280	310	6.0	6.5	0.230	0.23	5.3.13
32	5.0	4.98	1.09	1.09	66	69	3.1	3.7	0.39	0.40	5.3.14
62	8.0	8.35	1.10	1.08	97	100	4.5	4.9	0.370	0.42	5.3.15

TABLE 5.3.3

REACTIVITY FEEDBACK COMPONENTS

[illegible]	Sub-cooling (°C)	Core Flow (kg/s)	Reactivity Insertion ($)	Total Doppler plus Moderator Density (% of inserted)		Doppler Effect (% of Doppler plus Moderator Density)	Moderator Heating (% of Doppler plus Moderator Density)
				Exp	RAMONA		
22	79	0	0.77	34	35	83	17
18	80	0	0.90	24	24	90	10
49	76	0	1.00	8	8	90	10
43	74	0	1.21	16	14	93	10
32	187	126	1.09	9	7	89	10
62	53	756	1.10	9	10	78	10

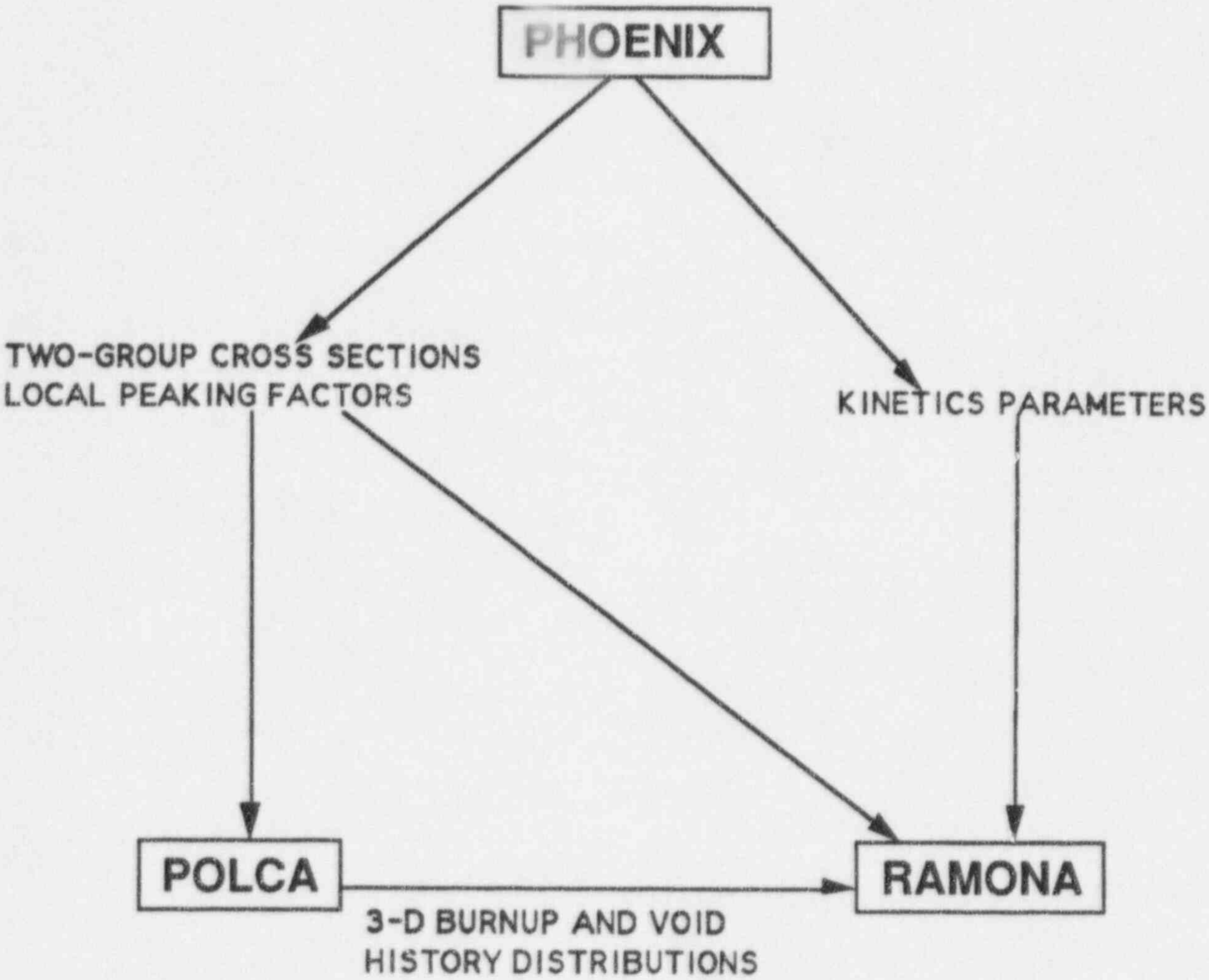
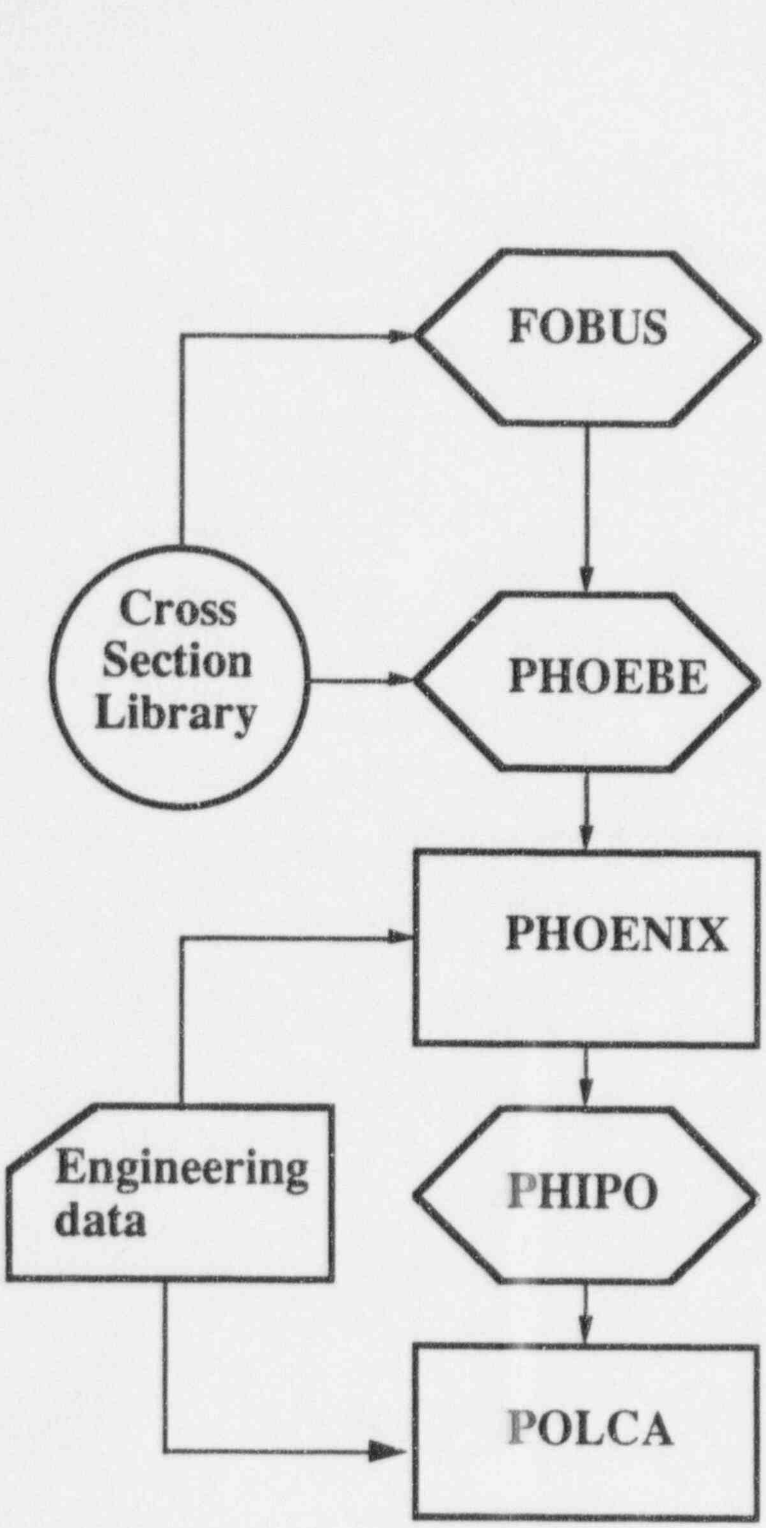


Figure 4.3.1 Data Flow to RAMONA-3B



FOBUS — **Monte Carlo transport code for burnable absorber cross-section generation**

PHOEBE — **Library processing code for nuclear cross-section data**

PHOENIX — **Two-dimensional multi-group transport theory code for lattice physics constants**

PHIPO — **PHOENIX output processing code for POLCA input generation**

POLCA — **3-dimensional modified one group nodal code for steady-state reactor core simulation**

Figure 4.3.2 Nuclear Design Codes

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59					6B	3A	5B	4A	5B	3A	6B				
55				5A	1B	9E	2B	10C2	2B	9E	1B	5A			
51			6A	3A	8B2	4A	7C2	3B	7C2	4A	8B2	3A	6A		
47		5A	1B	9D	2B	10B2	1A	9C2	1A	10B2	2B	9D	1B	5A	
43	6B	3A	8B1	4B	7B	3B	8A2	4B	8A2	3B	7B	4B	8B1	3A	6B
39	1B	9E1	2A	10B1	1A	9B	2A	10A2	2A	9B	1A	10B1	2A	9E1	1B
35	5B	4A	7C1	3B	8A1	4B	7A	3B	7A	4B	8A1	3B	7C1	4A	5B
31	2B	10C1	1A	9C1	2A	10A1	1A	9A	1A	10A1	2A	9C1	1A	10C1	2B
27	5B	4A	7C1	3B	8A1	4B	7A	3B	7A	4B	8A1	3B	7C1	4A	5B
23	1B	9E1	2A	10B1	1A	9B	2A	10A2	2A	9B	1A	10B1	2A	9E1	1B
19	6B	3A	8B1	4B	7B	3B	8A2	4B	8A2	3B	7B	4B	8B1	3A	6B
15		5A	1B	9D	2B	10B2	1A	9C2	1A	10B2	2B	9D	1B	5A	
11			6A	3A	8B2	4A	7C2	3B	7C2	4A	8B2	3A	6A		
7				5A	1B	9E	2B	10C2	2B	9E	1B	5A			
3					6B	3A	5B	4A	5B	3A	6B				

Figure 4.4.1 Group Assignment For The A Sequence

FIGURE 4.4.2 THROUGH FIGURE 5.3.1

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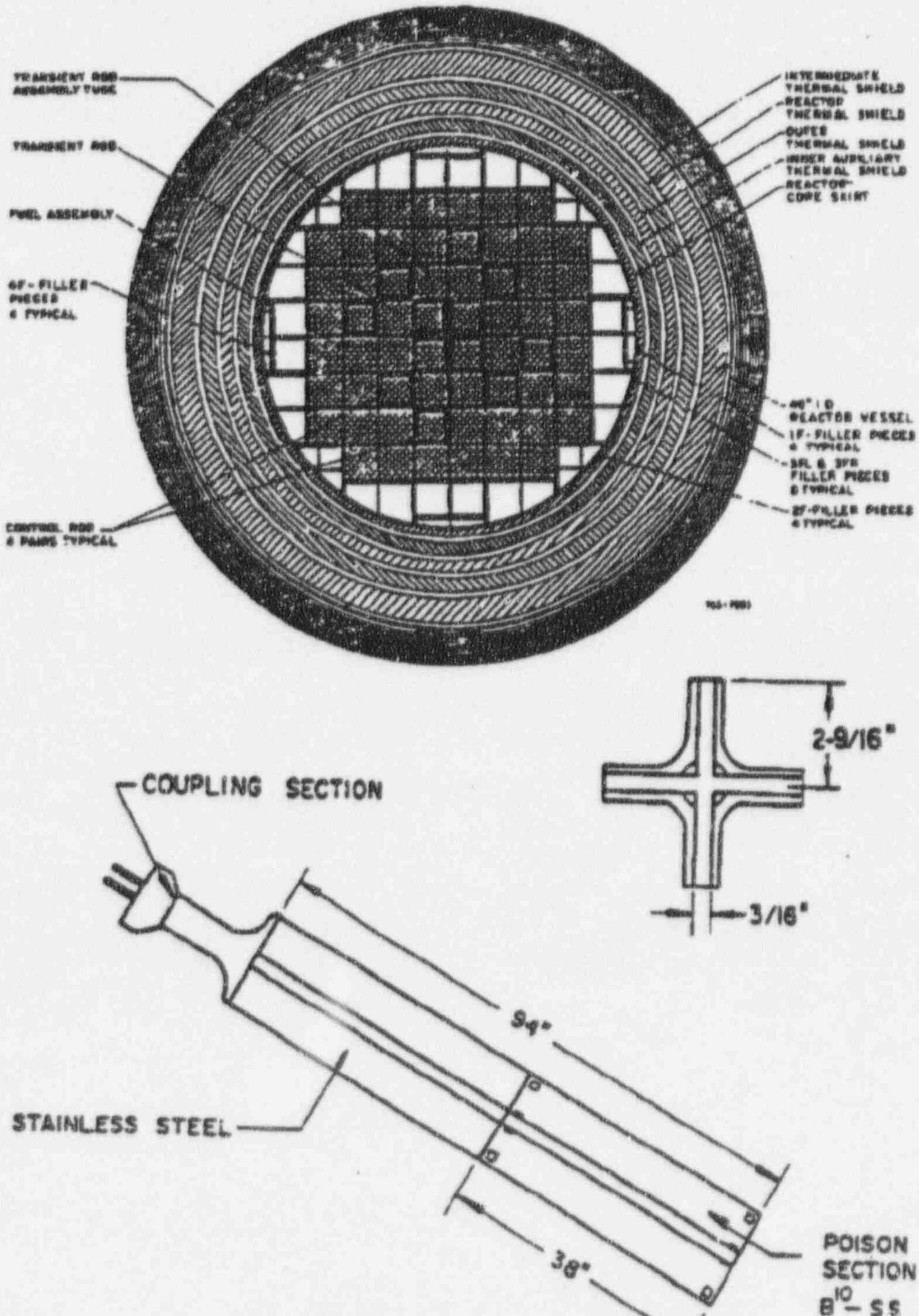
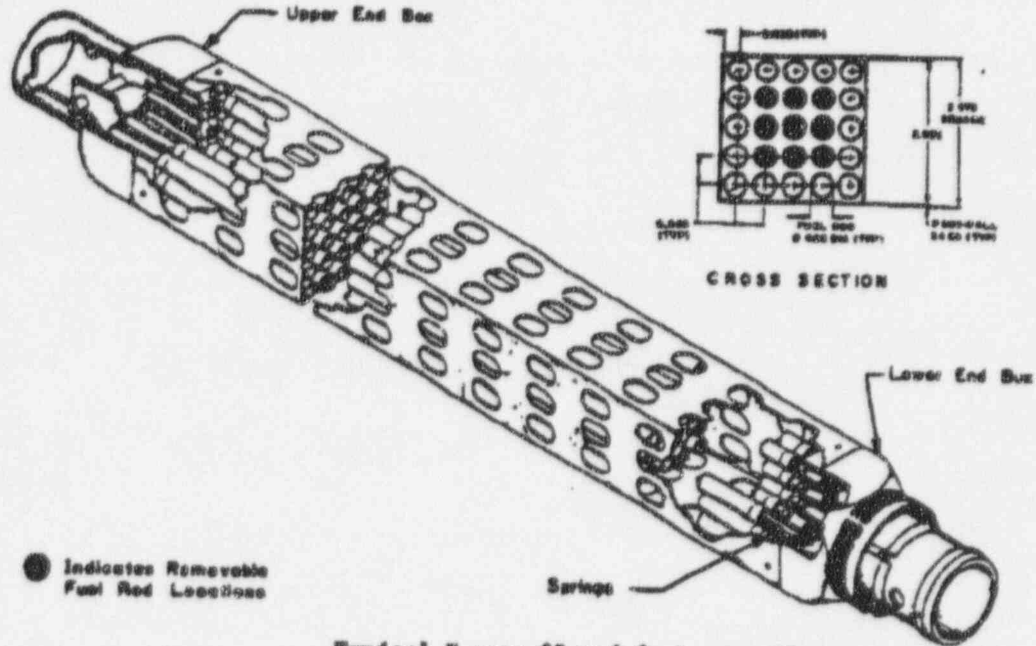
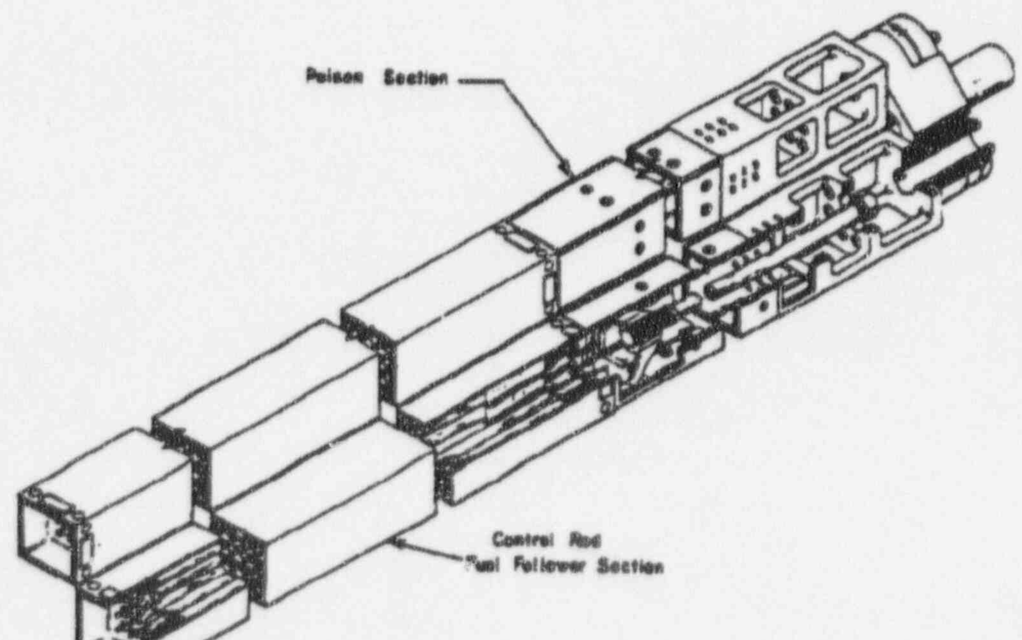


Figure 5.3.2 Cross Section of the SPERT-III E-core and the Transient Rod Assembly (From Reference 28)

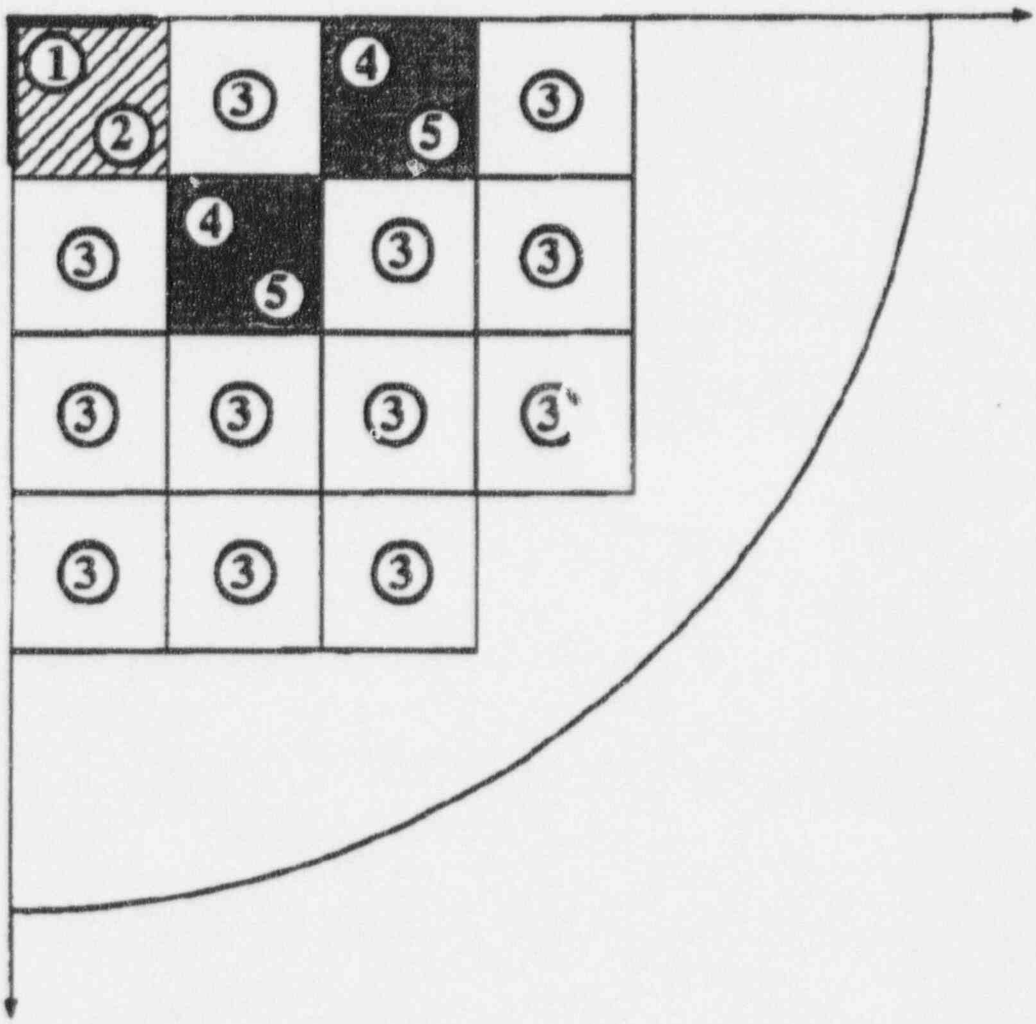


Typical E-core 25-rod fuel assembly



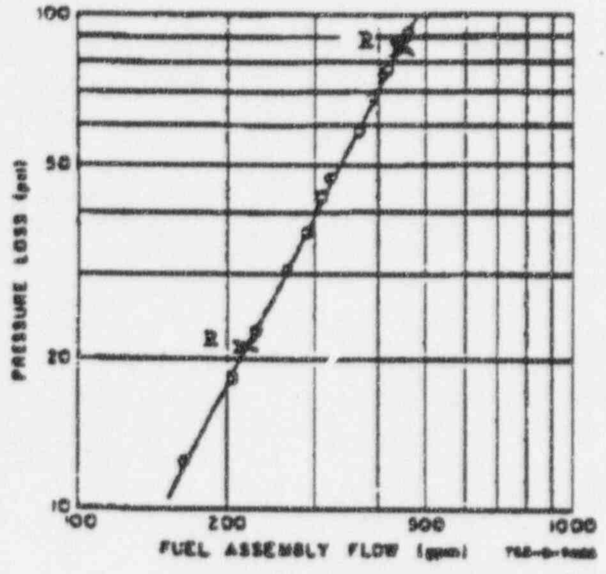
SPERT III E-core control rod pair

Figure /.3.3 SPERT-III E-Core 25-Rod Fuel Assembly and Control Rod Pair
(From Reference 28)

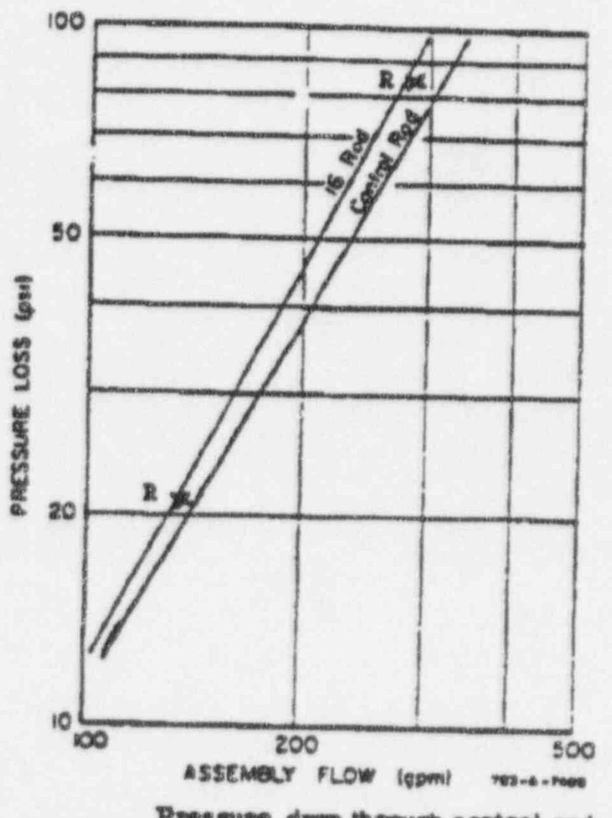


Fuel Type	Description
1	4x4 Assembly with Transient Rod Steel Section
2	4x4 Assembly with Transient Rod Poison Section
3	5x5 Fuel Assembly
4	4x4 Fuel Follower
5	Control Rod, Absorber Box

Figure 5.3.4 Fuel Type Assignments in RAMONA Model

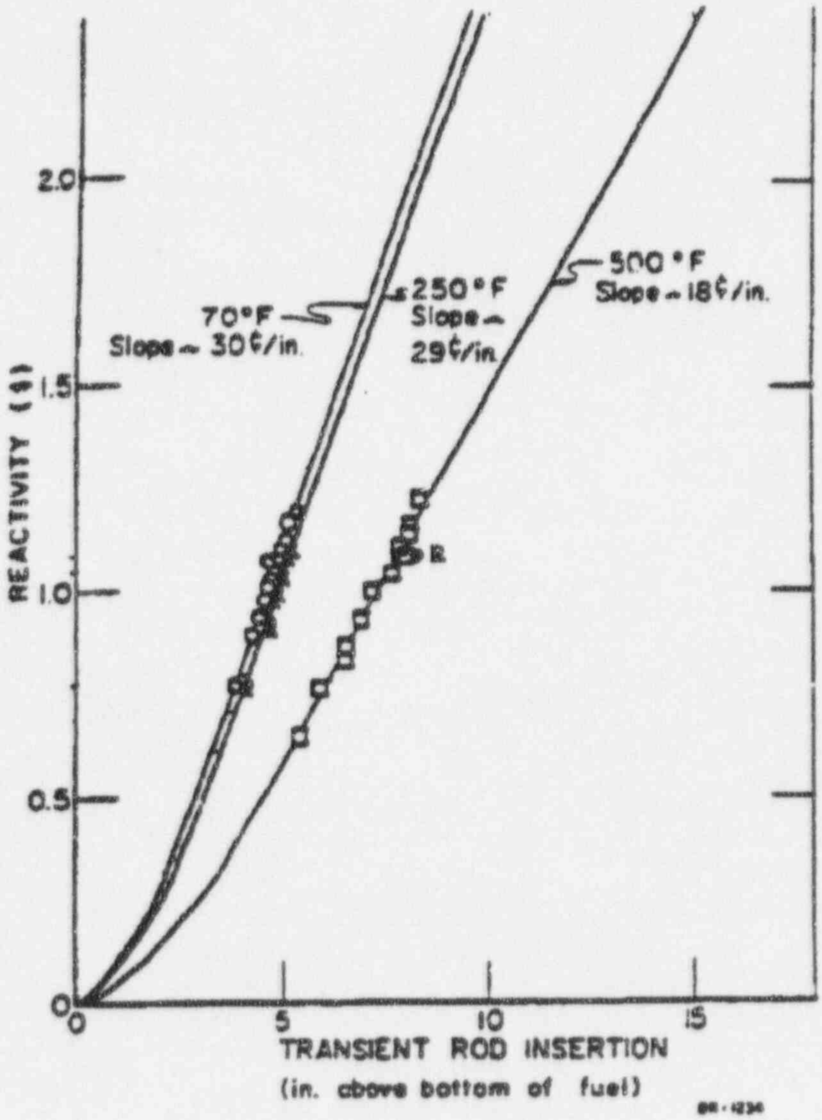


Pressure drop through 25-rod fuel assembly.



Pressure drop through control rod and 16-rod fuel assembly.

Figure 5.3.5 Experimentally Measured Pressure Drops from Reference 28



Note: X = RAMONA-3B Model

Figure 5.3.6 Comparison of Experimentally Measured Transient (central) Control Rod Worths from Reference 24 With RAMONA Predictions (shown as discrete points).

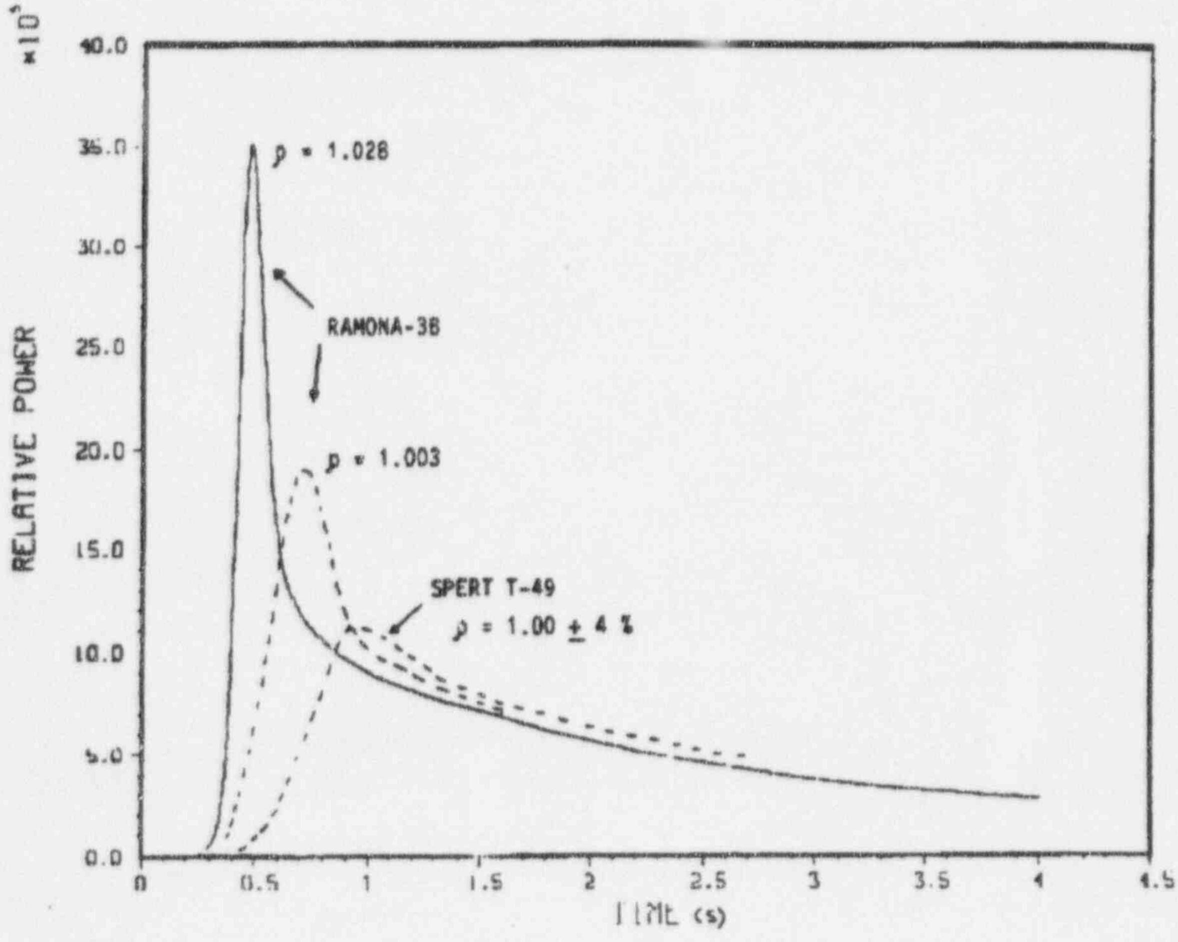


Figure 5.3.7 Impact of Small Variations in Reactivity Inserted by Dropped Rod on Core Power

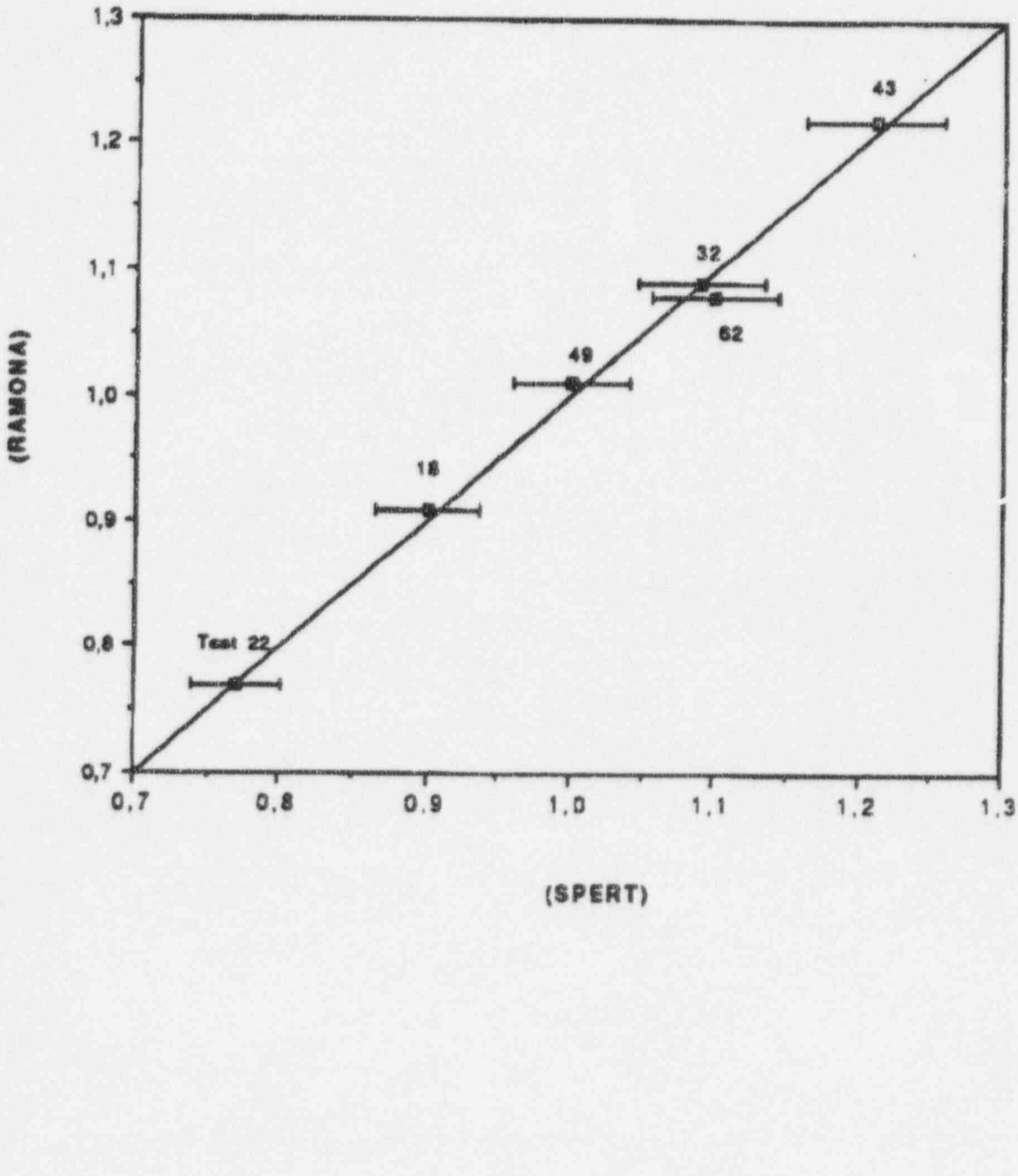


Figure 5.3.8 Comparison of Reactivity Inserted by Dropped Control Rod Predicted RAMONA Calculations with Experimental Values

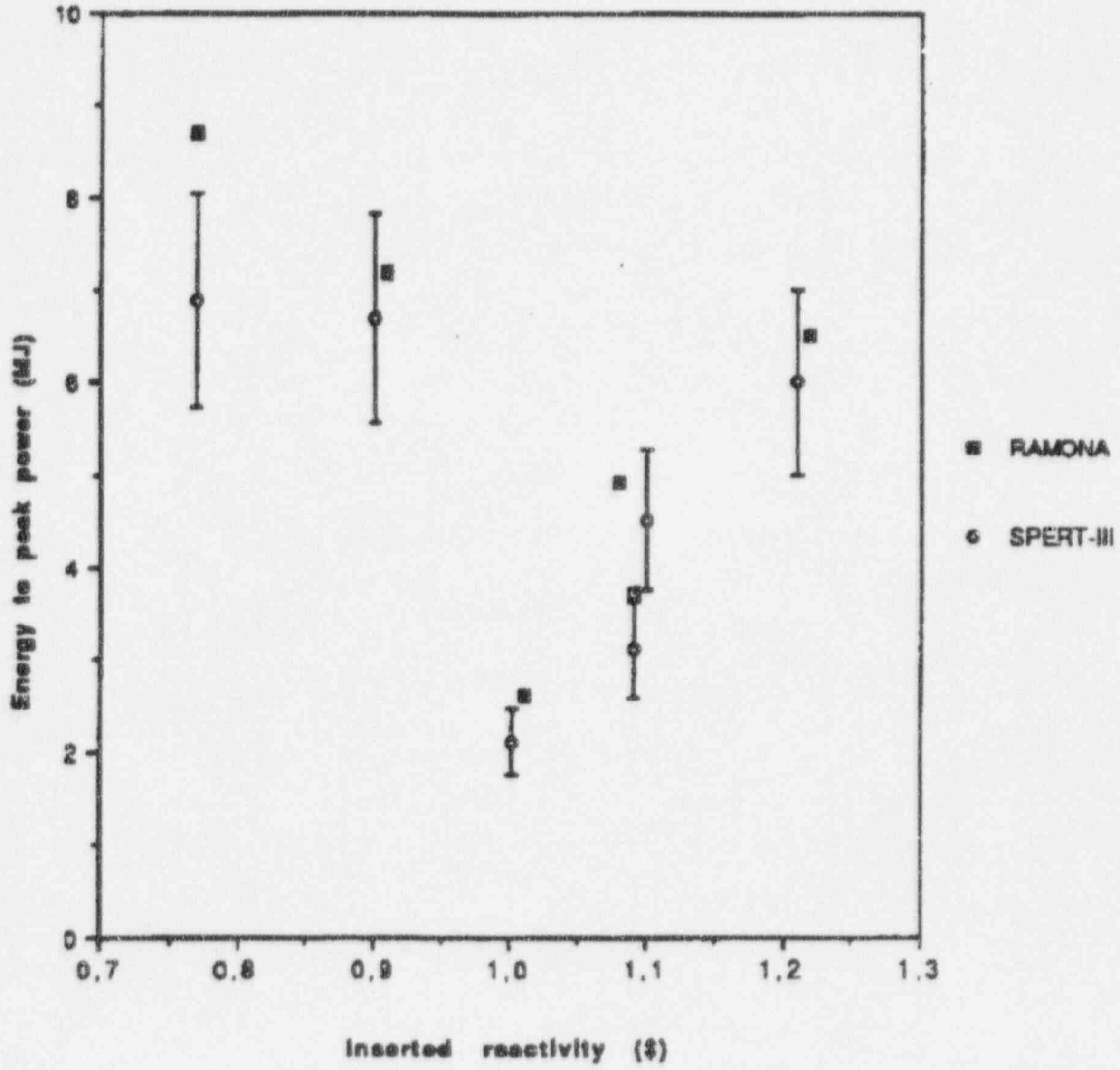


Figure 5.3.9 Comparison of Energy-to-Peak-Power as a Function of Inserted Reactivity Predicted by the RAMONA Calculations with Experimental Values

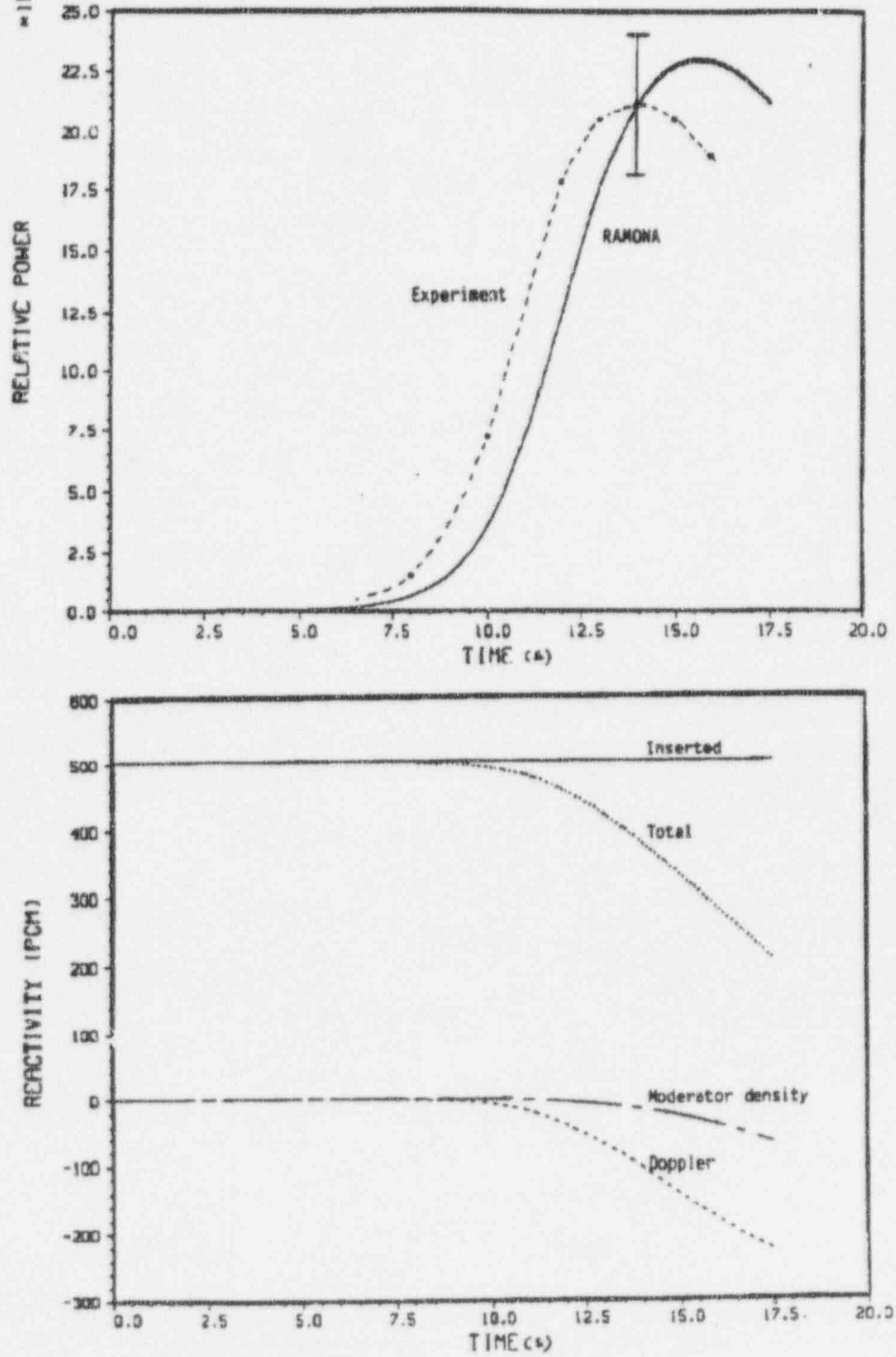


Figure 5.3.10 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values and RAMONA Reactivity Components for Test 22.

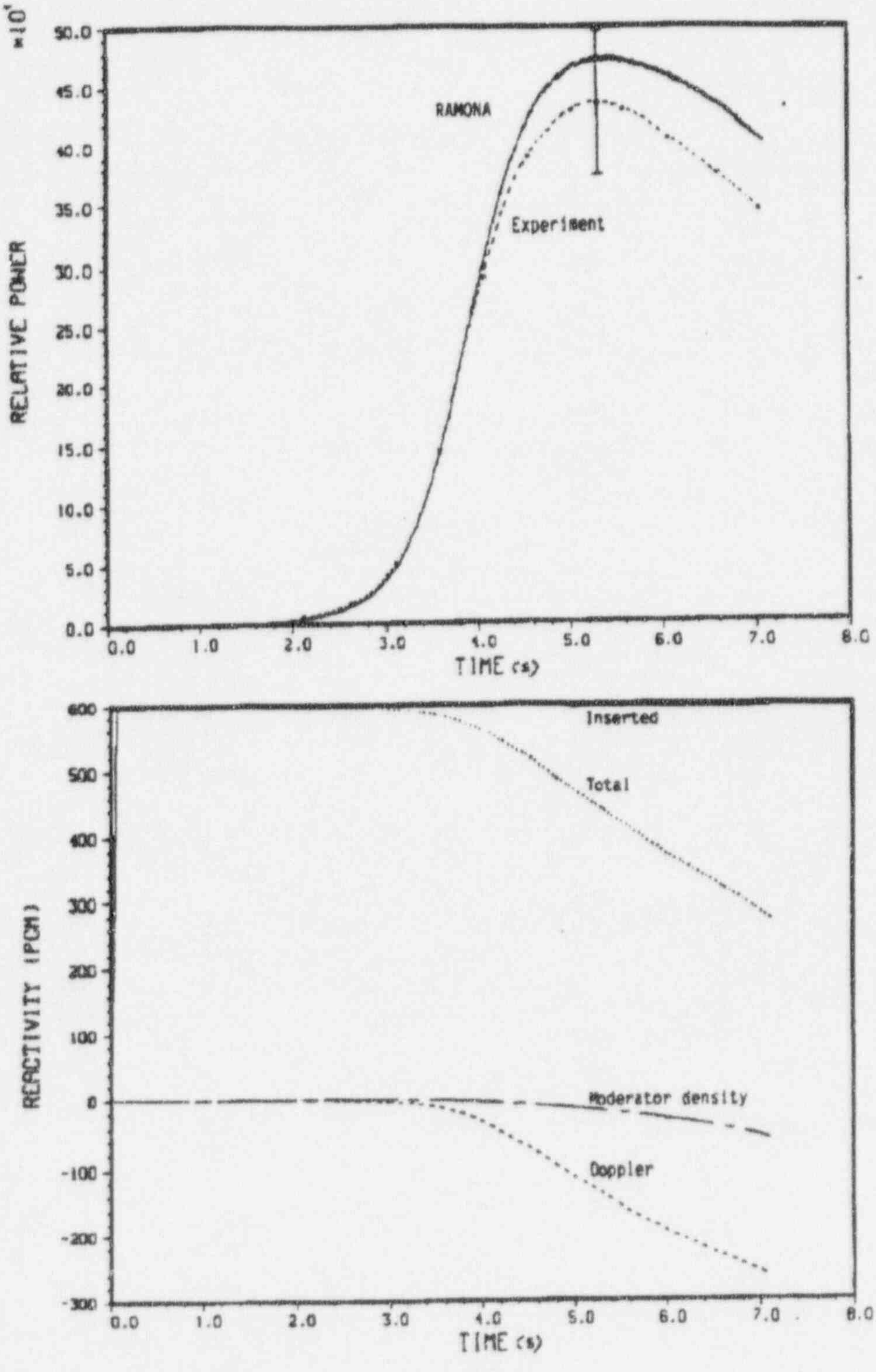


Figure 5.3.11 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values and RAMONA Reactivity Components for Test 18.

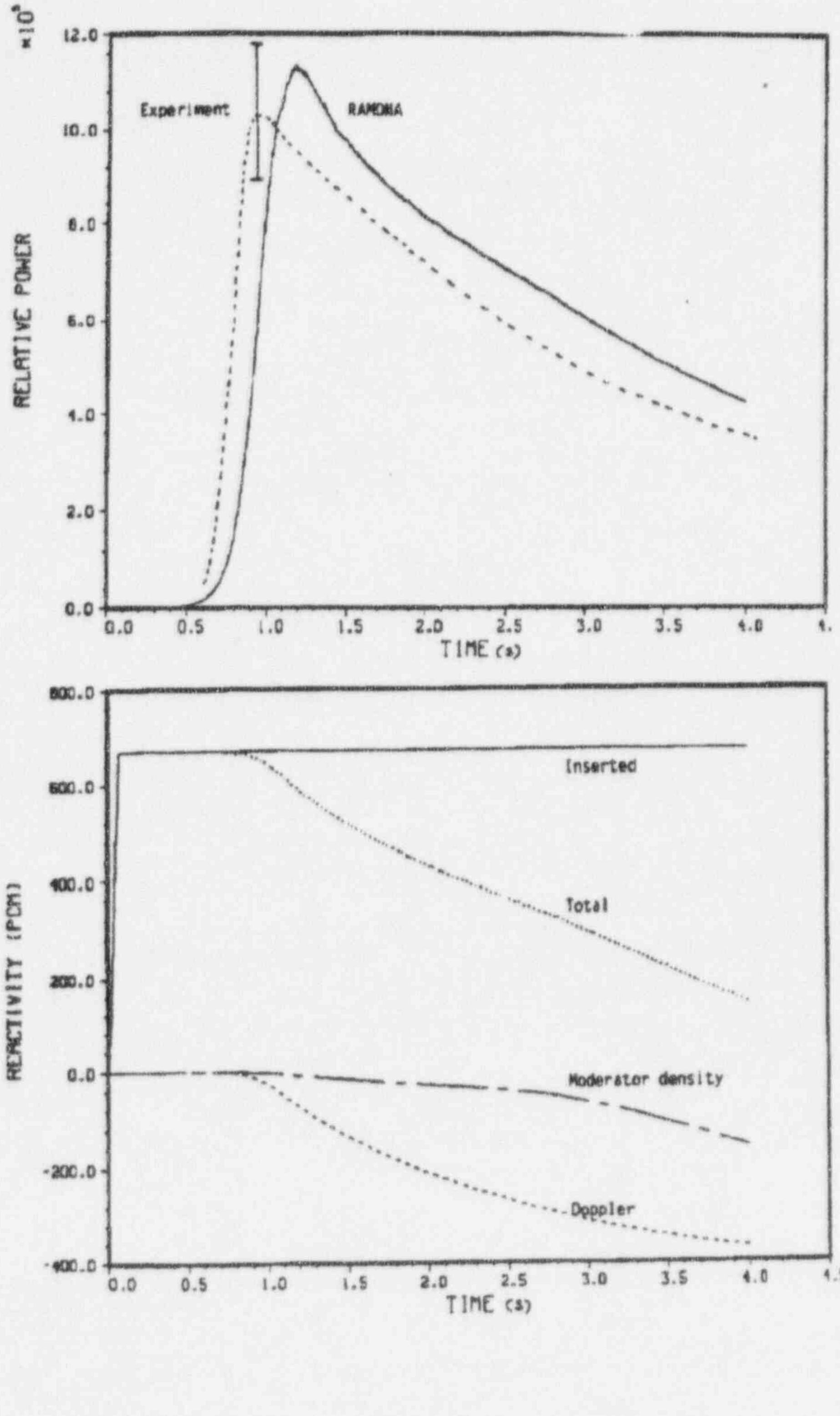
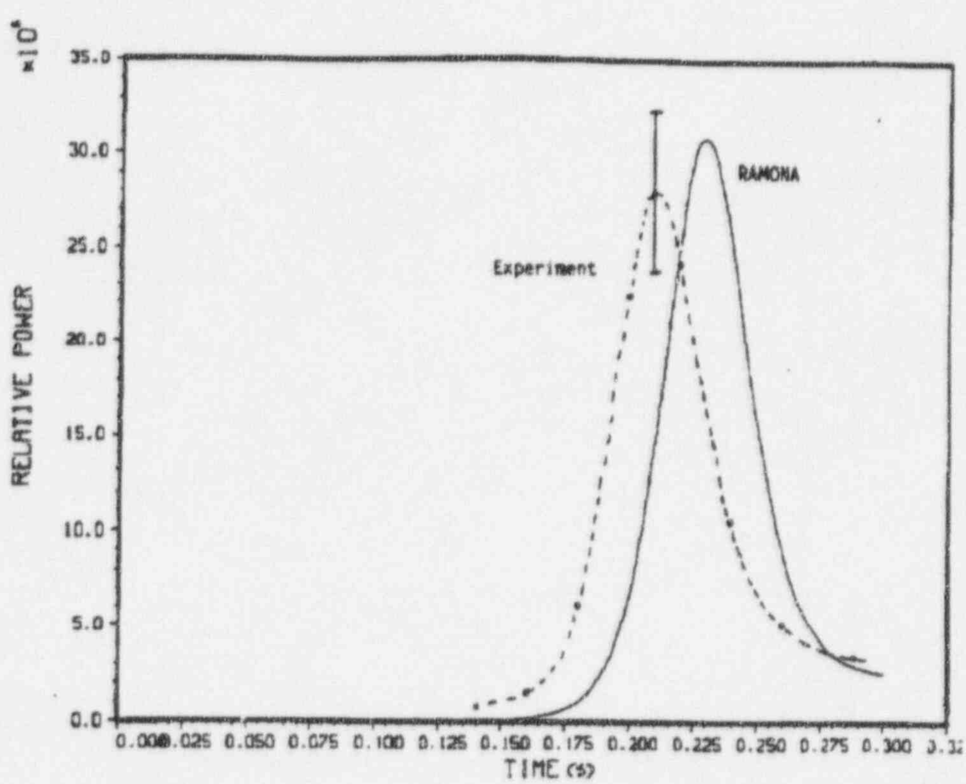


Figure 5.3.12 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values and RAMONA Reactivity Components for Test 49.



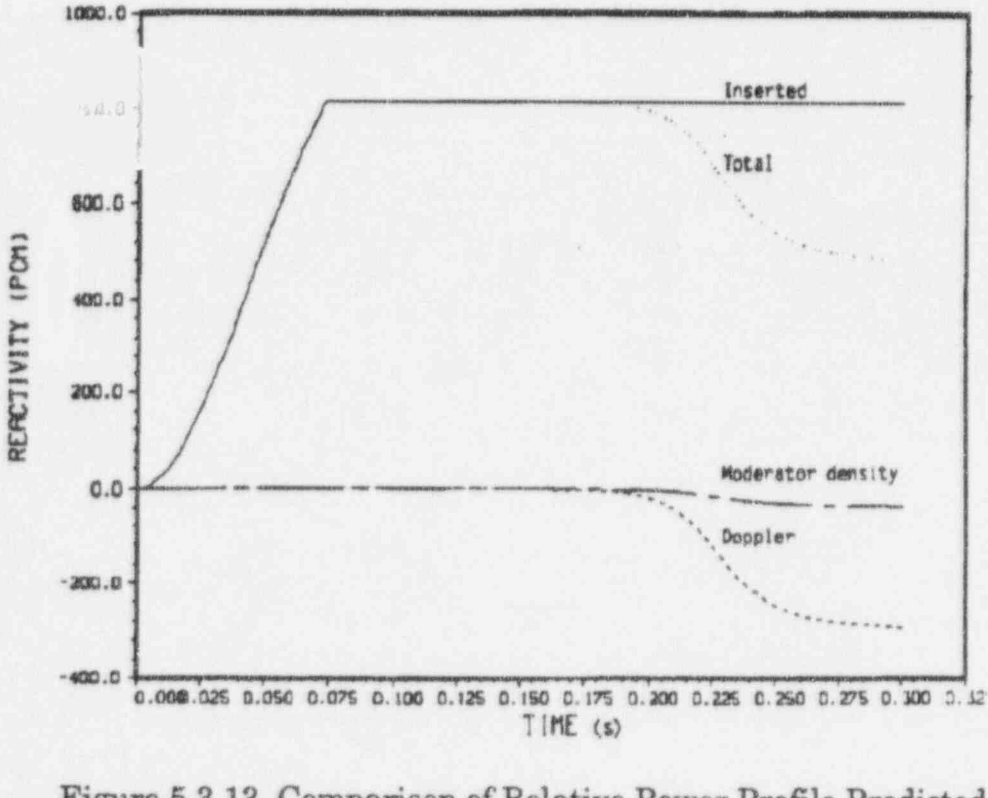


Figure 5.3.13 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values and RAMONA Reactivity Components for Test 43.

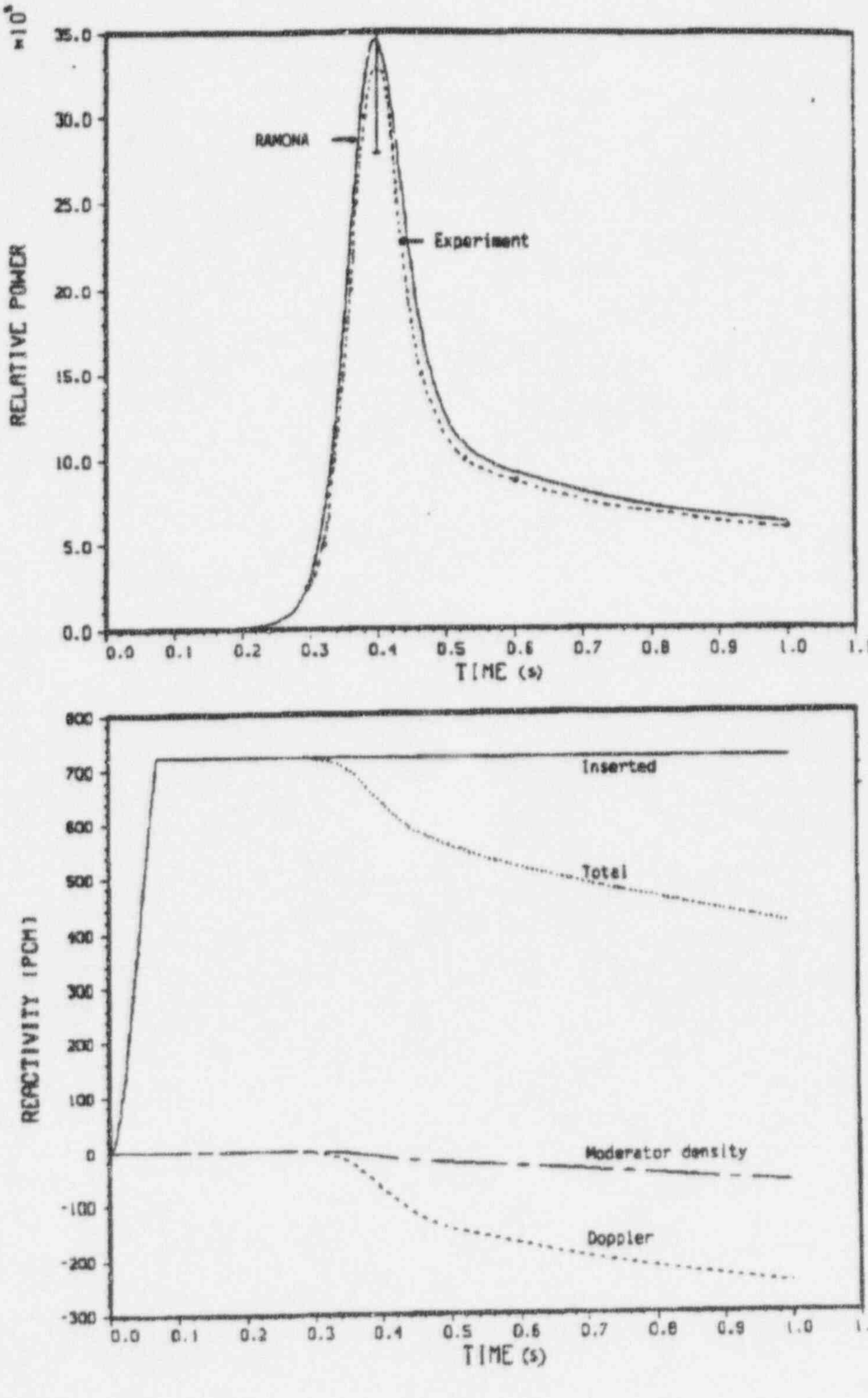


Figure 5.3.14 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values and RAMONA Reactivity Components for Test 32.

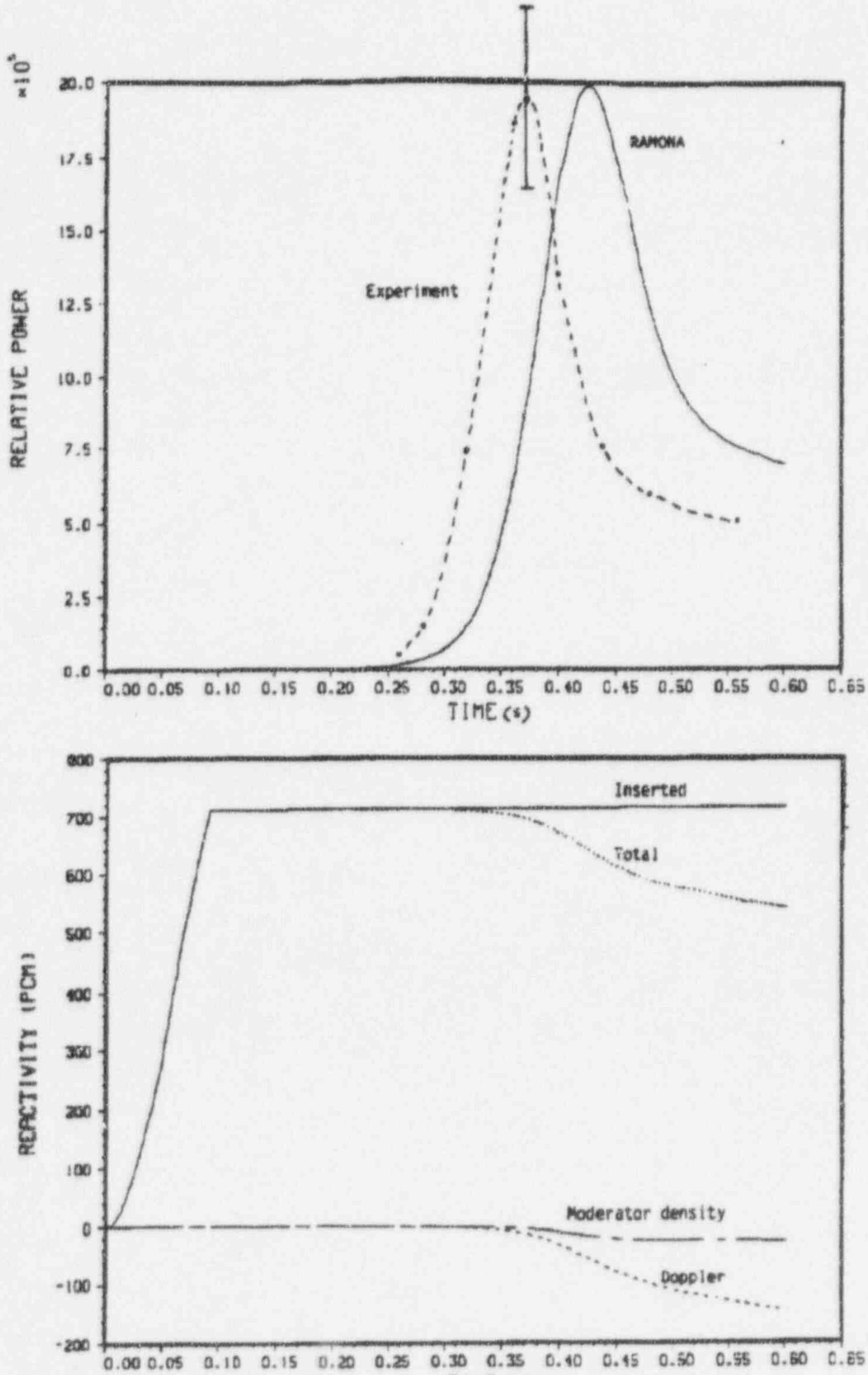


Figure 5.3.15 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values and RAMONA Reactivity Components for Test 62

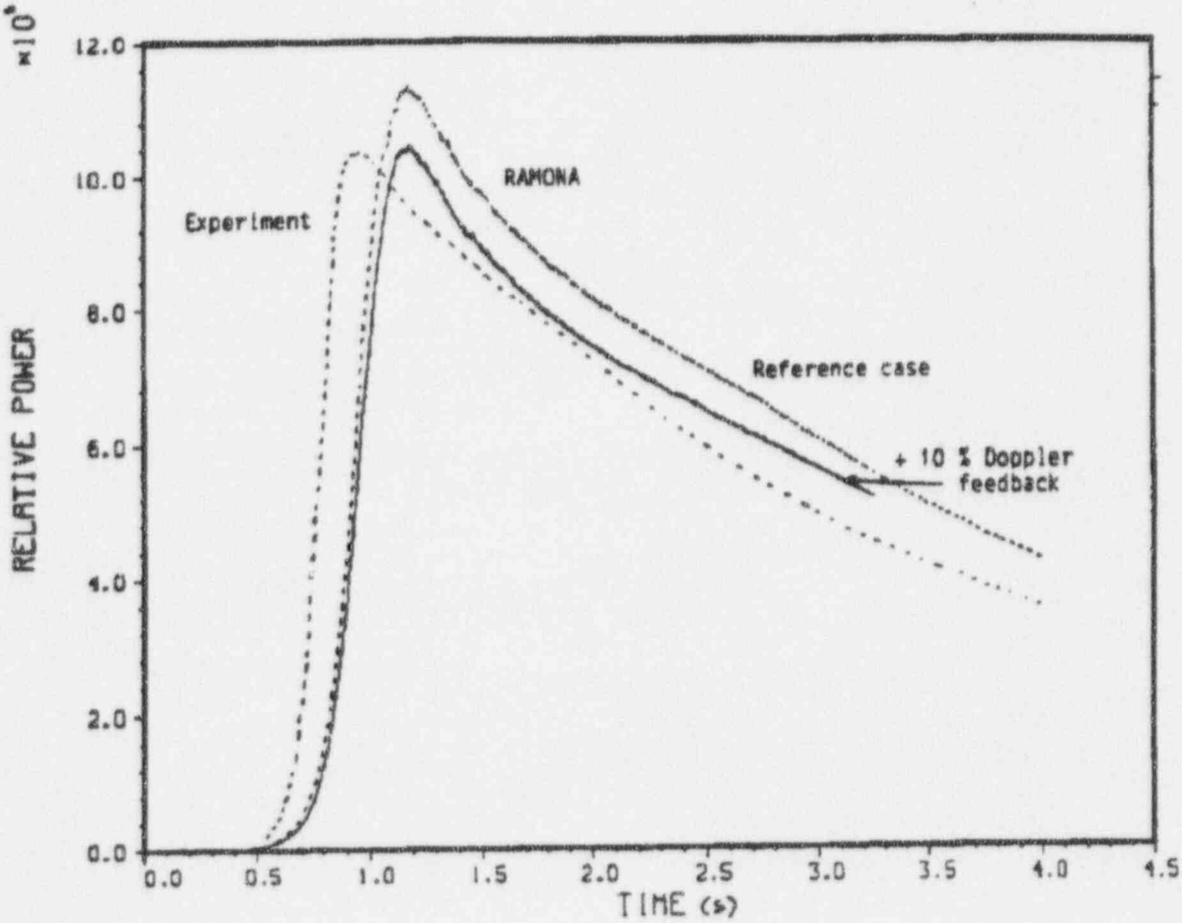


Figure 5.3.16 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values With a 10% Increase in Doppler Coefficient for Test 49.

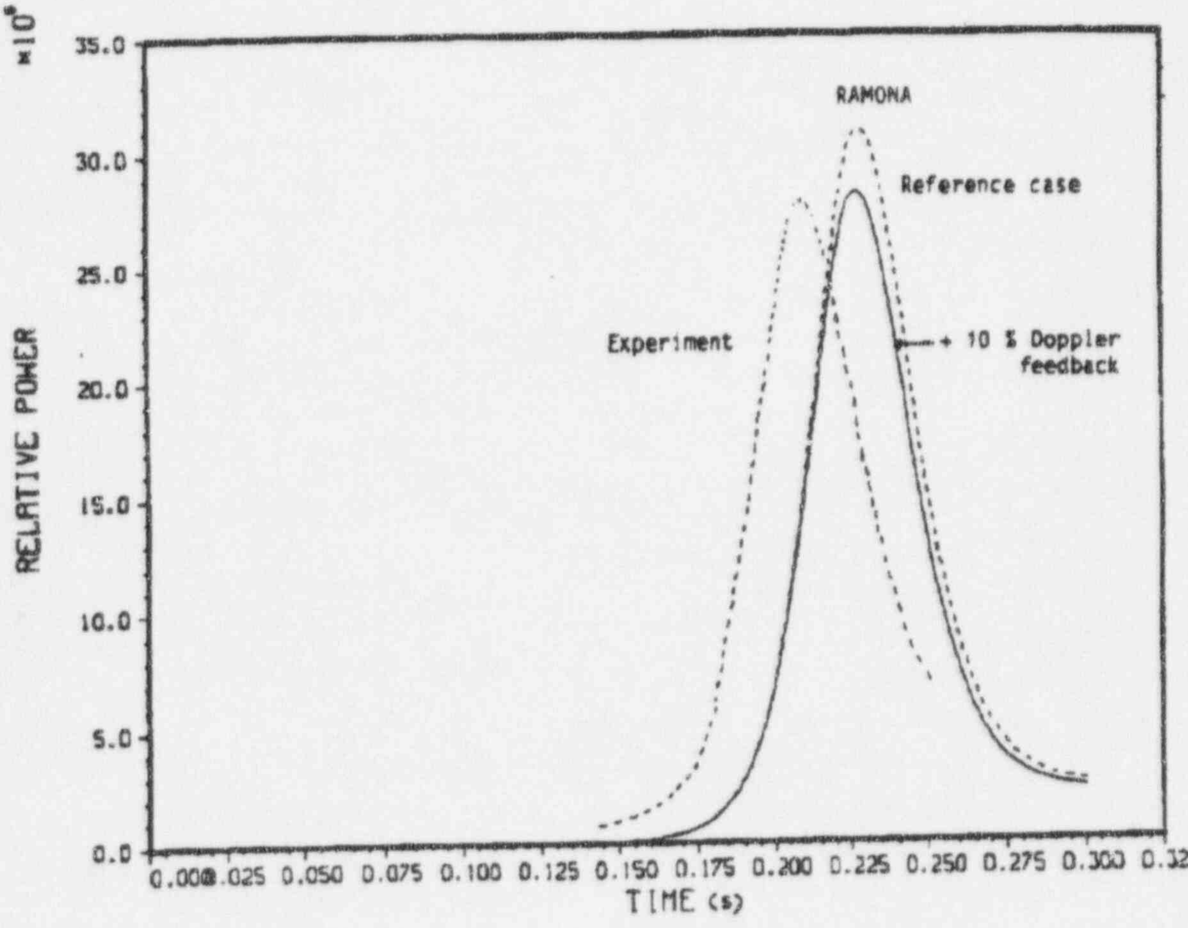


Figure 5.3.17 Comparison of Relative Power Profile Predicted by the RAMONA Calculations with Experimental Values With a 10% Increase in Doppler Coefficient for Test 43.

APPENDIX A: RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION ON RPA 89-112-P

A.1 Introduction

This appendix contains responses to the NRC Request for Additional Information regarding Reference 1 which was transmitted to ABB by the NRC letter identified in Reference 3. For convenience, all references used in this appendix are included in Appendix D.

Reference 1 provided a summary of the ABB methodology for analysis of the Control Rod Drop Accident (CRDA) using RAMONA-3B as well as a sample analysis for a 532-assembly BWR. The responses to requests for additional information regarding Reference 1 are included in this appendix.

Reference 2 provided an additional sample CRDA analysis illustrating the impact of a postulated CRDA in a plant equipped with high worth control rods. The responses to requests for additional information regarding Reference 2 are included in Appendix B.

Reference 4 was submitted in 1993 to clarify and summarize the ABB CRDA methodology as well as to provide further information supporting the qualification of the ABB CRDA analysis methodology using RAMONA-3B. The responses to requests for additional information regarding Reference 4 are included in Appendix C.

A.2 Questions and Responses

NRC Question A1

Describe any significant differences between the typical BWR/4-6 fuel and control rod designs and the designs to which this methodology will be applied.

ABB Response to Question A1

The ABB methodology described in Part II will be applied to BWR/2 through BWR/6 plants loaded with commercially available reload fuel. As discussed in Part II, nuclear data for RAMONA-3B will be calculated with a lattice code and three-dimensional core simulator accepted by the NRC for licensing applications. Specifically, the three dimensional static core simulator POLCA, in conjunction with the cross section generator code PHOENIX, are utilized for this evaluation and are documented in Reference 5. These are the same codes used for reload design purposes, and benchmark calculations are performed relative to available plant data to confirm that predictions of core reactivity and power shape are adequate. Section 5.2 of Part II provides a summary of the benchmark information in Reference 5.

The methodology will be applied to BWR/2 through BWR/6 plants equipped with commercially available control rods. The only significant sensitivity to control rod design expected for the CRDA might be the reactivity worth of the control rods installed in the reactor. For example, the "high worth" control rods referred to in Part IV might be installed rather than the "standard" control rods referred to in Part III. As discussed in Part II, the ABB static methods as well as the RAMONA-3B code are used to predict reactivity worths which are sufficiently accurate for evaluation of the CRDA. Part II describes in detail the ABB methodology for establishing limiting dropped rod candidates and evaluating the impact of a postulated CRDA for those candidates.

Please also see the response to Question C10 for a discussion of differences between ABB and GE control rod designs.

NRC Question A2

What are the differences between the GE methods of References 1-4 and the methods of RPA-89-112?

ABB Response to Question A2

The ABB methodology is described in detail in Part II. This methodology utilizes state-of-the-art methods based on the RAMONA-3B code and a systematic approach based on well-established sensitivities and application-specific calculations to identify and evaluate the consequences of a worst-case postulated CRDA. To the extent that the results in Parts II, III, and IV can be compared with those in References 1 through 4 of Part III, the results of the two methodologies are considered to be consistent.

NRC Question A3

Does PHOENIX use a pre-ENDF/B-V value for β and, if not, justify the value used?

ABB Response to Question A3

The delayed neutron fractions in the cross section library used for the calculations in Parts II, III, and IV were taken from [Proprietary Information Deleted]

The sensitivity of the peak fuel enthalpy following a postulated CRDA to delayed neutron fraction is discussed in Section 4.5 of Part II. This discussion indicates that the severity of the CRDA is [Proprietary Information Deleted]

NRC Question A4

What are the differences between RAMONA-3B and RAMONA-3B-SCP2, and what is their effect on the modeling, benchmarking and analysis of the rod drop accident (RDA)?

ABB Response to Question A4

ABB utilizes the Scandpower version of RAMONA-3B as explained in Section 4.3.1 of Part II. This version of RAMONA-3B, referred to as the "Scandpower version of RAMONA-3B" in Part II and RAMONA-3B-SCP2 in Parts III and IV, includes the features in BNL Level 10 and can be considered to be upgraded to BNL version "Level 10". The most important extensions relative to the version described in Reference 6 can be summarized as follows:

(1) The nuclear cross-section data representation is made compatible with Scandpower's static 3-D core analysis methods

(FMS) and the ABB CORE MASTER system which includes POLCA.

(2) The option to input a non-equilibrium xenon distribution which provides the capability to initiate the transient from a non-equilibrium xenon state.

(3) The option to treat effective delayed neutron fractions as nodal variables as a function of fuel design and burnup has been installed.

(4) The nodal coupling method dealing with the thermal flux diffusion has been upgraded to that of the static three-dimensional nodal simulator, PRESTO.

(5) The modeling of reverse flow conditions has been improved.

(6) The time integration of the hydraulics has been improved to optionally allow for higher order explicit methods as well as implicit integration of some of the equations.

A major effect on modeling of these improvements is [Proprietary Information Deleted] The Scandpower version of RAMONA-3B used by ABB is considered to represent a substantial improvement relative to the code version discussed in Reference 6. Therefore, while ABB has not performed benchmark calculations for the code version described in Reference 6, it is expected that such a benchmark [Proprietary Information Deleted]

NRC Question A5

How is the time dependence of the local rod-to-bundle power peaking factor accounted for?

ABB Response to Question A5

As discussed in Section 4.3 of Part II, at the state point for which the CRDA is to be simulated, [Proprietary Information Deleted]

NRC Question A6

Provide References 10, 19, 30, 31, 35, 36, and 42.

ABB Response to Question A6

Copies of References 19, 30, 31, and 35 are included with this transmittal.

Reference 10 of Part III is RPA 89-053 for which responses to NRC questions are provided in Appendix B.

Reference 36 is an Institute of Energy Technology Report documenting an early application of RAMONA-3 to control rod insertion. The report conclusions are that uncertainties in the fuel burnup for the test conditions made the comparison with data difficult, however reasonable agreement was obtained for the full scram test. For the single rod scrams, the simulations showed best agreement with experimental data obtained in positions close to the scrammed channels. The work and report are property of the Institute of Energy Technology and the document describes early benchmark work with RAMONA-3. In light of the information provided in this document, the contents of Reference 36 does not contain any more relevant information. The response to Question A9 explains the RAMONA versions used in the Reference 36 and other benchmarks efforts.

Reference 42 was submitted to the NRC in August of 1987, and a revised version is scheduled for submittal by ABB in November of 1994. This document has been provided to the NRC to give much more detailed and updated description of the CRDA application methodology than that provided in Reference 42 of Part III. Therefore, Part II should be utilized for an explanation of the application methodology.

NRC Question A7

Describe the qualification of the Version-SCP2 thermal diffusion option that has been performed for transients like the RDA in which strong spatial peaking occurs.

ABB Response to Question A7

As noted in the response to Question A4, the nodal coupling method dealing with the thermal flux diffusion has been upgraded to that of the static three-dimensional nodal simulator, PRESTO. Therefore, as discussed in Section 5.3.1 of Part II, the time-dependent analytical models for performing the coupled neutron flux-coolant void calculations in the version of RAMONA-3B utilized by ABB, the Scandpower version of RAMONA-3B, are equivalent to those in the PRESTO three dimensional core simulator (References 7 and 8) in the steady-state. Consequently, the capability of PRESTO to predict steady-state power distributions provides a good indication of the capability of the Scandpower version of RAMONA-3B to predict power distributions. The PRESTO code has been reviewed and accepted for steady-state neutronics applications by the NRC. As shown in Section 5.3.1 of Part II and References 7 and 8, PRESTO, and therefore, RAMONA-3B, provide a state-of-the-art capability to provide reliable power shapes under hot and cold conditions.

NRC Question A8

Describe the results of the Muhleberg and Brunswick-1 RAMONA-3B turbine trip test comparisons. How do the methods and models used in these calculations compare to the ABB-Atom RDA licensing analyses?

ABB Response to Question A8

The Muhleberg and Brunswick-1 turbine trip test comparisons were performed to confirm the capability of RAMONA-3B to predict power excursions caused by a core void collapse initiated by a turbine trip. Unfortunately, formal documentation of these comparisons is not readily available. Reference to applications of RAMONA-3B to occurrences other than the CRDA in Parts III and IV were intended to indicate the general reliability of the code to predict the response to a broad range of different occurrences and the broad experience of ABB and Scandpower in applying the code.

NRC Question A9

Were the Scandpower Peach Bottom-2, Muhleberg, Brunswick-1 and Gundremmingen A (KRB) comparisons made with Version-SCP2 and, if not, discuss the applicability of these models/comparisons as qualification for the RPA-89-112 RDA methodology?

ABB Response to Question A9

The Scandpower version of RAMONA-3B used for the calculations in Parts II, III, and IV is the same as that used for the Peach Bottom-2, and Brunswick-1 comparisons. The Gundremmingen A (KRB) comparison was performed with a previous code version. Reference to these applications of RAMONA-3B were intended to indicate the general reliability of the code to predict the response to a broad range of different occurrences and the broad experience in applying the code. The simulations of the SPERT-IIIE power excursion tests in Part II is more applicable to the CRDA methodology.

NRC Question A10

Discuss the results and applicability of the ABB-Atom and Scandpower RAMONA-3B RDA "actual plant applications" (referenced on p. 27) as qualification for the RDA licensing analyses.

ABB Response to Question A10

Reference to applications of RAMONA-3B to occurrences other than the CRDA in Parts III and IV were intended to indicate the general reliability of the code to predict the response to a broad range of different occurrences and the broad experience of ABB and Scandpower in applying the code.

Specific qualification of the ABB CRDA methodology is provided in Part II. The application to CRDAs referred to in Parts III and IV generally involved analyses to support plant operation rather than benchmark calculations. Reference was made to these applications to indicate the general usefulness and reliability of the Scandpower version of RAMONA-3B for this accident. Qualification of the ABB CRDA methodology RAMONA-3B is addressed in detail in Part II.

NRC Question A11

While applications of BNL and Scandpower are discussed in RPA-89-112, what qualification comparisons have been performed by ABB-Atom with the licensing Version-SCP2 of RAMONA-3B using the PHOENIX/POLCA/POLGEN cross section calculation.

ABB Response to Question A11

Part II contains a detailed discussion of the qualification basis for the ABB CRDA methodology using the Scandpower version of RAMONA-3B in conjunction with the PHOENIX/POLCA/POLGEN cross section calculation. Specifically, Part II contains information to verify that the ABB CRDA methodology is sufficiently accurate and conservative for licensing applications. The verification is provided by [Proprietary Information Deleted]

NRC Question A12

Discuss the quality assurance program and application under which RAMONA-3B-SCP2 was developed and qualified.

ABB Response to Question A12

The Scandpower version of RAMONA-3B was developed by Scandpower. ABB and Scandpower have jointly qualified the code as discussed in Part II. [Proprietary Information Deleted]

NRC Question A13

Provide the details of the calculations and results of the RAMONA-3B-SCP2 comparisons to the SPERT-III E-Core transient measurements.

ABB Response to Question A13

Simulations of six of the SPERT-IIIE power excursion tests using the ABB CRDA methodology are provided in Section 5.3.2 of Part II.

NRC Question A14

In representing an off-center control rod with a central rod, how are the local peaking and feedback at the off-center location preserved in the center rod drop calculation?

ABB Response to Question A14

As discussed in Part II, the dropped control rod is [Proprietary Information Deleted]

NRC Question A15

Are the same procedures used to model the core loading (number of fuel types, axial fuel zones, etc.) in RAMONA-3B-SCP2 as are used in the PHOENIX/POLCA/POLGEN modeling? If not, discuss the effect these differences will have on the RDA modeling, analysis and benchmarking.

ABB Response to Question A15

[Proprietary Information Deleted]

NRC Question A16

How is the coupling between the void and fuel temperature dependence accounted for in the RAMONA-3B-SCP2 cross section representation?

ABB Response to Question A16

As discussed in Section 4 of Part II, at the core conditions for which the CRDA is to be simulated [Proprietary Information Deleted]

In addition, fast absorption, removal, and fission cross sections are assumed to vary as the square root of the fuel temperature. Specifically, the dependence on coolant density and the coupling between the coolant density and fuel temperature in the Scandpower formulation of these cross sections is expressed in the form:

$a + b\rho + c\rho^2 + (d + e\rho + f\rho^2)(\sqrt{T_f} - \sqrt{T_{fo}})$, where

ρ = coolant density

T_f = fuel temperature

T_{fo} = reference fuel temperature, and

a, b, c, d, e, and f are constants at a given burnup and void history.

The moderator density is updated for the power calculation at each time step, thereby accounting for moderator temperature feedback. The fast group cross sections are updated by the current fuel temperature at each time step to account for Doppler feedback.

NRC Question A17

As validation of the POLGEN/RAMONA-3B-SCP2 cross section representation, provide comparisons of RAMONA-3B-SCP2 and POLCA calculated power distributions and feedback reactivities for typical RDA statepoints.

ABB Response to Question A17

As discussed in Section 4.5 of Part II, this type of validation is part of the ABB methodology. [Proprietary Information Deleted]

The type of global check suggested by the question is also performed. [Proprietary Information Deleted]

As indicated in the sensitivity studies in Section 4 of Part II and the response to Question C17, [Proprietary Information Deleted]

NRC Question A18

Provide an estimate of the uncertainty introduced into the feedback coefficient by combining the moderator temperature and moderator void dependence, and the impact on the RDA.

ABB Response to Question A18

The moderator void and temperature affects the impact of the dropped control rod through the feedback which these coolant properties have on the core power and power distribution as well as the thermal conductivity from the fuel rod to the coolant. It is our judgment that combining the moderator temperature and moderator void dependence in the cross section description does not introduce any significant uncertainties into the predicted feedback from these coolant properties.

[Proprietary Information Deleted]

The thermal-hydraulic model in RAMONA-3B treats both the moderator void and moderator temperature explicitly. The impact which the moderator void and temperature have on the heat conductance from the fuel rod to the coolant is treated in RAMONA-3B by utilizing convective heat transfer coefficients in the solution of the coupled heat conduction differential equations which depend on the fluid properties in the coolant.

NRC Question A19

Recognizing that a larger transient increase in fuel temperature results in an increased Doppler feedback, how is a conservative gap conductance determined for the RDA? How is the fuel design dependence and fuel burnup dependent gap closure and fission gas release accounted for in the determination of the gap conductance?

ABB Response to Question A19

Gap conductance is modeled as a quadratic function of average fuel temperature in the Scandpower version of RAMONA-3B used by ABB. [Proprietary Information Deleted]

NRC Question A20

If moderator feedback is to be included in licensing analyses of the RDA, recognizing the substantial degree of uncertainty in the magnitude and timing of the moderator voiding under the highly transient conditions of the RDA, provide detailed justification and model qualification for the relaxation of this conservatism.

ABB Response to Question A20

In licensing analyses, the influence of moderator void feedback on the calculated peak fuel enthalpy will be treated in a conservative fashion. [Proprietary Information Deleted]

Steady-State Heat Transfer and Vapor Generation Modeling

As discussed in Section 5.3.1 of Part II, the time-dependent hydraulic models in the Scandpower version of RAMONA-3 are equivalent to those in the PRESTO three-dimensional core simulator under steady-state conditions. Therefore, the capability of PRESTO to predict steady-state hydraulic conditions provides an indication of the capability of the Scandpower version of RAMONA-3 to reliably predict hydraulic conditions during a transient. Detailed descriptions and qualification of the PRESTO hydraulic models were submitted to the NRC in References 8 and 9. It is demonstrated in these documents that the hydraulic modeling of the BWR two-phase system under steady-state conditions in PRESTO is a state-of-the-art representation. [Proprietary Information Deleted]

Rapid Transient Heat Transfer Modeling

[Proprietary Information Deleted]

Rapid Transient Heat Transfer Impact on CRDA Calculations

[Proprietary Information Deleted]

Rapid-Transient Vapor Generation Modeling

[Proprietary Information Deleted]

Rapid Transient Vapor Generation Impact on CRDA Calculations

[Proprietary Information Deleted]

Uncertainty Treatment

[Proprietary Information Deleted]

NRC Question A21

How are conservative initial conditions (cycle burnup, power level, inlet subcooling, etc.) and modeling parameters (Doppler coefficient, delayed neutron fraction, scram worth, etc.) selected for RDA licensing analyses?

ABB Response to Question A21

The selection of initial conditions is addressed in some detail in Part II. [Proprietary Information Deleted]

NRC Question A22

Do any of the sensitivities provided in Section-6 change significantly for Version-SCP2?

ABB Response to Question A22

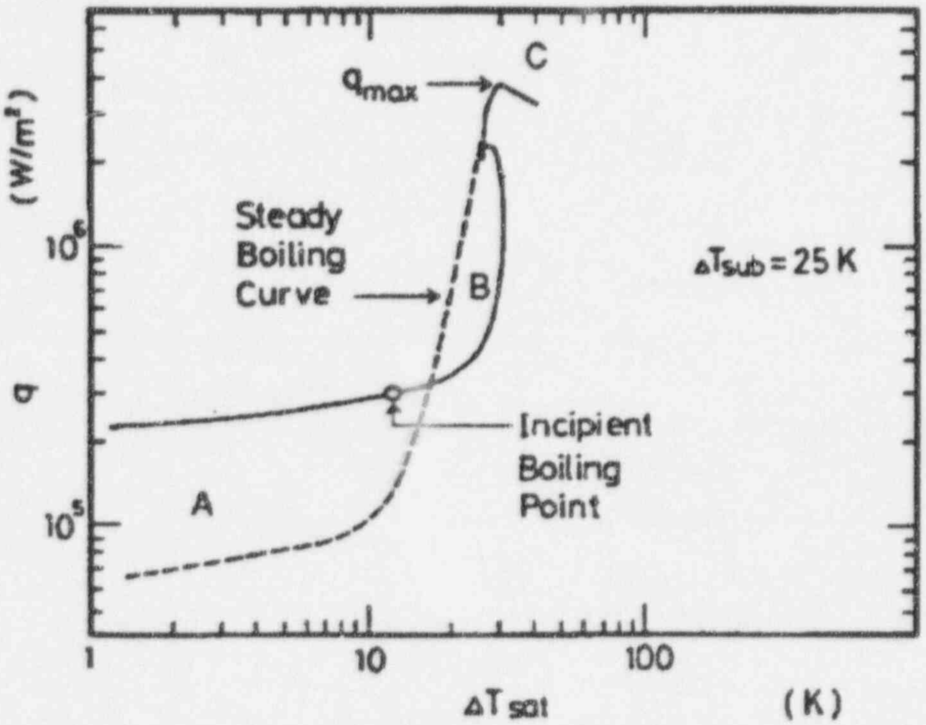
The designation "RAMONA-3B SCP2" in Part III refers to the Scandpower version of RAMONA-3B which has been upgraded relative to the code version described in Reference 6. Section 4.3.1 of Part III lists some of the more important upgrade features. All of the ABB calculations reported in Part III, specifically the results in Sections 5 and 6, were performed with this upgraded Scandpower version of the RAMONA-3B. Therefore, the sensitivity calculations in Section 6 as well as the base case calculations in Section 5 were performed with "RAMONA-3B SCP2".

TABLE A17-1

Proprietary Information Deleted

FIGURE A3-1 THROUGH FIGURE A19-1

Proprietary Information Deleted



Legend: $\Delta T_{sat} = T_{wall} - T_{sat}$
$\Delta T_{sub} = T_{sat} - T_{initial}$
$q(t) = q_o \exp(t / 0.049 \text{ sec})$

Figure A20-1 Measured Transient and Steady-State Boiling Curves (adapted from Reference 18)

FIGURE A20-2

Proprietary Information Deleted

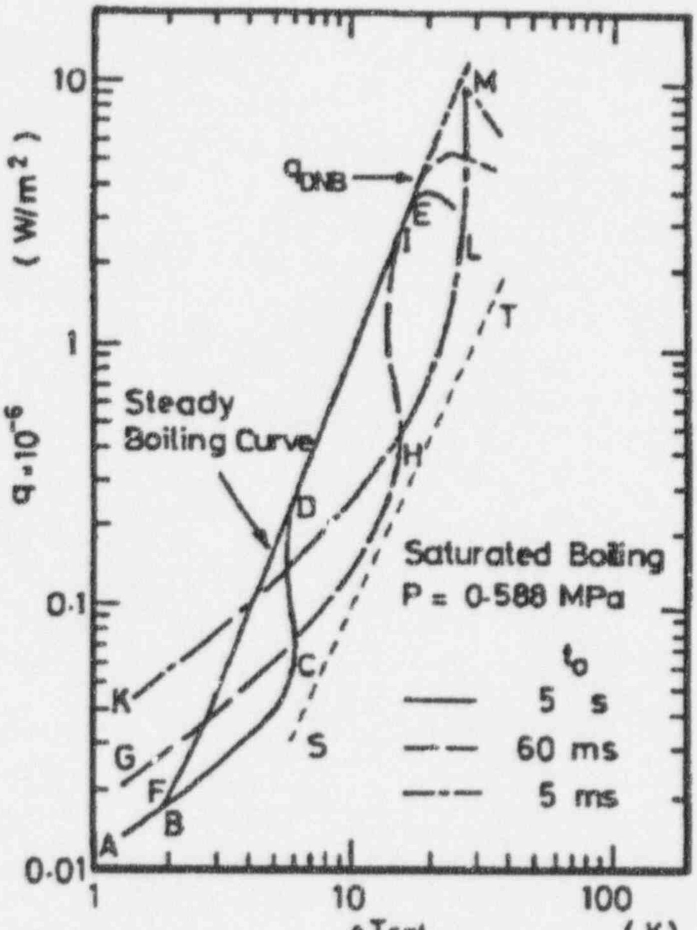
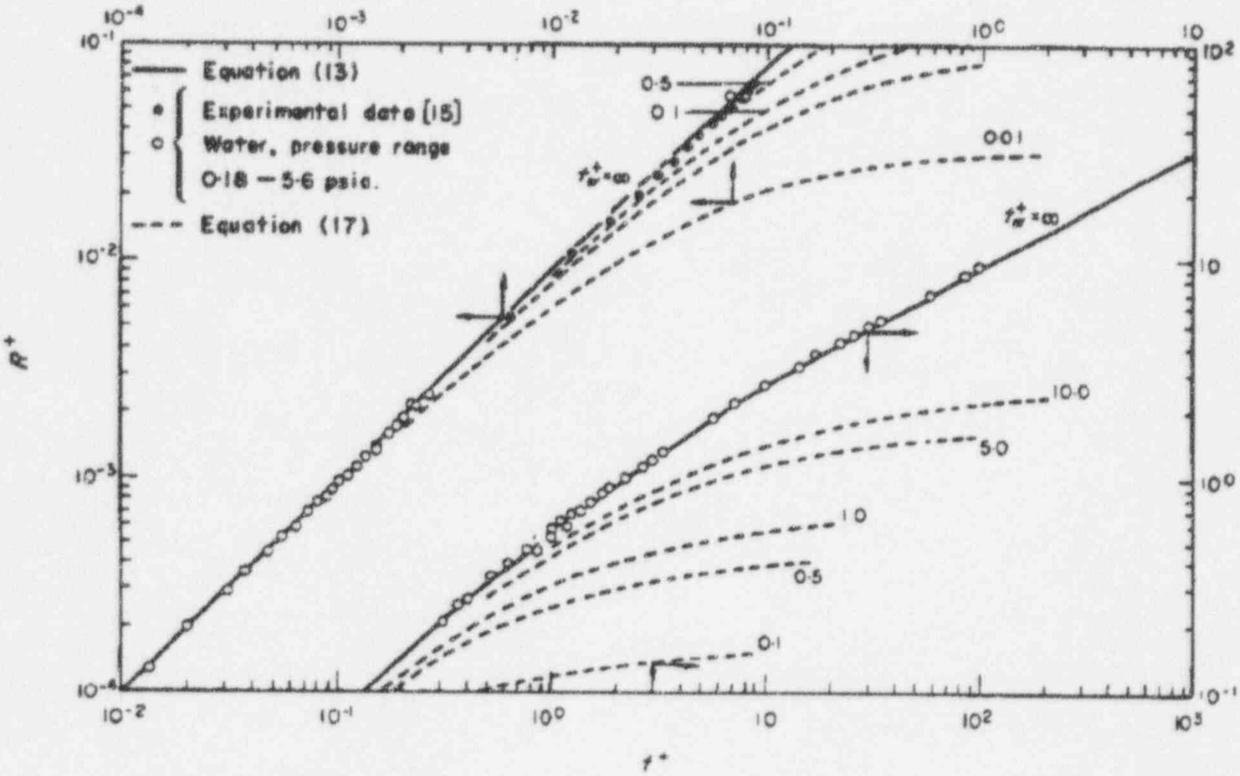


Figure A20-3 Transient Boiling Curves for Various Exponentially Increasing Heat Fluxes, $q=q_0 \exp(t/t_0)$ (adapted from Reference 19)

FIGURE A20-4 THROUGH FIGURE A20-7

Proprietary Information Deleted



Legend: $t^+ = \dfrac{A^2\,t}{B^2}$

$$t_w{}^+ = \frac{A^2\,t_w}{B^2}$$

$$R^+ = \frac{A\,R}{B^2}$$

where

$$A = \left(b\frac{h_{fg}\,\rho_v\,\Delta T}{\rho_l\,T_{sat}}\right)^{0.5}$$

$$B = \left(\frac{12\,\alpha_l}{\pi}\right)^{0.5}\left(\frac{\Delta T\,c_l\,\rho_l}{h_{fg}\,\rho_v}\right)$$

$$b = \begin{cases} 2/3 & \text{for bubble in an infinite medium} \\ \pi/7 & \text{for a spherical bubble growing attached to a surface} \end{cases}$$

Figure A20-8 Bubble Growth in Uniformly Superheated Liquid ($t_w{}^+=\infty$) and in Non-Uniform Temperature Fields (from Reference 20)

FIGURE A20-9 THROUGH FIGURE A20-10

Proprietary Information Deleted

A.3 Additional Information for Question A19: Treatment of Bounding Values and Uncertainties

This section provides a clarification and expansion of the treatment of bounding values and uncertainties relative to the discussion in Part II.

[Proprietary Information Deleted]

TABLE A.3.1

Proprietary Information Deleted

FIGURE A.3.1

Proprietary Information Deleted

APPENDIX B: RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION ON RPA 89-053-P

B.1 Introduction

This appendix contains responses to the NRC Request for Additional Information regarding Reference 2 which was transmitted to ABB by the NRC letter identified in Reference 3. For convenience, all references used in this appendix are included in Appendix D.

Reference 1 provided a summary of the ABB methodology for analysis of the Control Rod Drop Accident (CRDA) using RAMONA-3B as well as a sample analysis for a 532-assembly BWR. The responses to requests for additional information regarding Reference 1 are included in Appendix A.

Reference 2 provided an additional sample CRDA analysis illustrating the impact of a postulated CRDA in a plant equipped with high worth control rods. The responses to requests for additional information regarding Reference 2 are included in this appendix.

Reference 4 was submitted in 1993 to clarify and summarize the ABB CRDA methodology as well as to provide further information supporting the qualification of the ABB CRDA analysis methodology using RAMONA-3B. The responses to requests for additional information regarding Reference 4 are included in Appendix C.

B.2 Questions and Responses

NRC Question B1

What is the basis for assuming the worth of the HWCR is 15% greater than a standard control rod? Does this 15% difference provide a bound for all variations in core conditions (rod insertion, moderator density, etc.)?

ABB Response to Question B1

The 15% difference in total reactivity worth between the high worth control rod and the standard rod was selected as typical. The purpose of the analysis in Part IV was to provide an indication of the impact on a CRDA for a U.S. reactor containing high worth control rods relative to one containing standard control rods. [Proprietary Information Deleted]

Performance of licensing basis calculations is more clearly described in Part II and clarified in this document.

NRC Question B2

Provide quantitative justification for the spatial nodalization of the fuel rod heat transfer equations. How is the pellet/clad gap described?

ABB Response to Question B2

As discussed in Reference 6, the thermal energy distribution and heat conduction from the pellet to the coolant is performed by solving standard coupled, time-dependent radial heat conduction differential equations for the pellet, gap and cladding. The pellet and clad are nodalized into concentric rings. [Proprietary Information Deleted]

NRC Question B3

Provide a quantitative estimate of the uncertainty introduced by using the fuel Type-2 fuel temperature, moderator density and control rod insertion dependence for all fuel types. Does fuel Type-2 have the most conservative feedbacks and control dependence?

ABB Response to Question B3

Depletion calculations providing nuclear data (e.g. cross sections and local peaking factors) for the same void and void history were performed for each fuel type. The differentials in the nuclear data caused by changing the coolant void, fuel temperature, and control state calculated for Fuel Type 2 were applied to the appropriate mainline depletion results for each of the other fuel types. The U-235 enrichments for Fuel Types 1 and 3 are the same as for Fuel Type 2. The only difference in the fuel types was in the Gd_2O_3 design, and the calculations were performed sufficiently late in the cycle that this difference is considered to be minor. Therefore, it is judged that conclusions regarding the relative impact of a core containing high worth control rods relative to one containing standard control rods would not be altered by this approximation. Fuel Type 4 is sufficiently unimportant to the determination of the peak fuel enthalpy that this approximation will not significantly impact the calculated peak fuel enthalpy.

It should be noted that the analyses in Part IV were performed as a sensitivity to evaluate the impact of installing high worth control rods. The ABB methodology for CRDA licensing applications is summarized in Part II. [Proprietary Information Deleted]

<u>NRC Question B4</u>

What flux shape is used to determine the importance-weighted delayed neutron fraction? Is this conservative for the RDA?

<u>ABB Response to Question B4</u>

As noted in Part II, beta-effective is provided as a function of burnup for each fuel type on a nodal basis in the Scandpower version of RAMONA-3B used by ABB. Optionally, it can be provided on a core-average basis. [Proprietary Information Deleted] The average importance weighted core value of beta referred to in Section 5 of Part IV is a core average value edited by the code for information and is not used in the calculations. The value quoted corresponds to the core power distribution reflected by the average axial power shape shown in Figure 6.3 of Part IV.

[Proprietary Information Deleted] The current delayed neutron fractions used by ABB are also discussed in the response to Question A3.

<u>NRC Question B5</u>

Recognizing that the Doppler coefficient decreases at higher temperatures and the RDA is sensitive to the initial power level, how will ABB-Atom insure that the hot-zero-power and low-power events are bounded by the cold-zero-power RDA?

ABB Response to Question B5

[Proprietary Information Deleted]

Please also see the response to Question C17.

NRC Question B6

How are inoperable rods accounted for in the selection of the maximum worth rod? How will the limiting control rod pattern be selected in the RDA licensing analysis?

ABB Response to Question B6

[Proprietary Information Deleted]

Please also see the response to Question C2.

NRC Question B7

What is the effect of assuming a "linear" rod insertion on the peak fuel enthalpy?

ABB Response to Question B7

[Proprietary Information Deleted]

NRC Question B8

Are the conditions of Section-6, at which the rod worth is calculated to be 0.01174, the same as the conditions that were used to calculate the 0.012 rod worth of Reference-5? If not, how do these rod worths compare at identical conditions?

ABB Response to Question B8

The conditions in Section 6 of Part IV are not the same as those in Reference 5 of Part IV. For example, the analyses were performed for different cores. Reference 5 of Part IV identified this reactivity worth as the maximum incremental dropped rod worth when the maximum number of rods are bypassed in the particular 748-assembly core for which the analysis was performed. The reference to Reference 5 of Part IV was intended only to put in perspective the value of 0.012 as a relatively high reactivity worth which would not be expected to be encountered under normal conditions.

The specific core evaluated for the CRDA in Reference 5 of Part IV has not been evaluated by ABB. [Proprietary Information Deleted]

Part II provides further discussion of conclusions based on the ABB methodology relative to those of previous work.

NRC Question B9

The extrapolation of the fuel enthalpy in the RDA with feedback, from 2 seconds to the time at which the peak occurs (typically 5 seconds), based on the time dependence of the RDA without feedback is highly uncertain. Therefore, if the licensing calculations will be performed with feedback, provide the RAMONA-3B-SCP2 calculation beyond the time of the peak fuel enthalpy.

ABB Response to Question B9

[Proprietary Information Deleted]

NRC Question B10

Provide a quantitative estimate of the uncertainty introduced by the limited number of thermal-hydraulic channels used to represent the core thermal-hydraulics (e.g., in Figure 5.3).

ABB Response to Question B10

[Proprietary Information Deleted]

NRC Question B11

Describe the benchmarking and testing performed to validate the PHOENIX/POLCA and RAMONA-3B-SCP2 neutronics schemes for application to HWCRs.

ABB Response to Question B11

As discussed in Section 5.3.1 of Part II, the time-dependent analytical models for performing the coupled neutron flux-coolant void calculations in the version of RAMONA-3B utilized by ABB, the Scandpower version of RAMONA-3B, are equivalent to those in the PRESTO three dimensional core simulator (Reference 7 and 8) in the steady-state. Therefore, the capability of PRESTO to predict control rod worths and power distributions provides a good indication of the capability of the Scandpower version of RAMONA-3B to predict control rod worths and power distributions. The PRESTO code has been reviewed and accepted for steady-state neutronics applications by the

NRC. Qualification of PRESTO is addressed in depth in References 7 and 8. [Proprietary Information Deleted]

NRC Question B12

Has the axial expansion of the fuel pellet been accounted for in the determination of the Doppler coefficient?

ABB Response to Question B12

[Proprietary Information Deleted]

Section 5.3.1 of Part II contains a comparison of Doppler coefficients calculated with PHOENIX with the MCNP-3A benchmark results reported in Reference 16. Some clarification and amplification of this comparison may be useful.

The Doppler coefficient calculations in Reference 16 were performed for an increase in fuel temperature of 300 °C in an infinite array of pin cells. Changes in fuel pellet and homogenized cladding-gap composition number densities were assumed to be associated with this change in temperature. However, no changes in pellet and cladding dimensions, or pin cell pitch, were assumed to be associated with the temperature increase. [Proprietary Information Deleted]

Reference 17 contains comparisons of the predictions of other deterministic codes currently used in the industry with the MCNP-3A results in Reference 16. Specifically, predictions of the ONEDANT discrete ordinate code package with an ENDFB-V library, as well as those of the CELL-2 code in the EPRI-PRESS reactor physics code package and the DSN and PERSEUS neutron transport methods in WIMS-AECL with the full 89-group neutron energy structure in the ENDF/B-V data Library Version 89-03-29, were compared with the MCNP-3A results in Reference 16. [Proprietary Information Deleted]

FIGURE B12-1

Proprietary Information Deleted

APPENDIX C: RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION ON CENPD-284-P

C.1 Introduction

This appendix contains responses to the NRC Request for Additional Information regarding Reference 4 which was transmitted to ABB by the NRC letter identified in Reference 3. For convenience, all references used in this appendix are included in Appendix D.

Reference 1 provided a summary of the ABB methodology for analysis of the Control Rod Drop Accident (CRDA) using RAMONA-3B as well as a sample analysis for a 532-assembly BWR. The responses to requests for additional information regarding Reference 1 are included in Appendix A.

Reference 2 provided an additional sample CRDA analysis illustrating the impact of a postulated CRDA in a plant equipped with high worth control rods. The responses to requests for additional information regarding Reference 2 are included in Appendix B.

Reference 4 was submitted in 1993 to clarify and summarize the ABB CRDA methodology as well as to provide further information supporting the qualification of the ABB CRDA analysis methodology using RAMONA-3B. The responses to requests for additional information regarding Reference 4 are included in this appendix.

C.2 Questions and Responses

NRC Question C1

Recognizing the substantial degree of uncertainty in the magnitude and timing of the moderator voiding under the highly transient conditions of the RDA, provide detailed justification and model qualification for the use of this previously unapproved additional transient feedback.

ABB Response to Question C1

Please see the response to Question A20.

NRC Question C2

Discuss how the single equipment malfunction and operator error are included in the determination of the maximum worth rod. For plants using the rod worth minimizer, the rod sequence control system or the rod pattern control system, how are bypassed rods selected and accommodated?

ABB Response to Question C2

The existing plant-specific worst-case credible single equipment malfunction and operator error allowed by the design and administrative procedures are utilized by ABB in the CRDA evaluation. The existing worst-case situation is defined by the plant reactor control system, the plant technical specifications, and current licensing basis CRDA evaluation. Since the limiting assumptions for equipment malfunction and operator error are not fuel-type specific, substantial revisions to existing assumptions regarding single equipment malfunction and operator error are not anticipated to be required for most applications when ABB reload fuel is installed in a particular plant. [Proprietary Information Deleted]

As noted in the question, the assumed worst-case single equipment malfunction and operator error will depend on the control rod withdrawal system utilized for a given plant. The example provided in Section 4 of Part II is for a Banked Position Withdrawal Sequence (BPWS) plant. Technical Specifications for BPWS plants typically require that no more than 8 rods be inoperable and that each inoperable control rod be separated from all other inoperable rods by at least two control cells in all directions. [Proprietary Information Deleted]

NRC Question C3

How is it assured that CRDAs like the case in Figure 4.4.3, with a ~990 pcm rod worth and a nodal peaking of ~64, are not limiting?

ABB Response to Question C3

[Proprietary Information Deleted]

NRC Question C4

In Step-6 of the cycle-specific evaluation, how will conservative values for the scram worth, velocity and delay be determined?

ABB Response to Question C4

[Proprietary Information Deleted]

NRC Question C5

Describe and justify the important RAMONA-3B core and systems modeling assumptions made in the CRDA licensing analyses (neutronic and thermal-hydraulic channels, fuel rod nodalization, etc.)

ABB Response to Question C5

[Proprietary Information Deleted]

The fuel rod nodalization in the RAMONA-3B calculations is discussed in the response to Question B2.

NRC Question C6

Describe how the fuel burnup and void history dependence is included in the nodal neutronics data.

ABB Response to Question C6

[Proprietary Information Deleted]

NRC Question C7

Does PHOENIX use a pre-ENDF/B-V value for β and, if so, justify the value used?

ABB Response to Question C7

Please see the Response to Question A3.

NRC Question C8

Discuss the modeling of reverse flow and the effect on the CRDA licensing analyses.

ABB Response to Question C8

[Proprietary Information Deleted]

NRC Question C9

For what specific fuel/core designs is the CRDA methodology intended? Has the PHOENIX/FOBUS/PHOEBE/PHIPO/POLCA code system been approved for these applications? Has the STAV fuel performance code system been approved for these applications?

ABB Response to Question C9

[Proprietary Information Deleted]

NRC Question C10

What are the significant differences between the ABB-CE and GE fuel and control rod designs, and what is their impact on the CRDA?

ABB Response to Question C10

A detailed description of the ABB SVEA-96 assembly being utilized in the U.S. can be found in Reference 11. As discussed in Part II, nuclear data for RAMONA-3B will be calculated with a lattice code and three-dimensional core simulator accepted by the NRC for licensing applications. [Proprietary Information Deleted]

A description of the ABB control rod and a comparison with GE blades are provided in Section C.3. [Proprietary Information Deleted]

NRC Question C11

The reactivity insertion rate and resulting transient peak fuel enthalpy are increased if the reactivity is inserted over a reduced axial span (assuming a constant rod drop speed). In the case that the control rod drops to a rod drive located above the bottom of the core, is the total reactivity in the RAMONA-3B calculation (which may have been *precalculated) inserted over this reduced axial span? If not, justify the method used.*

ABB Response to Question C11

[Proprietary Information Deleted]

NRC Question C12

Discuss how a maximum rod drop speed less than 3.1 ft/sec is justified.

ABB Response to Question C12

[Proprietary Information Deleted]

NRC Question C13

Describe the difference between a power scram and a flux scram, and provide an estimate of the conservatism included by using the power scram rather than a flux scram.

ABB Response to Question C13

The "power scram" assumes that the process to initiate a scram is started when the core power reaches a certain level. The "flux scram" assumes that the process to initiate a scram is started when the core flux reaches a certain level. [Proprietary Information Deleted]

NRC Question C14

When the RAMONA-3B model is expanded to full core geometry, are reductions made in the number of neutronic and thermal-hydraulic channels (per octant)? If so, how have these approximations been validated?

ABB Response to Question C14

[Proprietary Information Deleted]

NRC Question C15

Can the control rod insert additional reactivity by dropping past the axial location of the drive mechanism?

ABB Response to Question C15

[Proprietary Information Deleted]

NRC Question C16

How will conservative values be determined for the Doppler coefficient, gap conductance and thermal conductivity for a given cycle statepoint in the licensing analyses?

ABB Response to Question C16

Please see Section A.3.

NRC Question C17

The 5% and 10% power cases of Figure 4.5.19 indicate a substantial increase in the CRDA peak fuel enthalpy with increasing power. In order to justify the licensing analysis at low power, provide an evaluation of the effect of the reduction in Doppler feedback reactivity and the reduction in CRDA rod worth that occurs at higher powers.

ABB Response to Question C17

[Proprietary Information Deleted]

NRC Question C18

While it is recognized that the minimum service limit pressure is at ~20°C, can a CRDA initiated from low subcooling conditions result in a closer approach to the service limit than the assumed CRDA from 20°C?

ABB Response to Question C18

[Proprietary Information Deleted]

NRC Question C19

In Step-1 of the cycle-specific evaluation, what specific criteria are used to conclude from existing CRDA analyses that a dynamic analysis is not necessary?

ABB Response to Question C19

[Proprietary Information Deleted]

NRC Question C20

In Step-4 of the cycle-specific evaluation, in cases where all parameters of a previous dynamic analyses do not bound the cycle-specific CRDA, *will a cycle-specific CRDA be performed? If not, describe how a bounding CRDA will be determined.*

ABB Response to Question C20

[Proprietary Information Deleted]

NRC Question C21

Does moderator voiding have a significant effect on the CRDA calculated for the "analysis condition" statepoint of Figure 4.7.1?

ABB Response to Question C21

[Proprietary Information Deleted]

NRC Question C22

Provide justification for the conservatisms that are relaxed in the "more realistic" calculations that will be performed when the design criteria are not satisfied by the bounding CRDA.

ABB Response to Question C22

[Proprietary Information Deleted]

NRC Question C23

As validation for the RAMONA-3B core analysis capability, provide comparisons of the RAMONA-3B and POLCA prediction of power distribution, bank and rod worths, and Doppler defect.

ABB Response to Question C23

Please see the response to Question A17.

NRC Question C24

What specific version of the RAMONA-3B code will be used in the CRDA licensing analyses?

ABB Response to Question C24

ABB utilizes the Scandpower version of RAMONA-3B. Significant differences relative to the version of RAMONA-3B described in Reference 6 are described in the response to Question A4.

[Proprietary Information Deleted]

TABLE C17-1 THROUGH TABLE C18-1

Proprietary Information Deleted

FIGURE C17-1 THROUGH FIGURE C17-2

Proprietary Information Deleted

C.3 Additional Information for Question C10: ABB Control Rods

This section provides a description of the ABB control rod and a comparison with GE blades relative to the discussion in the response to Question C10.

ABB Control Rod Description

The ABB control rod design consists of neutron-absorbing materials, B_4C and Hafnium, contained in horizontally drilled holes in solid sheets of high grade 316L stainless steel. This results in a design where the blades act both as structural and containment elements. Thus, the design is completely free from crevices and other cavities, which is optimum from the corrosion viewpoint. In addition, this design has one-third the surface area of GE original equipment control rods, which yields a benefit in terms of cobalt activation.

Spacing within the gap between fuel assemblies is maintained by the use of Inconel X-750 buttons. These wear resistant buttons serve as the contact points between the control rod and adjacent fuel channels during operation. When used in a GE designed BWR, the ABB control also has a velocity limiter identical to that used on the GE original equipment control rods.

[Proprietary Information Deleted]

A very detailed description of the ABB control rod design, adopted to the various GE BWR lattice types, is contained in Reference 14.

[Proprietary Information Deleted]

APPENDIX D: REFERENCES USED IN THE RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

1. ABB Report RPA 89-112, "ABB Atom Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors, The RAMONA-3B Computer Code," November, 1989.

2. ABB Report RPA 89-053, "ABB Atom High Worth Control Rods for US BWRs, Rod Drop Accident Analysis," August, 1989.

3. NRC letter from Timothy E. Collins (NRC) to Derek Ebeling-Koning (ABB), "Request for Additional Information for Topical Reports RPA-89-112 and RPA-89-053", May 19, 1994.

   NRC letter from Timothy E. Collins (NRC) to Derek Ebeling-Koning (ABB), "Request for Additional Information for ABB-CE Topical Report CENPD-284-P (TAC. NO. M88025)," August 5, 1994.

4. ABB Report CENPD-284-P, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," October 1993.

5. "ABB Atom Nuclear Design and Analysis Programs for Boiling Water Reactors Programs Description and Qualification," BR 91-402-P-A (proprietary), BR 91-403-NP-A (non proprietary), May 1991.

6. W. Wulff et. al., "A Description and Assessment of RAMONA-3B Mod. 0 Cycle 4: A Computer Code with Three Dimensional Neutron Kinetics for BWR System Transients," NUREG/CR-3664, 1984.

7. S. Borresen, L. Moberg, J. Rasmussen, "Methods PRESTO-B-A Three Dimensional BWR Core Simulation Code," NF-1583.03, US-NRC Topical Report Submitted by Carolina Power & Light Company, 1983.

8. K. E. Karcher, W. K. Cantrell, and D. W. Schroeder, "A Description and Validation of Steady-State Analysis Methods For Boiling Water Reactors," NF-1583.01, US-NRC Topical Report Submitted by Carolina Power & Light Company, 1983.

9. "Fuel Rod Design Methods for Boiling Water Reactors," ABB Report CENPD-285-P-A (proprietary), CENPD-285-NP-A (nonproprietary), July 1996.

10. C. J. Paone, R. C. Stirn, J. A. Woolley, "Rod Drop Accident Analysis for Large Boiling Water Reactors," GE Report NEDO-10527, March 1972.

11. "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors," ABB Report CENPD-287-P-A (proprietary), CENPD-287-NP-A (nonproprietary), July 1996.

12. "Quality Assurance Program, A Description of the CENO Nuclear Quality Assurance Program" ABB Report CENPD-210A, latest revision.

13. ANL-5800, Reactor Physics Constants, July, 1963.

14. "ABB BWR Generic Control Rod Design Methodology", CENPD-290-P (proprietary), CENPD-290-NP (nonproprietary), February, 1994.

15. ABB Report CENPD-284-P-RAI, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Response to Request for Additional Information," October 1994.

16. R. D. Mosteller, L. S. Eisenhart, R. C. Little, W. J. Eich. and J. Chao, "Benchmark Calculations for the Doppler Coefficient of Reactivity," Nuclear Science and Engineering, V. 107, p. 265-271, 1991.

17. I. Soares, W.F. Miller, R.T. Perry, "Analysis of ONEDANT Code Package for the Calculation of the Doppler Coefficient of Reactivity, Nov. 25, 1992.

18. A. Sakuri and M Shiotsu, "Transient Pool Boiling Heat Transfer, Part 1: Incipient Boiling Superheat," Journal of Heat Transfer, V. 99, pp. 547-553, 1977.

19. A. Sakuri and M Shiotsu, "Transient Pool Boiling Heat Transfer, Part 2: Boiling Heat Transfer and Burnout," Journal of Heat Transfer, V. 99, pp. 554-560, 1977.

20. B. B. Mikic, W. M. Rohsenow, and P. Griffith, "On Bubble Growth Rates," Int. J. Heat Mass Transfer, V. 13, pp. 657-666, 1970.

21. W. M. Rohsenow, "General Boiling," Handbook of Multiphase Systems, G. Hetsroni ed., Hemisphere Publishing Corp., p. 6-24, 1982.

22. C. Y. Han and P. Griffith, "The Mechanism of Heat Transfer in Nucleate Pool Boiling, Parts I and II," Int. J. Heat Mass Transfer, V. 9, p. 995-1046, 1965.

CENPD-284-NP-A, RPA 89-112-NP-A, and RPA 89-053-NP-A REPORT

Part III

Body of RPA 89-112-NP-A Report

Note that the responses to requests for additional information regarding this part of the report are included in Appendix A of Part II of this Report

ABB Atom

Report

Distribution			Ref. RPA 89-112	
NRC	From RPA	Date 89-11-17	Reg.	Page 1 (49)
	Author S. Andersson/R. Jadrny		Issued by	
	Examined by		Approved by	

Title

ABB Atom Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors. The RAMONA-3B Computer Code

Abstract

The ABB Atom BWR control rod drop accident analysis methodology is described.

The methodology described is appropriate for licensing evaluations of the control rod drop accident when appropriate initial conditions and the maximum expected individual control rod worths are used.

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1. INTRODUCTION

The control rod drop accident is analyzed for commercial Boiling Water Reactors as a design basis accident which is bounding for all postulated accidents involving additions of prompt reactivity. The method of analysis chosen must be capable of treating the effects of the rapidly changing power distribution which is caused by the rapid control rod movement.

This report describes the method of analysis that ABB Atom employs to evaluate the consequences of a rod drop in terms of specified fuel failure and fuel damage criteria.

General Electric Company (GE) has published several reports on the Control Rod Drop Accident (CRDA) applicable to GE BWR's (References 1 through 4). These reports contain a discussion of the mechanics of the accident, and parametric studies of the consequences in terms of a range of inputs which may vary in plant operations, such as control rod patterns and fuel type and exposure. Other publications (References 5 through 9) have examined various aspects of the analytical models which can be applied to this event.

The reports mentioned are generically applicable and cover a large number of input variables including different fuel types and core designs at different exposures and initial conditions. These reports also covered the considerations which must be made in determining the limiting cases of dropped rod reactivity shape function and scram reactivity shape function for a sufficiently large number of cases to be bounding for most fuel cycles in most US BWR plants.

The purpose of this report is to describe the methodology used by ABB Atom to evaluate the consequences of a CRDA in BWR's containing fuel or control rods of the ABB Atom design or other vendors' design. The methodology is illustrated with typical results including an assessment of the sensitivity to major analysis options and parameters. The impact of stronger control rods (from a reactivity point of view) is addressed by noting the difference that stronger control rods cause to those parameters that are significant to the outcome of the CRDA. This report is not intended to give bounding generic results for BWR cores containing high worth control rods. The methodology described in this report is applied and reported separately to provide such results. Results describing the consequences of introducing high worth control rods have been reported in Reference 10.

This report together with Reference 11 provide the methodology description and the qualification of using the methods for CRDA analysis.

2. DESCRIPTION OF ACCIDENT

The control rod drop accident assumes the breakage and disconnection of an inserted rod drive from the control blade. It is postulated that the drive mechanism is withdrawn while the control blade sticks in position and that the blade subsequently falls at its maximum velocity to the position of the drive. Since it is assumed that the event can occur in any reactor operating state consideration must be given to all the control rod configurations which can occur in normal operation and also those that can occur as a result of equipment malfunction or operator error (e.g., operator selection of an out of sequence control rod).

The accident is most severe when it is assumed to occur at low or zero power conditions when the control rod patterns required to establish criticality provide the highest values of incremental (dropped) single control rod worth. Further, the presence of voids in the core at any significant power level will decrease the consequences of the accident through the significant, negative, moderator density reactivity (void) coefficient in BWR's. For the same reason, conditions with large subcooling, e.g. start-up from cold shut down, which do not have any significant rapid moderator density feedback, usually provide the most severe initial states for the transient.

To determine the incremental rod worth and to determine the control rod configuration for modelling the accident, consideration must be given to the hardware employed for rod sequence control and the technical specifications on inoperable rods for the particular plant for which the analysis is being performed.

For Banked Position Withdrawal Sequence (BPWS) plants (see Reference 4) the Rod Worth Minimizer (RWM) is used below a specified power (typically 20 %) to enforce the rod withdrawal sequence. To limit the worth of the rod which could be dropped in a Group Notch Plant a group notch Rod Sequence Control System (RSCS) is installed to control the sequence of rod withdrawal.Further discussion of the control rod patterns which must be considered is given in Section 7.

The sequence of the accident is as follows:

(a) At some time a fully inserted rod becomes decoupled from its drive and sticks in the fully inserted position.

(b) During the startup sequence rod patterns are employed which are permitted by the constraints on rod movement imposed by Technical Specification or hardware. At some time, at critical reactor conditions, the rod pattern exists for which the decoupled rod has the maximum incremental worth from fully inserted to the position of its drive.

(c) The decoupled rod drops at the maximum velocity determined from experimental data (3.11 feet per second) to the position of its drive.

(d) The reactor goes on a positive period and the initial power burst is terminated by the fuel temperature reactivity feedback.

(e) The 120 % APRM power signal scram occurs (no credit is taken for the Intermediate Range Monitor or set down APRM scram).

(f) All withdrawn rods except the decoupled rod scram at technical specification rate.

(g) Scram terminates the accident.

3. EVALUATION CRITERIA

The criteria against which the consequences of the control rod drop accident are evaluated are based on meeting the requirements of General Design Criterion 28 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 sufficent damage to impair significantly the capacity to cool the core.

They are given in NUREG-0800, Standard Review Plan, as:

1. Reactivity excursions should not result in radially averaged fuel rod 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 Code.

3. The number of fuel rods predicted to reach assumed thresholds and associated parameters such as the amount of fuel reaching melting conditions will be an input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

4. METHOD OF EVALUATION OF CRDA

The dropped rod causes a large local increase in reactivity and a substantial change in the power distribution during the course of the accident. The method of analysis must represent this power shape change properly to account for feedback effects and to calculate the energy deposited in the fuel rods.

The ABB Atom method of analysis is based on the RAMONA-3B code which has full 3-D core modelling capability equivalent to that of a static 3D core simulator. The POLCA code (Reference 14) is used for providing the core history (burnup and void history distributions), cf. Figure 2. The PHOENIX code (Reference 13) provides the homogenized nuclear constants to both RAMONA-3B and POLCA, including delayed neutron data for RAMONA-3B.

4.1 RAMONA-3B Code

RAMONA-3B is a systems transient code for prediction of the dynamic behaviour of a BWR. It is specifically designed to simulate normal and abnormal operational plant transients, as well as accidents such as the control rod drop accident and ATWS transients. Because of its unique feature of full 3-D modelling of the reactor core, it is particularly suited for transients showing large local effects in the core.

RAMONA-3B originates from a development project carried out in cooperation between the Nuclear Research Institutes of the Scandinavian countries (at Kjeller, Risø and Studsvik) and ABB Atom (formerly ASEA ATOM) in the early 70's. The project resulted in a 3-D BWR systems transient code named ANDYCAP. The code was verfied against start-up measurements in Oskarshamn 1, the first ABB Atom BWR. It was used by ABB Atom during the 70's for generic analyses of the Rod Drop Accident.

The code development has then been continued, as designated by the code name RAMONA-3B, up to the present date, by Scandpower International Consultants and Brookhaven National Laboratory. A substantial part of this development work was funded by the U.S. NRC.

The code has been selected by NRC as reference 3-D BWR systems transient code, and is being applied to special studies as well as for verification of other transient methods. In their work as consultants for the nuclear industry, Scandpower has applied, or provided licencing rights to use the code, to several BWR plant operators in Europe and the United States.

4.1.1 Modelling Characteristics

This section presents a summary of modelling characteristics in RAMONA-3B for neutron kinetics, thermal conduction and thermohydraulics. A complete description of the code is given in Reference 12.

A 1-1/2 energy group, coarse mesh diffusion model in a three-dimensional rectangular coordinate system is used to predict transient three-dimensional fission power distributions in the core. Six delayed neutron groups are accounted for. Decay heat from fission products is computed in RAMONA-3B from ANS Standard 5.1 (1979). All feedback mechanisms between neutron kinetics and thermohydraulics are modelled.

The neutron kinetics equations are solved using a box integration procedure to treat the space variables and an implicit time differencing scheme to treat the time variable. The core symmetry can be octant, quarter, half or full-core.

Thermal energy storage and conduction in fuel elements (pellet, gas gap and fuel cladding), each one representing all the fuel in a computational cell of the three-dimensional mesh for neutron kinetics calculations, is computed using spatial discretization in the radial direction in a finite difference form. Axial conduction and the temperature dependencies of thermal conductivity in the cladding are ignored. Heat capacity in the fuel is modelled as a fourth-order polynomial in temperature. The gap conductance is defined specifically for each fuel type, as a quadratic polynomial in average pellet temperature, but independent of burnup. Implicit iterative time integration is used to solve the conduction equations.

RAMONA-3B has models for two-phase flows with unequal phase velocities described by a slip correlation, subcooled or superheated liquid phase and with transient boron concentration. The Bankoff-Malnes slip correlation is used in all ABB Atom and Scandpower applications. Four equations of vapor mass, mixture mass, momentum and energy conservation describe the coolant dynamics in the vessel.

Two equations of vapor mass and momentum conservation describe the acoustic effects from valve closures in the (adiabatic) steam lines. One boron mass conservation equation is used to predict the transport of boron.

A single pressure is used in the entire system to compute all phasic properties. This technique eliminates efficiently the effects from phasic properties, neglects acoustic effects in the vessel, and contributes significantly to the computing economy in RAMONA-3B.

One closed-contour momentum equation each is used to predict the individual axial velocities in a chosen number of parallel core flow channels. This method increases significantly the computing speed.

The partial differential equation of mixture mass conservation is integrated by a simple quadrature in space. This method also significantly increases computing speed without loss in accuracy.

Without these three advanced modelling features, RAMONA-3B would not be able to compute three-dimensional neutron kinetics and thermohydraulics for multichannel core geometries in the context of a systems code and produce results at acceptable costs.

RAMONA-3B accounts for nonequilibrium vapor generation, unequal phase velocities, wall shear and heat transfer for single-phase and two-phase flow conditions. The prediction of slip is deemed reliable for vapor void levels up to 85-90 %, that of wall shear for forced turbulent flow, and that of heat transfer for forced turbulent convection in single-phase flows and for nucleate boiling in churn-turbulent two-phase flow.

RAMONA-3B has individual component modelling to accommodate BWR systems of US design. All recirculation loops and all steam lines are represented in RAMONA-3B by a single recirculation loop with a single jet pump and a single steam line, respectively.

For the integration of the thermohydraulic equations and the boron transport equations the following options exist in the code:

1) First order Euler explicit integration

2) Second order explicit integration

3) As 1) or 2) but with the momentum equations integrated implicitly

Option 1) is used as a standard.
Option 2) is used for stability analysis
Option 3) is used in slow transients

The time step is determined by user specified error bounds. The acoustics in the steam line are predicted by a fourth-order Runge-Kutta method, coupled with the Simpson rule to control the time-step from specified error bounds.

4.2 Major computer codes supporting RAMONA-3B

The CRDA is studied with RAMONA-3B using an extended code version (RAMONA-3B-SCP2) that has been linked to the standard ABB Atom codes for static core design, PHOENIX and POLCA (References 13 and 14).

The PHOENIX and POLCA codes have been reviewed and approved by the NRC (September 3, 1985 and June 14, 1988). Figures 1 and 2 show the relation and interaction between computer codes used by ABB Atom in the CRDA analysis.

The PHOENIX code is a two-dimensional, multi-group transport theory code which is used to calculate the lattice physics constants of fuel assemblies having varying complexities. The POLCA code is a modified one-group nodal code which is used for the three-dimensional simulation of the nuclear and thermal-hydraulic conditions typical of boiling water reactor cores. Auxiliary codes used with PHOENIX and POLCA include FOBUS, PHOEBE, PHIPO, and POREF.

An auxiliary code for nuclear data polynomial fitting for RAMONA-3B is POLGEN.

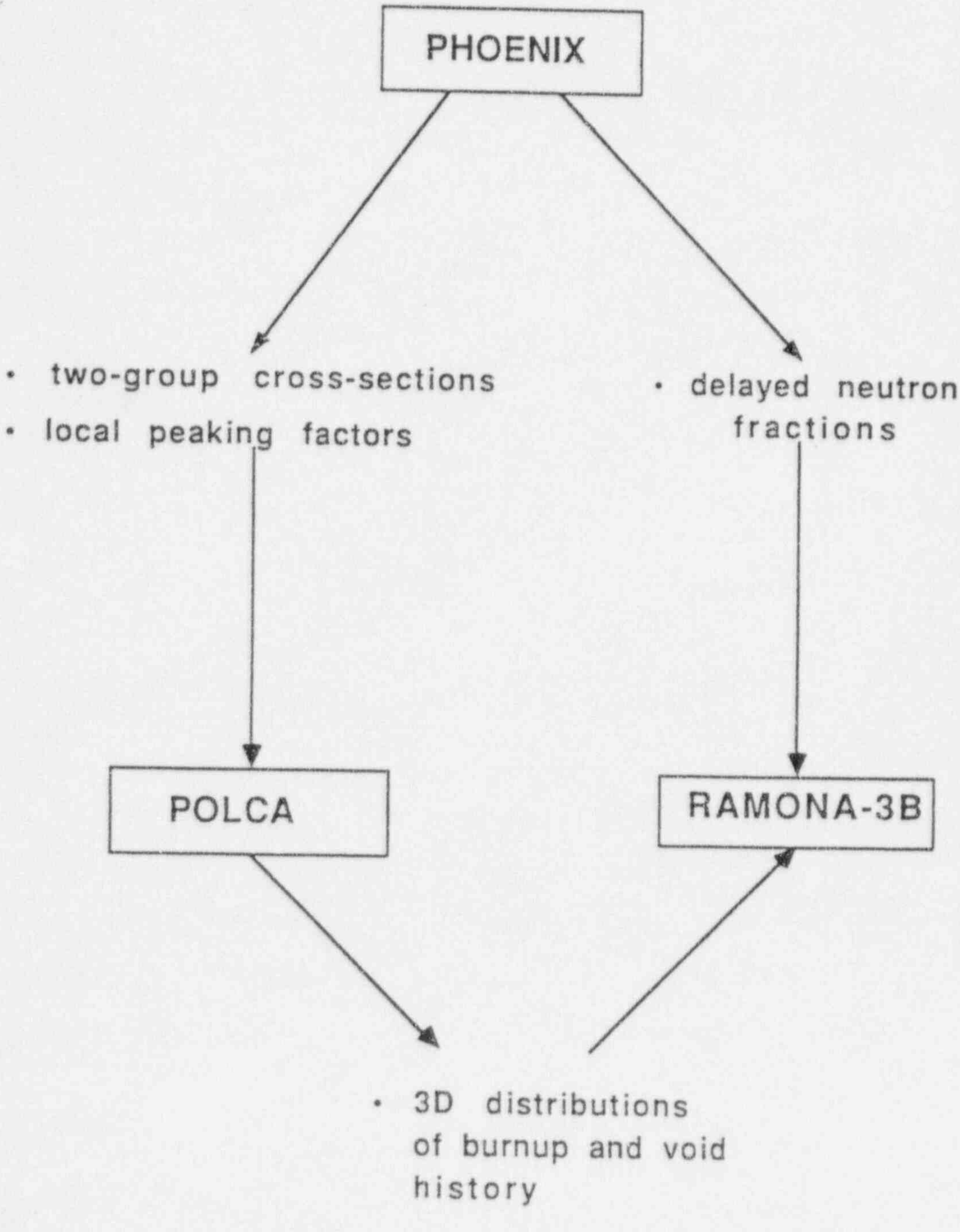
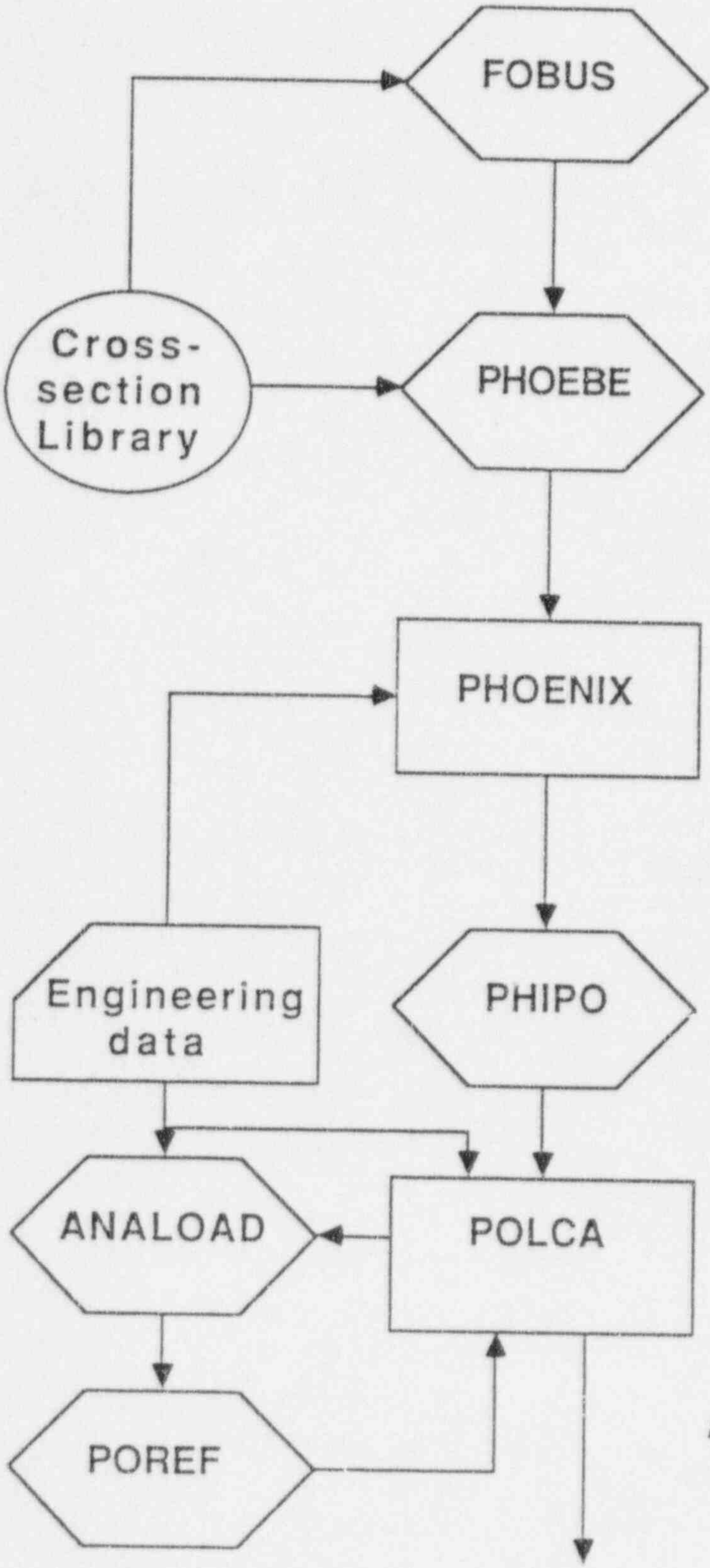


FIGURE 1. Interaction between codes



FOBUS — Monte Carlo transport code for burnable absorber cross-section generation

PHOEBE — Library processing code for nuclear cross-section data

PHOENIX — Two-dimensional multi-group transport theory code for lattice physics constants

PHIPO — PHOENIX output processing code for POLCA input generation

POLCA — 3-dimensional modified one group nodal code for steady-state reactor core simulation

ANALOAD — Fuel shuffling optimization code for core loading pattern generation

POREF — Fuel shuffling code for performing POLCA refuelling

FIGURE 2. Nuclear Design Codes

The PHOEBE code is used to prepare the nuclear data library for PHOENIX. The FOBUS code generates the self-shielded multi-group microscopic absorption cross-sections of the gadolinium burnable absorber isotopes for use in PHOENIX. The PHIPO code serves as the linking code between PHOENIX and POLCA. The POREF code allows user-selected fuel shuffling and fuel loading pattern optimization between refuelings.

4.2.1 PHOENIX Code

PHOENIX is a two-dimensional. multi-group transport theory code which is used for the calculation of eigenvalue, spatial flux and reaction rate distributions, and depletion of rod cells for BWR and PWR fuel assemblies. The code can simulate BWR cruciform control blades containing cylindrical absorber elements, PWR cluster control rods, water gaps, burnable absorber rods, burnable absorbers that are integral with the fuel, water rods, and the presence of objects in the water gaps such as neutron detectors.

PHOENIX is supported by the burnable absorber program FOBUS and by the PHOENIX library service program PHOEBE. PHOENIX is the standard ABB Atom depletion program for BWR fuel assembly and rod cell calculations. Each of the fuel rods is individually treated throughout the calculations; there is no limitation on the number of different rod types that can be represented in the PHOENIX problem. The code can accommodate a variety of geometric configurations including fuel rods with different radii, plutonium fuel, burnable absorber rods, and water holes.

In the water gaps, any number of objects may be specified, such as detectors, control blades, and control blade tips. These are either treated homogeneously or, in the case of a control blade with absorbing rods, heterogeneously. In addition to rod cell and fuel assembly calculations, quadruple assembly problems can be run, consisting of four assemblies in a 2 x 2 array. This option permits the detailed calculation of rodwise power distribution, reaction rates, reactivities, and detector constants, e.g. for the case of an ABB Atom fuel assembly adjacent to three other types of fuel assemblies.

The principal output of PHOENIX is fuel assembly reactivity, isotopics versus burnup, rod-by-rod power distribution, two-group homogenized controlled and uncontrolled cross-sections for POLCA, tables of detector signals, local peaking factor, factors related to the rodwise power distribution used in the critical power ratio correlation, and xenon, samarium and boron microscopic cross-sections. Apart from producing a well ordered compact printout, the results are also written, in an easily retrievable form, on magnetic media for later use by POLCA and other programs.

4.2.2 POLCA Code

POLCA is a modified one-group nodal model designed to provide realistic three-dimensional simulation of the nuclear, thermal and hydraulic conditions in boiling water reactors.

The three-dimensional neutronics of the reactor core are described by a modified one-group nodal model. The nodal equations are the result of a specially adapted coarse-mesh diffusion approximation. A set of coupling coefficients describes the inter-nodal coupling. These coefficients are evaluated from two-group data which are stored as a number of three-dimensional tables. The table entries are burnup, void, and void history. The void affects the neutron energy spectrum and cross-sections, while the void history affects the isotopic composition per node. The neutronics equations are solved by Gauss-Seidel inner iterations with a Chebyshev iteration of the fission source. A thermal coupling correction, based on the asymptotic thermal fluxes of the direct neighbours, is made by modifying the removal cross-sections prior to the iteration process.

The hydraulic calculations are performed by a special version of the CONDOR design thermal-hydraulic code. The CONDOR code is described in Reference 15. It has been reviewed and approved by the NRC (October 11, 1985).

In addition to linear heat generation rate and CPR edits, POLCA also edits bundlewise, core average axial, and three-dimensional nodewise distributions of power, burnup, void, xenon and iodine concentrations, inlet flow distribution, local power range monitor (LPRM), and traversing incore probe (TIP) predicted signals. A criticality search can be made in POLCA with the search conducted on such parameters as reactor power, recirculation pump flow, inlet subcooling, or control rod position. POLCA can be run in quarter-, half-, or full-core configurations.

Each fuel assembly is modelled radially using one node per assembly and axially using up to 25 nodes, which permits the explicit modelling of the top and bottom natural uranium blanket regions.

4.3 Documentation and Qualification of RAMONA-3B

The code version used by ABB Atom is RAMONA-3B SCP2. This code is an extension of the version used, documented and released by Brookhaven National Laboratory (BNL) in 1983 (Reference 12). This report therefore only describes the modifications and extensions that have been incorporated to generate the RAMONA-3B SCP2 version.

4.3.1 Code Documentation

Documentation of basic methods, code features and limitations is found in Reference 12. This report also describes some results from applications and can be considered as a complete documentation of the code version released by BNL in 1983 (RAMONA-3B Mod 0 Level 4).

The present code version, RAMONA 3-B SCP2, includes several additional features and can be considered to be upgraded to BNL version 'Level 10'; the only feature from Level 10 that has not been implemented is the feedwater controller model.

The features of version SCP2 are summarized in the User's Manual (Reference 16). Of the most important extensions (beyond that of Reference 12) the following can be mentioned:

* The nuclear cross-section data representation is made compatible with Scandpower's static 3-D core analysis methods (FMS) and the ABB Atom CORE MASTER system, which includes POLCA. Therefore, RAMONA-3B can directly use the data files employed in the core follow work and describe the core conditions with the same amount of detail, (References 17 and 18).

* Option to input a non-equilibrium xenon distribution, and thus initiate the transient in a non-equilibrium xenon state.

* Effective delayed neutron fractions are treated as nodal variables, being a function of fuel design and burnup state (Reference 19).

* The nodal coupling method (concerning thermal flux diffusion) has been upgraded to that of the static simulator PRESTO (References 19 and 30).

* Reactivity calculations have been implemented (Reference 20).

* Improvements to the Hydraulics have been introduced concerning level tracking and reverse flow conditions (BNL code development work).

* The time integration of the hydraulics has been improved to optionally allow for higher order explicit methods as well as implicit integration of some of the equations (not important for CRDA analyses, References 21 and 22).

* A systematic method to reduce the core model from 3-D to 1-D has been introduced (Reference 23).

4.3.2 Qualification Base

The comparison of calculated results with experimental data is the most important means of validating a systems transient code such as RAMONA. The ultimate proof of the code's ability to predict the plant behaviour is to apply it and compare against actual plant tests. Unfortunatelty, such experimental data bases are relatively limited and show mostly rather small excursions from normal operating conditions. Therefore, one also has to rely on separate effects testings, where certain modules of the code are exercized and compared to test data. Furthermore, for many effects, experimental data exist only for systems in equilibrium, whereas the models in the code need to be applied under transient conditions.

In this section a compilation of the qualification data base for RAMONA-3B is presented, both for separate effects testing and BWR plant tests. In addition, some predictive calculations are summarized in order to demonstrate other applications of the code, even though no direct experimental qualification exists.

4.3.3 Separate Effects Testing

Thermal-Hydraulics Model

In the development of a hydraulics model one has to rely, to a large extent, on experimental data. The RAMONA hydrodynamic model was developed mainly based on early loop experiments (References 24 and 25).

Later on more detailed data became available for electrically heated 6x6 BHWR fuel bundles in the ABB Atom FRIGG loop, operating under various flow, pressure, subcooling and power shape conditions (References 26-28). The following features were tested:

- void distribution
- pressure drop distribution
- transfer functions between different variables
- dynamic response to power ramps
- stability limits

The model, as implemented in the PRESTO code, was qualified against measured void data in electrically heated 8x8 BWR fuel bundles in the FRIGG loop (Reference 29) and reported in the US-NRC topical report on PRESTO, Reference 30. These calculations were performed using the PRESTO Code, but are also applicable to RAMONA-3B since the void models are identical.

Neutronics Model

The RAMONA-3B neutronics model is, for static calculations, identical to that of PRESTO, but is extended to kinetics by means of including time-dependent terms and delayed neutron precursor tracking. A general assessment of the RAMONA-3B neutronics modelling is presented in Reference 31. An extensive qualification base for calculating steady-state power distributions is presented in Reference 30. For transient conditions, on the other hand, the experimental data are more sparse. However, some reactor experiments with recorded LPRM readings are presented in section 4.3.4 below.

Steam Line Modelling

The steam line model has been assessed (Refereces 12 and 32) by comparison with both analytical results and experimental data in pressure variations in the steam line recorded during the Peach Bottom Turbine Trip Tests (Reference 33).

Critical Heat Flux Correlations

When RAMONA-3B is applied for licensing calculations, the code user generally implements CHFR or CPR-correlations for which a qualification base and approval already exists. Such correlations are normally proprietary.

RAMONA-3B contains, in addition, a publicly available CHF correlation package, based mainly on the Condie-Bengston correlation. This has been qualified against experiments, reported in Reference 34.

4.3.4 BWR System Transient Tests

Peach Bottom Turbine Trip Tests

RAMONA-3B has been independently qualified against the Peach Bottom Turbine Trip Tests (Reference 33) by both Brookhaven National Laboratory and Scandpower.

BNL report their results in Reference 12. The steam line model was inlcuded in the simulation, i.e. boundary conditions were imposed on the turbine and bypasss valve action. A relatively coarse mesh model was assumed for the neutronics and the pressure was assumed to vary uniformly throughout the reactor vessel.

Scandpower's results are reported in Reference 35 and illustrated in Figures 3 and 4. The calculations differed from those by BNL in the following respects:

- no steam line was included, the recorded steam dome pressure was imposed as a boundary condition

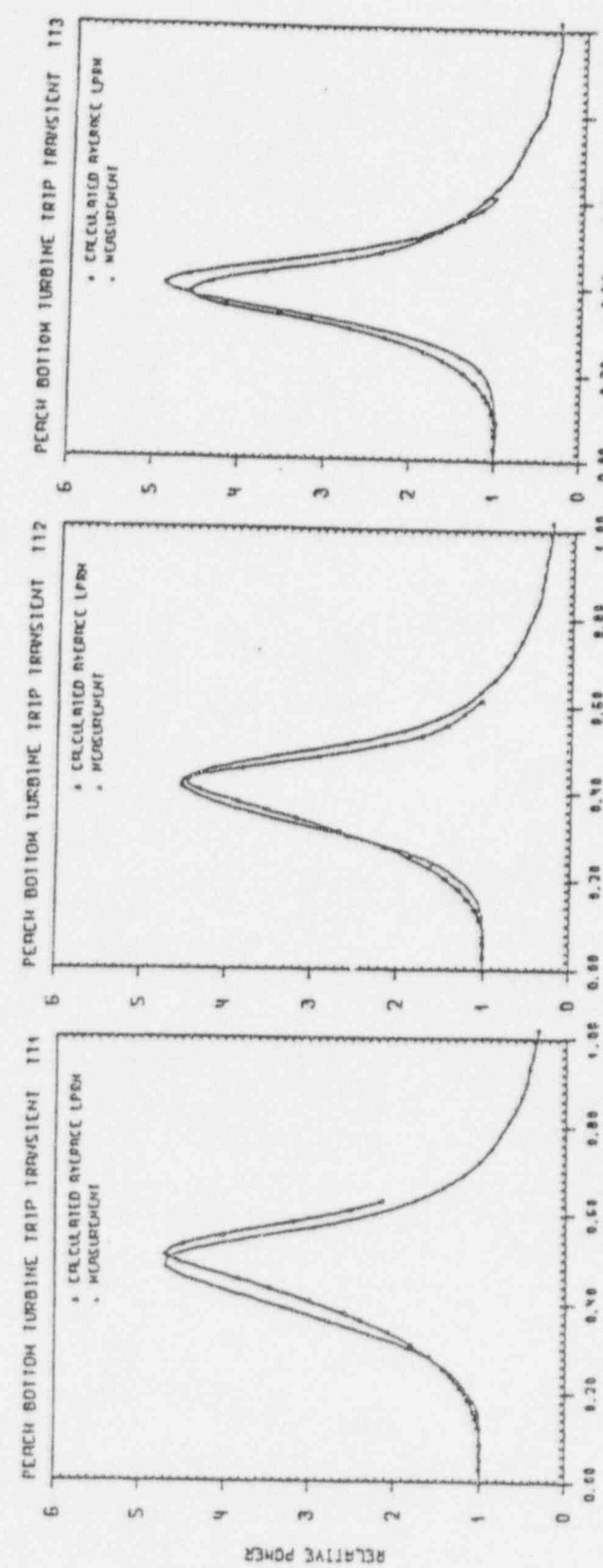


FIGURE 3. Example of Results from the Analysis of Peach Bottom Turbine Trip Tests (from Reference 35)

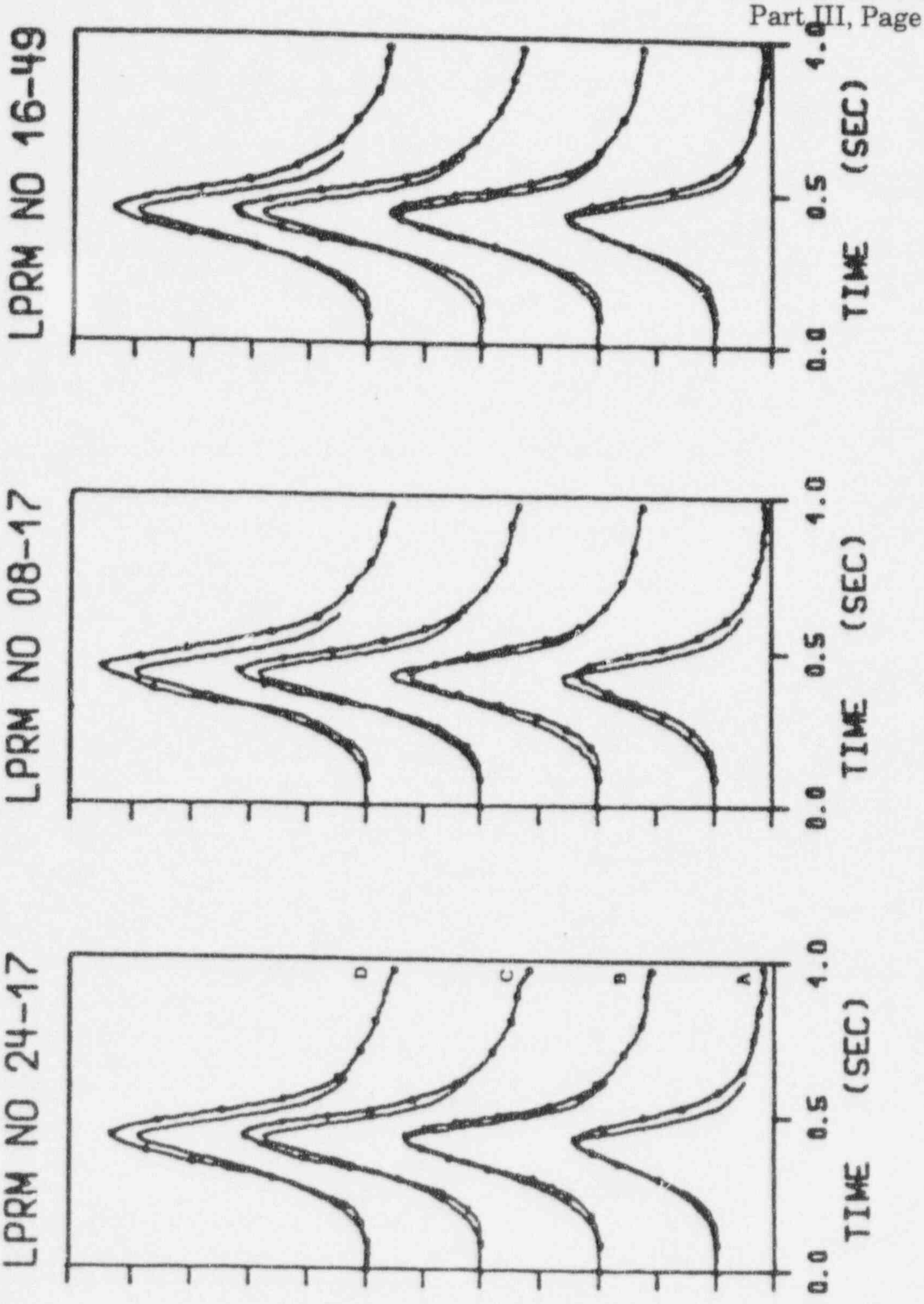


FIGURE 4. Example of Results from the Analysis of Peach Bottom Turbine Trip Tests (from Reference 35)

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- the pressure was allowed to vary non-uniformly within the vessel

- a more detailed core model was used and with a different cross-section data base.

As can be seen in the figures, the rapid increase in total power and the transient variation in local power shape is predicted well in accordance with experimental data.

RAMONA-3B has also been applied to other turbine trip tests (Muhleberg, Brunswick-1), but results have not been published.

Scram Tests in Gundremmingen

An early application of RAMONA-3B was the calculation of control rod induced transients in Gundremmingen A (KRB). This work is reported in Reference 36, and include simulations of a full scram as well as single rod insertions. Comparisons were made between predicted and recorded LPRM response.

CAORSO Stability Tests

Thermohydraulic instability (density-wave oscillations) may appear in BWR reactors at low flow/high power conditions.Stability tests performed in the CAORSO reactor in 1983 demonstrated reactor states in natural circulation that were unstable and showed sustained flow and power oscillations with a constant amplitude (limit cycles). RAMONA was applied (Reference 36) to these experiments by means of time-domain analysis.

4.3.5 Predictive BWR Calculations

ATWS Calculations

In Reference 38, BNL reports results from a study on Anticipated Transients Without Scram (ATWS) initiated by an inadvertant closure of all Main Steam Isolation Valves (MSIV). Even though the study was generic in nature, the actual plant modelled was that of Browns Ferry. Several transient scenarios were investigated in order to provide a better understanding of mitigative effects of operator actions during ATWS, and helpful in the development of adequate Emergency Procedure Guidelines (EPG). This application demonstrates the ability of RAMONA-3B to simulate transients with a duration in the range of minutes up to an hour, and in an extreme flow situation with very low water level and reverse flow conditions.

Rod Drop Accident Analysis

RAMONA-3B is well suited for Rod Drop Accident analyses, bacause of the detailed three-dimensional neutron kinetics modelling. In addition, hydraulic feedback effects, that are of second order importance in this transient at limiting initial conditions but nevertheless not negligible, may be included in the analysis.

Both Scandpower and BNL have performed feasibility studies of this transient (References 7, 12, 39). In Reference 8, BNL compares results from the BNL-TWIGL code and RAMONA-3B for the Rod Drop Accident. In Reference 9 the effects of dropping the central and non-central rods (with the same reactivity worth) are compared, in order to qualify the conventional approach of dropping the center rod. ABB Atom and Scandpower have also performed actual plant applications, but reports are proprietary.

Other BWR Applications

Many other BWR plants have been modelled with RAMONA-3B. Table 1 contains a complete list of applications up to 1989.

Table 1 RAMONA-3B Applications

Plant	Application
Browns Ferry	Half ATWS
Brunsbüttel	Rod drop, rod withdrawal error at startup
Brunswick-1	Overpressurization, Rod drop
CAORSO	Stability test
Dodewaard	Rod drop
Forsmark-1	Overpressurization
Fukushima-III	Overpressurization
Gundremmingen A	Scram tests
Laguna Verde 1	Overpressurization
Mühleberg	Turbine trip test
Oyster Creek	ATWS
Peach Bottom	Turbine trip tests
Oskarshamn 3	Rod drop
Krümmel	Rod withdrawal error at startup
TVO	Stability Analysis
Philippsburg	Rod withdrawal error at startup

4.3.6 Comparison with SPERT Experimental Data

In the late 1960's a program of reactivity accident tests was performed with the SPERT III E-Core, a small oxide fuel pressurized water reactor (References 40 and 41). A number of tests were conducted at cold startup initial conditions which are applicable also to the startup condition of a BWR. The tests involved reactivity insertions ranging from 0.7 to 1.2 dollars and reactor periods from 1900 to 10 ms.

Results from the RAMONA-3B SCP2 code, as applied to some of these tests, are presented in Reference 11.

4.4 CRDA Problem Description

To determine the consequences of a control rod drop, the response of the core and of the fuel adjacent to the dropped rod is simulated, typically, by representing the rod as the central control rod in a one eighth core symmetric problem.

Each fuel bundle is represented by one mesh square in the X-Y plane and by twenty-four axial mesh in the Z direction.

The two-group cross-sections for different types of fuel are obtained from the nuclear design code PHOENIX (Reference 13 and 14) which is used to provide bundle average cross-sections for three dimensional steady state core simulations in normal operation. Cross-sections, which are a function of local burn-up and void history, are provided for each mesh block for a reference fuel temperature and water density.

The burn-up dependence in the cross-sections is fitted to a 4th order polynomial spline function. The void history dependence is evaluated with 2nd order polynomial interpolation. The control rod effect is represented by effective thermal absorption cross-section terms, expressed as 2nd order polynomials in void and burn-up. The void dependence, together with the fuel temperature dependence, is represented by a 2nd order polynomial function, defined at discrete burn-up and void history states.

Doppler coefficients are given for each fuel type for a selected number of burnup levels and void history states.

Kinetic parameters such as effective delayed neutron parameters and neutron velocities are also provided in the nuclear data base as nodal parameters.

RAMONA-3B accommodates the feedback models that are important for a CRDA analysis. The representation of the void feedback is given for both the uncontrolled (control rod out) and the controlled (rod in) condition. The moderator temperature feedback is reflected through the change in moderator density. The Doppler feedback effect is taken into account in the fast group cross-sections.

The model of fuel heatup includes heat transfer to the coolant, but the process can also be treated adiabatically, i.e. with heat transfer neglected. Direct heat components are subtracted in the calculation of fuel heat deposition.

The typical core configuration which produces the highest incremental dropped rod worth in the CRDA has approximately 50 % of the control rods fully withdrawn and the remaining 50 % partially or fully inserted. This applies to a core at hot zero power conditions. To achieve a prescribed large rod worth in a core at cold zero power, where the favourable effect of feedback mechanisms is reduced, but where the core configuration has a larger control rod density, selection of appropriate core loading at the center (neglecting possible contradiction with shut-down margin requirements) is applied.

Moving control rods are represented in RAMONA-3B as a fast and thermal group poison cross-section added to the unrodded fuel. Control rods that are fully inserted and are stationary through the transient are modelled as separate fuel types, that is they have a separate set of cross-sections for rodded fuel.

The dropped center rod is modelled as a boundary which moves at a constant speed, alternatively at constant acceleration. For positions of the boundary interior to an axial node, a weighted average of poisoned and unpoisoned cross-sections is used in that node.

The scram rods whether fully or partially inserted, are represented as poison cross-sections which move into the core at constant speed, or constant acceleration, after a set delay when activated by core power reaching the APRM scram limit (typically 120 %).

5. TYPICAL PROBLEM

To illustrate the application of the RAMONA-3B code to the CRDA, a typical problem is presented here.

The core contains 532 assemblies with three bundle types containing axially zoned enrichment and gadolinia. The analysis is performed at beginning of cycle. The initial conditions represent a core at cold zero power with a large subcooling of 80°C. Bundle averaged cross-sections are obtained from PHOENIX for each axial zone of each bundle type. Figure 5 gives the geometrical representation used in RAMONA-3B.

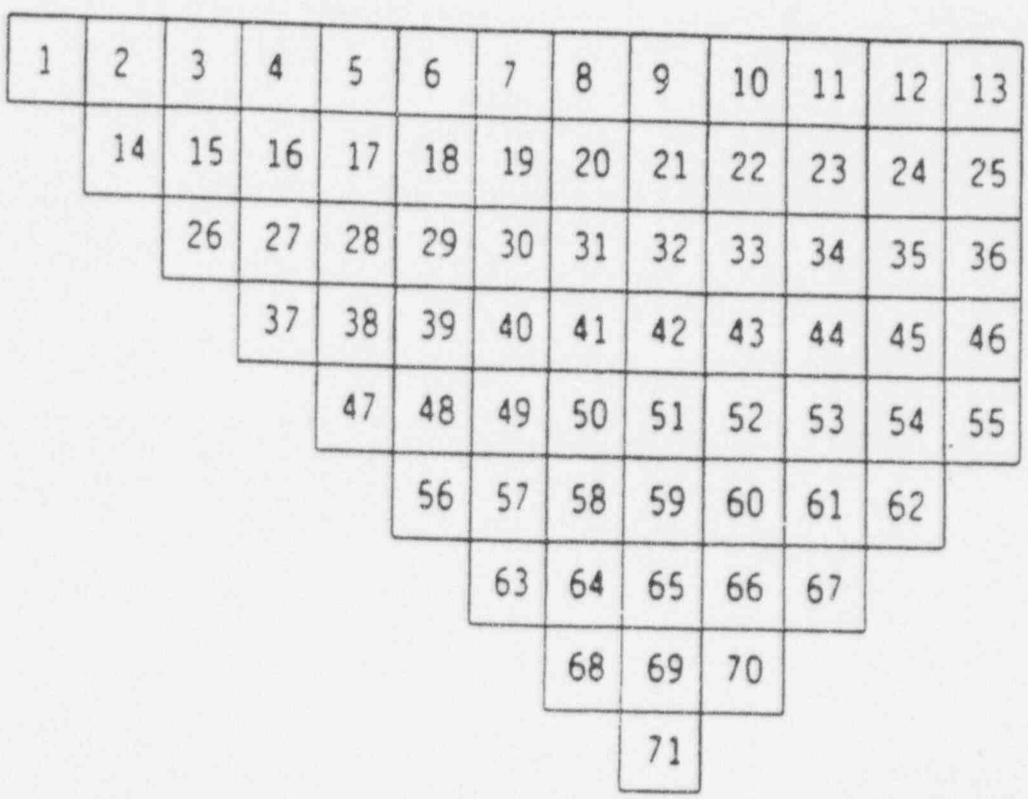

FIGURE 5. Core Representation in RAMONA-3B for 532 Assembly Core

The modelling actually refers to a specific BWR/4 core, and the initial conditions represent beginning of equilibrium cycle 7 exposure conditions, taken from actual core follow calculations with POLCA. The core model parameters can thus be said to represent best-estimate values. However, the effect on the consequences of the CRDA of various parameters and assumptions are addressed in the next section and further discussed in the original work underlying the example illustrated here, Reference 10.

A fuel loading and control rod pattern is selected to represent, at the center of the symmetric problem, the environment of the control rod which is determined from steady state three-dimensional studies to be a candidate for the most limiting dropped rod. Dropped rod worth, rate of reactivity addition ("reactivity shape function") and relative local power are important parameters to be represented conservatively in this simulation. The most essential parameter is the dropped rod worth.

For the problem illustrated here, a selected high target rod worth of approximately 1.2 % is achieved, as stated in the previous section, by selection of core loading at the center, irrespective of possible contradiction with e.g. shut-down margin requirements.

The rate of reactivity addition is calculated intrinsically by RAMONA-3B. Since the core studied is cycle 7, it has a much higher reactivity in its upper parts due to the axial distributions of both burnup and U-238 conversion. These two effects contribute to a conservative "reactivity shape function" and a large power peaking towards the top of the core.

Figure 6 shows the initial, normalized, radial power distribution in the axial zone, node 21 out of 24, receiving the greatest energy deposition during the transient (hottest node).

The result if this procedure is a dropped rod worth of 1.17 %. Rod worth is determined by static RAMONA-3B calculations at zero power and with no change in fuel or moderator temperatures. The "reactivity shape function" is determined from intermediate state point calculations, also with no change in fuel or moderator temperatures and is shown for this problem in Figure 7

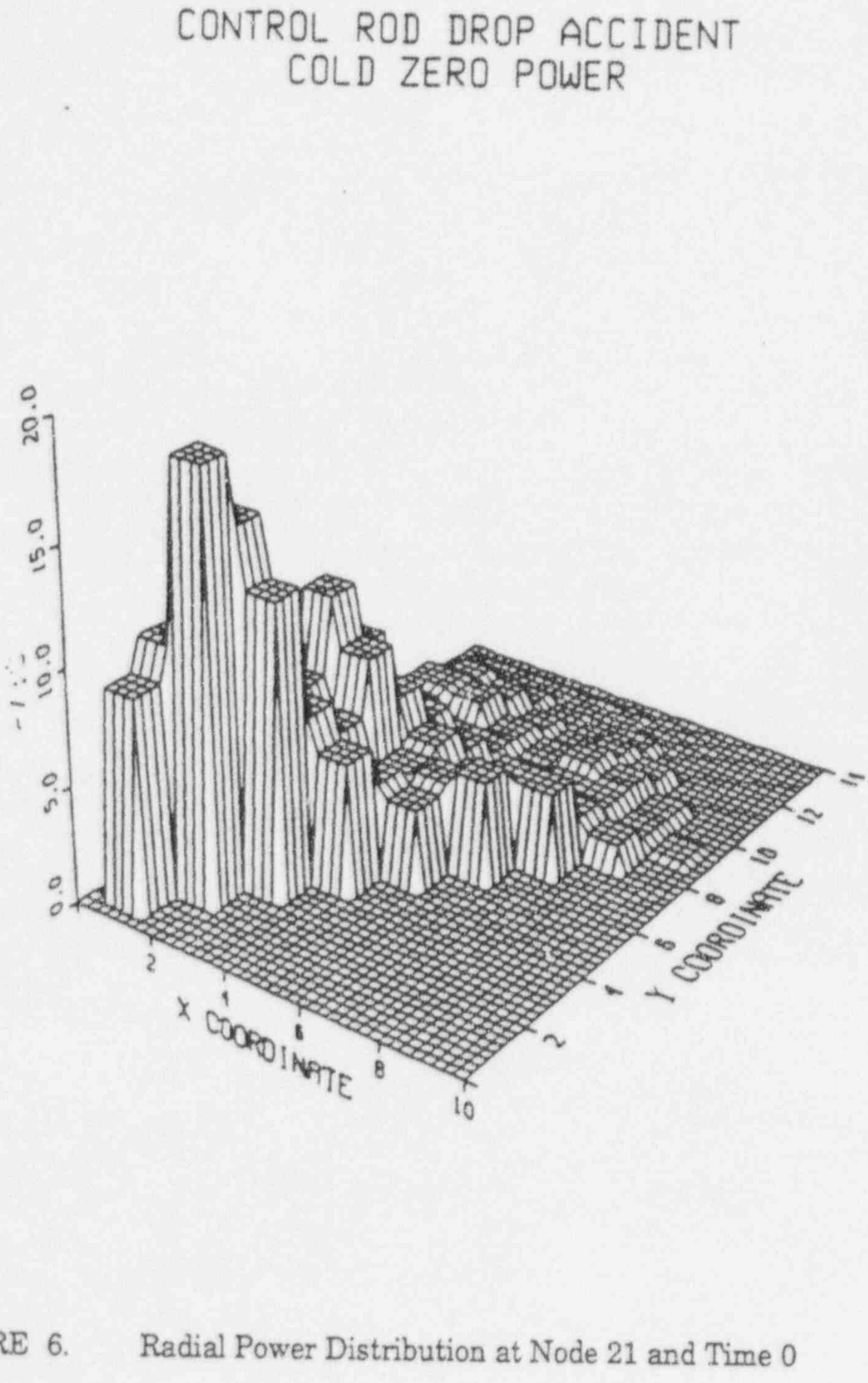

FIGURE 6. Radial Power Distribution at Node 21 and Time 0

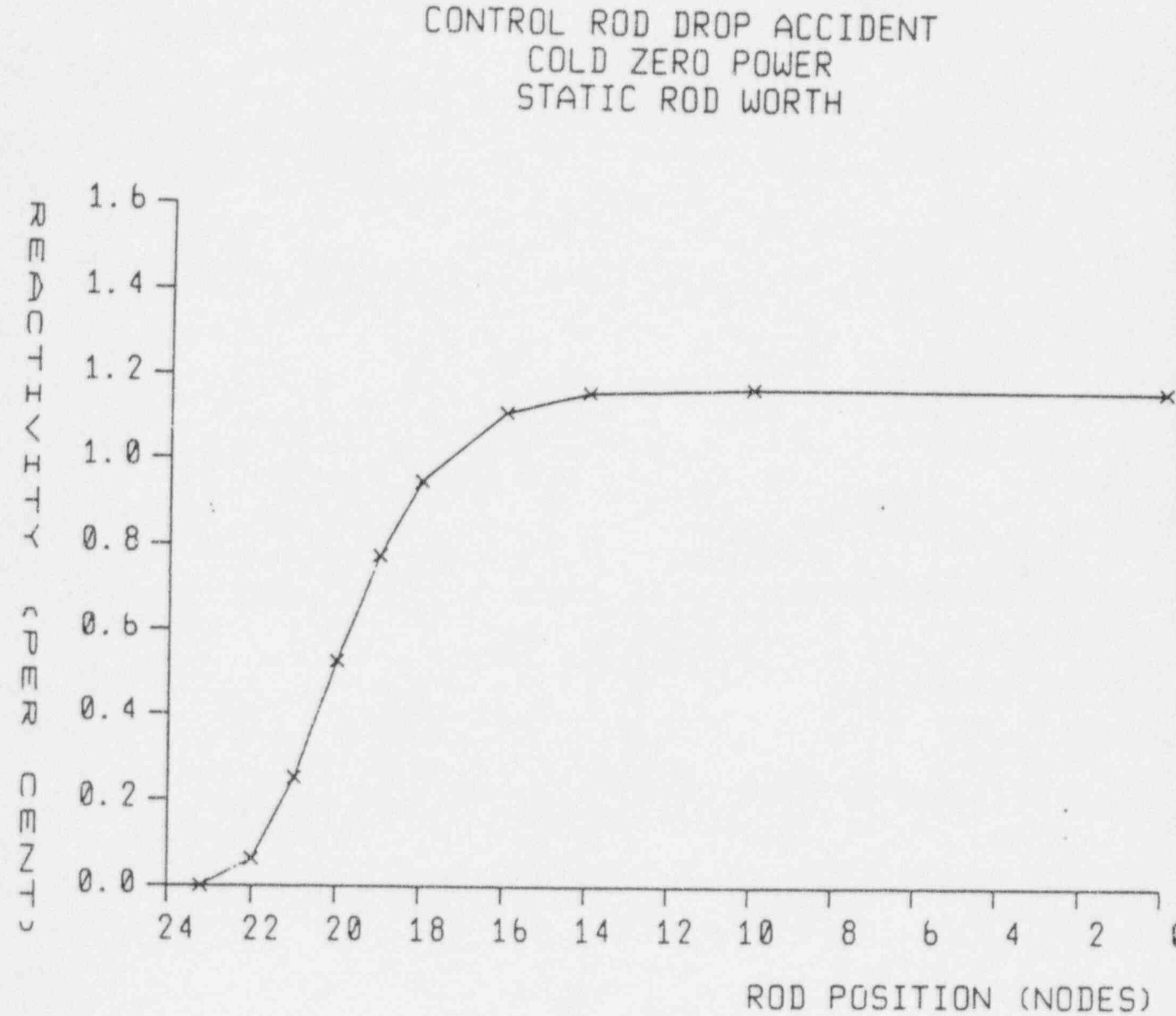

FIGURE 7. Reactivity ($\Delta k/k$) versus Control Rod Position for 1.17 % Dropped Rod

For this simulated CRDA the center rod is dropped from fully inserted to fully withdrawn and core power (logarithmic) as function of time is shown in Figure 8. The fuel temperature of the hottest node is shown in Figure 9 together with the average fuel temperature.

The peak radial average fuel enthalpy, including a pin power peaking factor, reached in this case is 112 cal/gm, see Figure 10. If the peak enthalpy exceeds 170 cal/gm, then the number of rods exceeding 170 cal/gm will be calculated from the pin power distributions for those bundles whose peak enthalpy exceeds 170 cal/gm at any axial level. Pin power distributions within the bundles are provided by the steady state PHOENIX calculations.

The case shown, for cold zero power conditions, includes the neutronics model, fuel heat transfer and hydraulics models of RAMONA-3B. As can be seen from a reactivity edit (Figure 11), however, the moderator density reactivity effect in this case is very small, and could have been neglected without changing the result drastically. (This is shown in the sensitivity study in the next section.)

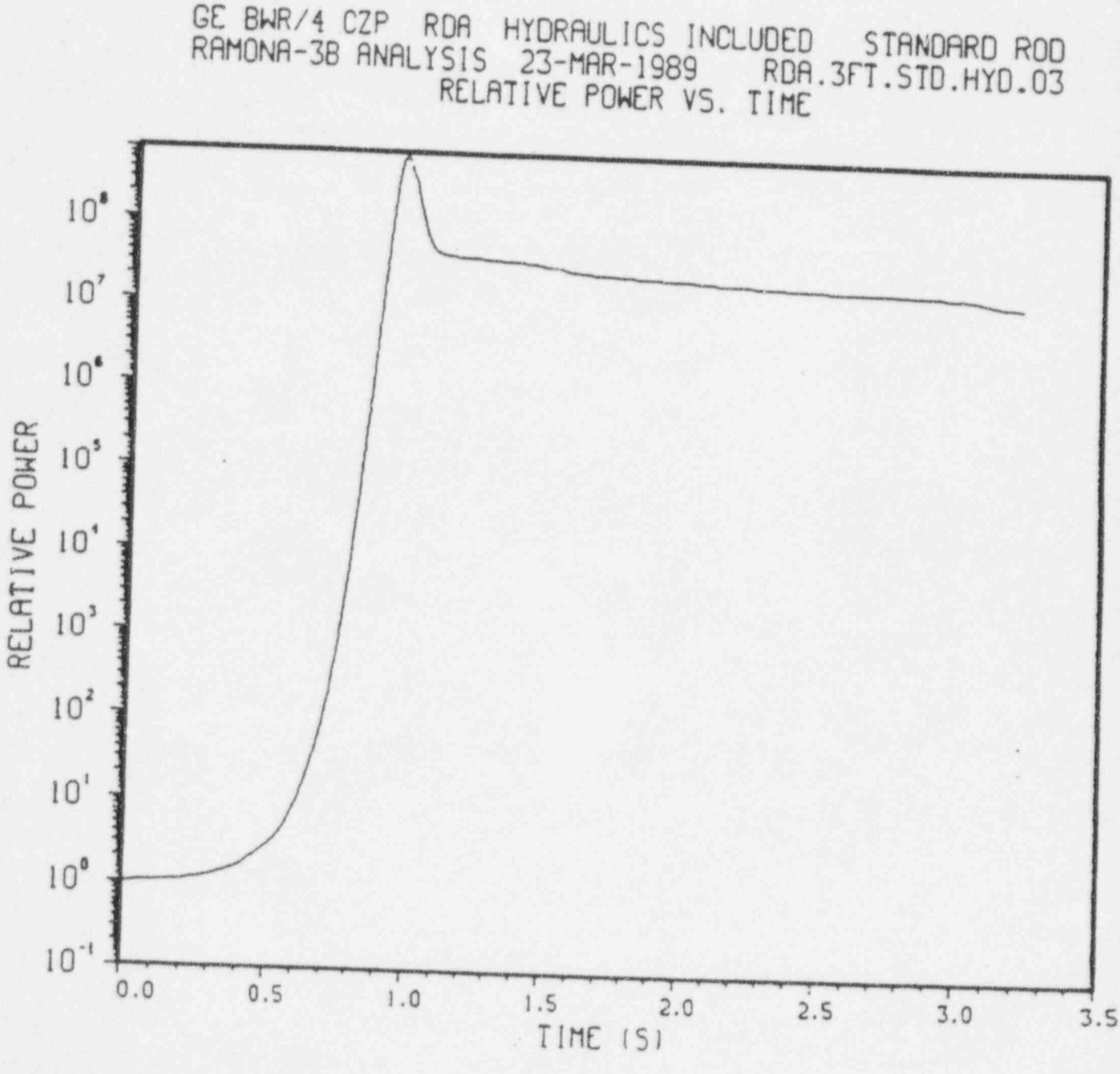

FIGURE 8. Power vs. Time for Control Rod Drop Accident

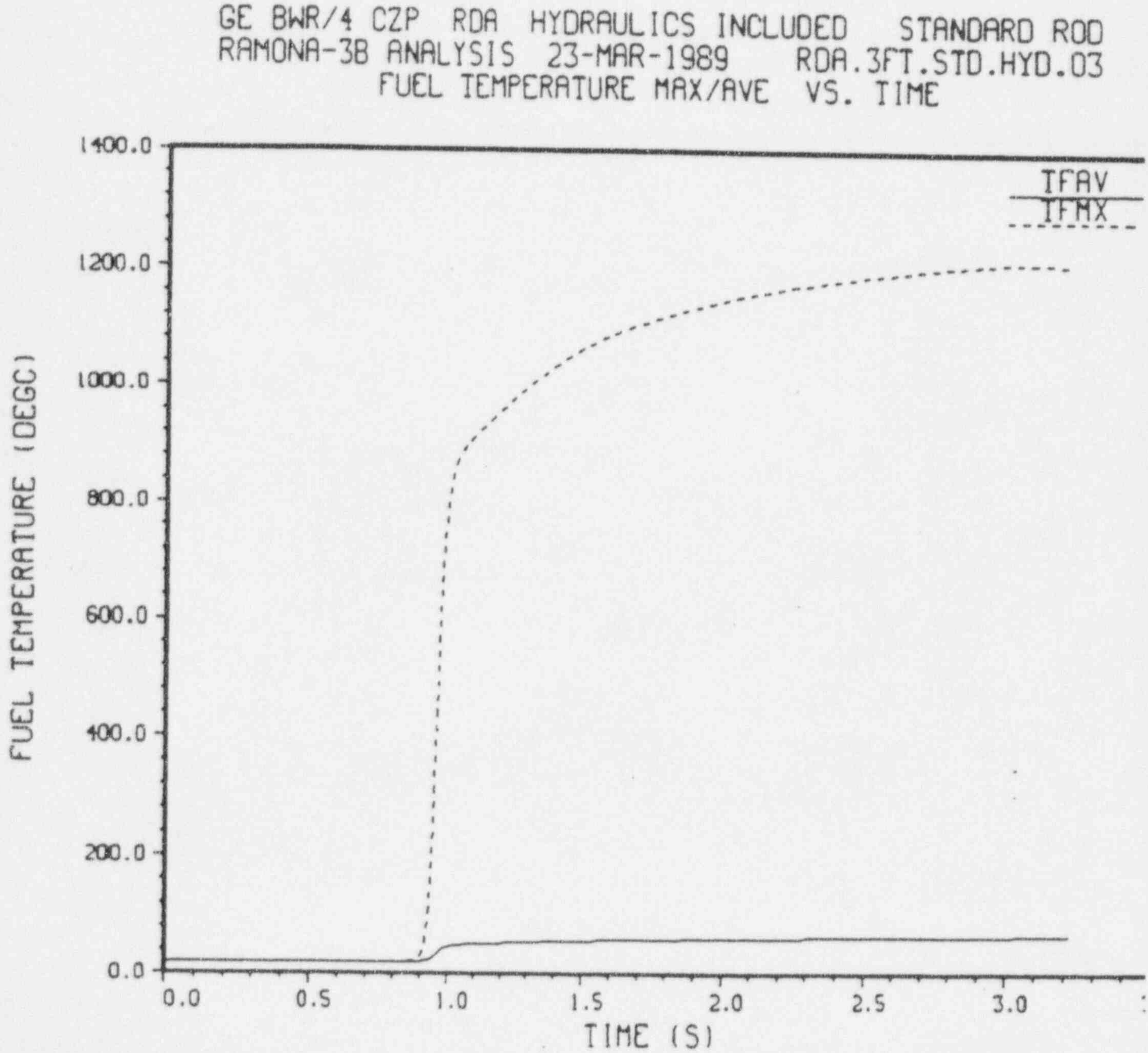

FIGURE 9. Fuel Temperature, Maximum and Average, vs. Time for Control Rod Drop Accident

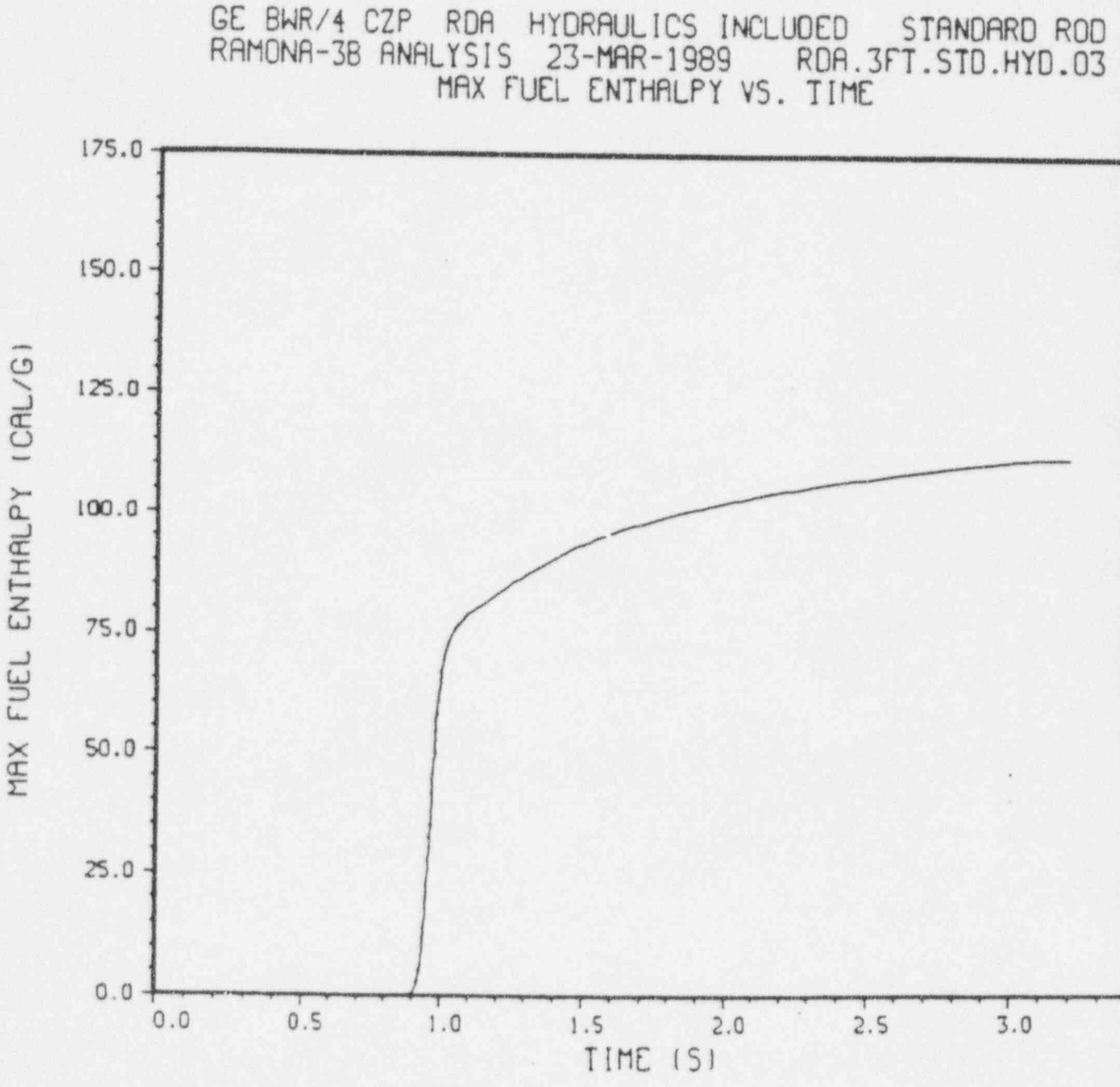

FIGURE 10. Peak Fuel Enthalpy, vs. Time for Control Rod Drop Accident

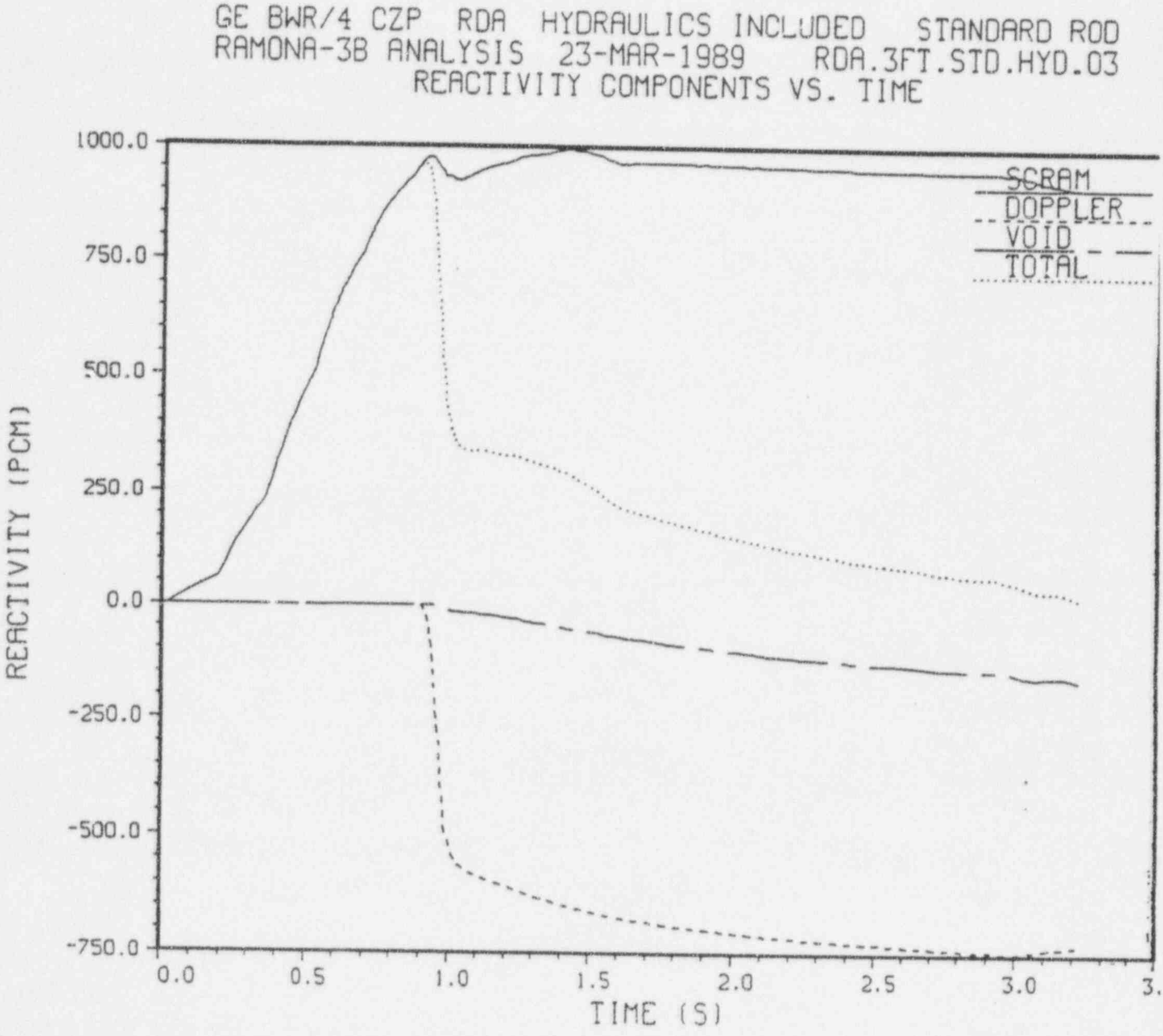

FIGURE 11. Reactivity Components vs. Time for Control Rod Drop Accident

6. SENSITIVITY STUDIES

The method of analysis described includes the effect of moderator density feedback. This means that the method treats explicitly the significant factors related to this feedback mechanism, principally the moderator temperature and void coefficients of reactivity, core inlet subcoling, core flow and prompt power/heat transferred to the coolant. Also the method includes the option to neglect completely the moderator density reactivity effect, i.e. to disregard the hydraulics model.

To evaluate the effect of essential input data and the calculational options described, and to further illustrate the RAMONA-3B methodology for the CRDA, perturbation cases were run with the previously described problem as a base case. The results of the peak fuel enthalpy calculation were compared for the following parameter changes:

(1) Moderator density feedback
(2) Heat transfer to coolant
(3) The insertion rate of reactivity after the initial 0.5 % (i.e. when the inserted reactivity approaches prompt criticality)
(4) Core layout and control rod pattern
(5) The total worth of the dropped rod.

The base case was analyzed with moderator density feedback included. The peak fuel enthalpy was 112 cal/gm. When the effect of changing moderator density was neglected the peak enthalpy reached 133 cal/gm.

With heat transfer to coolant also turned off the base case value increased further to 215 cal/gm.

The effect of accident reactivity insertion rate was examined by changing the control rod drop speed, in the case with moderator density feedback excluded, from 3.11 ft/s to 5 ft/s. Reference 1 justifies 3.11 ft/s as being the maximum rod drop speed that could be achieved allowing, at the 3 σ level, for tolerances in physical dimensions. It is the licensing basis for this accident in many BWR plants. The speed of 5 ft/s was selected to match the speed chosen for study by others (References 2 and 3).
The faster drop speed resulted in a slightly higher peak fuel enthalpy of 136 cal/gm.

Another aspect of insertion rate of reactivity is the neutronic importance of the region of the core first affected by the moving tip of the falling control rod blade. This is a necessary consideration in selecting core configurations for study in the plant specific analysis.

The comparatively small effect of the core configuration for CRDA consequences, once the dropped control rod worth has been selected, was demonstrated in a case where the control rod pattern was changed from a pattern with about 25 % of the rods withdrawn (every fourth rod) to one with about 50 % withdrawn (every second rod). In order to achieve the same rod worth as in the base case (1.174 %) the central fuel bundle was exchanged. The construction of this configuration resulted in a central dropped rod worth of 1.166 %. In this case the drop speed was 5 ft/s and moderator density feedback was neglected. The peak fuel enthalpy increased from 136 cal/gm to 144 cal/gm.

To illustrate the sensitivity to the reactivity worth of the dropped control rod a case with a reactivity worth of 1.43 % was run. The result was 150 cal/gm.

The above results are summarized in Table 2.

Some of the parameters that were not subject to a sensitivity study but which were selected with a conservative margin, or are less important, in the base case are the delayed neutron fraction and scram insertion rate.

RAMONA-3B treats the effective delayed neutron parameters as nodal parameters, in line with the cross-section data. The average importance-weighted β-value was 0.00585, i.e. a value typical for equilibrium cycle conditions. Values are higher for beginning of life cores. A change in delayed neutron fraction to, say, 0.00740, which is typical for a BOL core, would lower the peak fuel enthalpy by 5 to 10 cal/gm (Reference 7). Thus one may conclude that the ß-value is less important for superprompt critical excursions, in accordance with elementary theory.

The scram speed in the calculations was 2.5 ft/s with a delay of 0.5 s from the time the total core power of 120 % was reached.

Technical Specification limits are placed on scram rate in the form of maximum times from de-energization of the scram solenoid valve for control rods reaching 5 %, 20 %, 50 % and 90 % of insertion.

The required values do not vary greatly from plant to plant and they correspond to a delay of approximately 0.2 s in the scram system, with a scram speed of 3 ft/s.

In the model used, all scram rods are initially fully withdrawn from the core. Since the power distribution in the core is shifted very much to the top of the core - an effect which is typical for an equilibrium core at start-up conditions - the negative reactivity contribution of scram will not be significant until the scram rods are inserted considerably into the core.

An actual startup control rod configuration would contain rods at intermediate axial positions. This would lead to a more favorable scram reactivity insertion than in the situation reported here, since scram reactivity contribution from partly inserted rods would be supplied very much earlier in the transient.

The above results are summarized in Table 2.

A significant conclusion to be drawn from this section is that a careful selection of the candidates for the limiting dropped rod, i.e. the rod worth, must be performed. The range of investigation can usually be confined to the startup range from cold critical to hot standby. Dropped control rod worth, which obviously directly affects initial reactivity insertion rate, is the basic parameter against which survey results should be screened. Once a region of the startup process is determined in which the highest incremental rod worths are found consideration must be given to the core conditions which can affect the results for a given rod worth. For realistic rod worths there will usually be enough margin in the results to permit conservative bounding values of such parameters as Doppler coefficient, delayed neutron fraction and scram reactivity insertion. A description of the application methodology that provides the assurance that design bases are not exceeded is found in Reference 42.

Table 2
Summary of Peak Enthalpy Calculations

	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
Parameter change	-	Mod. Dens. Feedb.	Heat Transfer	Drop Velocity	Control Rod Dens	Rod Worth
Rod Worth (%)	1.174	1.174	1.174	1.174	1.166	1.426
Moderator Density Feedback	Yes	No	No	No	No	Yes
Heat Transfer to Coolant	Yes	Yes	No	Yes	Yes	Yes
Rod Drop Velocity (ft/s)	3.11	3.11	3.11	5.0	5.0	3.11
Control Rod Density (% withdrawal)	25	25	25	25	50	25
Peak Fuel Enthalpy (cal/gm)	112	133	215	136	144	150

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7. DETERMINATION OF LIMITING ROD

As discussed in the previous section the severity of the rod drop event is dependent to a large extent upon the reactivity worth of the dropped rod. For this reason, the current mode of operation in most US BWR's is designed to reduce possible control rod worths to values below that which would result in dropped rod enthalpies approaching the design limit if 280 cal/g.

This mode of operation is termed Banked Position Withdrawal Sequence (BPWS) and is described in Reference 4.

Analyses performed by ABB Atom for BWR's with BPWS will follow the operational constraints of BPWS as described in Reference 4.

This section describes the methodology for determining the worst dropped rod. In addition, examples of rod drop calculations are provided which demonstrate the application of the methodology on a specific BPWS core.

The dropped rod resulting in the worst consequences is a high worth rod which occurs during the operation of the core in an unvoided state. Void feedback greatly reduces the severity of a rod drop. Therefore, the worst dropped rod occurs at core states in the range from cold startup to hot standby conditions.

The BWR simulator code POLCA (References 13 and 14) is used to determine the worst dropped rod candidates. Calculations are done at both cold and hot standby conditions to determine the highest worth rods. All unique rods in each BPWS group are evaluated.

A worst rod worth calculation was performed on the Nine Mile Point Unit 2 initial core. In Table 15.4-9 of the FSAR (Reference 43), the worst rod is described as rod 26-35 in BPWS group 7 of Sequence A, with Sequence A groups 1, 2, 3, and 4 fully withdrawn and the remaining rods of Group 7 at notch 12. The core conditions are beginning of cycle with no xenon present and the plant at hot standby. The worth of rod 26-35 dropping from 0 to 48 (full-in to full-out) is given in the FSAR as 0.004658 increase in k_{eff}. For the same problem, the POLCA calculated increase in k_{eff} is 0.00471. This comparison demonstrates that the ABB Atom methodology is consistent with the FSAR results.

8. CONCLUSIONS

The following conclusions can be drawn:

a) A BWR control rod drop accident analysis methodology has been described which includes moderator feedback.

b) The methodology described is acceptable for licensing evaluations of the control rod drop accident when appropriate initial conditions and the maximum expected individual control rod worths are used.

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CENPD-284-NP-A, RPA 89-112-NP-A, and RPA 89-053-NP-A REPORT

Part IV

Body of RPA 89-053-NP-A Report

Note that the responses to requests for additional information regarding this part of the report are included in Appendix B of Part II of this Report

ABB Atom

Report

Ref.	RPA 89-053		
From	RPA	Date	89-08-01
Reg.		Page	1 (55)
Author	R Jadrny, 7051	Issued by	RJ
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Title

ABB Atom High Worth Control Rods for US BWRs.
Rod Drop Accident Analysis.

Abstract

The Control Rod Drop Accident, CRDA, has been analyzed using the standard ABB Atom methodology, which employs the RAMONA-3B computer code as the basic tool. The influence on peak fuel enthalpy of replacing a standard control rod with a High Worth Control Rod, HWCR, having 15 % higher rod worth, has been determined by performing calculations for both types of control rods and comparing the results. Heat transfer to the coolant as well as reactivity feedback due to density changes in the coolant have been accounted for. Calculations have also been performed where these effects have been neglected. Sensitivity studies have been performed to show the influence of the control rod worth, rod drop velocity, and initial control rod pattern.

The results show that the replacement of standard control rods by HWCR´s will increase the maximum radial average fuel enthalpy following the CRDA by approximately 40 cal/g when accounting for hydraulic feedback and heat transfer from the fuel rods during the accident. At the control rod worths assumed in this work, and which are typically limiting for US plants, the peak fuel enthalpies are 112 cal/g and 150 cal/g for the standard control rod and the HWCR respectively.

The sensitivity studies that have been performed show that the effect of neglecting the hydraulic feedback, compared with including it, is to increase the peak fuel enthalpy with 20 to 35 cal/g . Neglecting the heat transfer from the fuel rods leads to an additional enthalpy increase of 70 to 80 cal/g.

The conclusions from the analyses are that even though the peak fuel enthalpy is increased by replacing standard control rods by HWCR´s, there is still ample margin to the limitations set forth in the Standard Review Plan, in particular to the peak fuel enthalpy limit of 280 cal/g.

Copies: BX Lindner, Jonsson, URA4, RPA
Summary: RP, UR, URA, MB Thunberg

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FIGURE

1. INTRODUCTION

The primary benefit of the High Worth Control Rod (HWCR) is that a larger amount of negative reactivity is inserted into the reactor core to achieve and maintain a sub-critical configuration. The HWCR thus has a fundamental advantage for reactor safety.

Safety analysis associated with the use of control rods is focused on the opposite phenomenon - reactivity increase in a critical core configuration upon rod withdrawal or rod drop. The postulated Control Rod Drop Accident (CRDA) is used as a design basis accident in the safety analysis of BWRs and is considered as the bounding case covering credible accidents involving reactivity insertion.

The purpose of the work reported here is to show the influence on the outcome of a control rod drop accident by replacing standard control rods in a US BWR by HWCR's. The HWCR's are assumed to have 15 % higher rod worth than the standard control rods. The rod worth for the standard control rod has been taken as 0.012 Δk which is the highest value reported for conditions with a maximum number of inoperable control rods. The results are considered to be generically applicable to all US reactors employing the so-called Banked Position Withdrawal Sequence, i.e. all BWR/4-6, and for all fuel types in these reactors.

Section 2 of this report gives an introductory background concerning control rod drop analyses for US BWR's. Section 3 describes the CRDA accident. The evaluation criteria are outlined in section 4. Section 5 summarizes the calculation models that have been used and describes the application. (Reference 1 contains a more detailed description of the RAMONA-3B models with an emphasis on the CRDA application.) Section 6 presents initial conditions and the results from the calculations are reported in section 7. Section 8 contains a discussion on the applicability of the results to different reactors and different fuel types. A summary is presented in section 9.

It will be shown that the use of BPWS limits the rod worths sufficiently to give ample margin to the 280 cal/g limit when HWCR's replace standard control rods.

2. Background

General Electric Company (GE) has published several reports on the CRDA for US BWRs (Reference 2-5). These reports are generically applicable for standard GE control rods and cover a large number of input variables including different fuel types and core designs at different exposures and initial conditions. They also cover the considerations which must be made in determining the limiting cases of drop accidents for a sufficiently large number of cases to be bounding for most fuel cycles in US BWR plants.

The severity of the CRDA depends to the largest extent upon the reactivity worth of the dropped rod. For this reason, the current mode of operation in most US BWRs is designed to reduce possible control rod worths to values below that which would result in peak fuel enthalpies approaching the design limit of 280 cal/g. This mode of operation is termed the Banked Position Wihdrawal Sequence (BPWS) and is described in Reference 5.

The BPWS method of rod withdrawal limits incremental rod worths to an average value of approximately 0.005 Δk, as evaluated in Reference 5. The highest value was found to be 0.0083 for normal conditions, and 0.012 for conditions with a maximum number of inoperable (fully inserted and bypassed) control rods. The corresponding calculated peak fuel enthalpies in Reference 5 (using the GE methods) are 135 and 232 cal/g respectively. Both these values are well below the acceptance limit 280 cal/g and thus confirm the efficiency of the BPWS on a generic basis.

The CRDA has also been studied by Brokhaven National Laboratory with similar assumptions as in the GE analyses (Reference 6). BNL have also analyzed the CRDA using RAMONA-3B. These studies have shown that the above results are conservative (Reference 7).

3. Description of the CRDA accident

The CRDA assumes the breakage and disconnection of an inserted rod drive from the control blade. It is postulated that the drive mechanism is withdrawn while the control blade sticks in position and that the blade subsequently falls at its maximum velocity to the position of the drive. Since it is assumed that the event can occur in any reactor operating state, considerations must be given to all the control rod configurations which can occur in normal operation. Equipment malfunction and the consequences of an operator error (e.g., operator selection of an out of sequence rod) must also be considered. For BPWS plants, the Rod Worth Minimizer (RWM) is used below a specific power level (typically 20%) to enforce the rod withdrawal sequence.

The accident would have its severest consequences at low or zero power conditions when the control rod patterns required to establish criticality provide the highest values of incremental (dropped) single control rod worth. The presence of voids in the core at any significant power level will significantly decrease the consequences of the accident through the negative void rectivity feedback. There is also a shorter delay at higher initial power before fuel heat up causes the Doppler feedback to offset the inserted reactivity.

The sequence of the accident is as follows:

a) At some time a fully inserted rod becomes decoupled from its drive and sticks in the fully inserted position.

b) During reactor startup, rod patterns are employed which are permitted by the constraints on rod movement imposed by Technical Specifications (e.g., regarding maximum number of inoperable rods) or hardware (e.g., RWM). At some time, at critical reactor conditions, the rod pattern exists for which the decoupled rod has the maximum incremental worth from fully inserted to the position of its drive.

c) The decoupled rod drops at its maximum velocity (conservatively assumed 5 ft/s, or 3.1 ft/s as determined from experimental data) to the position of its drive.

d) The reactor goes on a positive period and the initial power burst is terminated by inherent reactivity feedback (Doppler).

e) Scram is initiated by high neutron flux (IRM or APRM).

f) Moderator heating, and possibly voiding, contributes to the reactivity feedback.

g) After some delay in scram initiation all withdrawn rods except the decoupled rod scram the reactor at technical specification rate and terminate the accident.

4. Evaluation Criteria

The criteria against which the consequences of the control rod drop accident are evaluated are based on meeting the requirements of General Design Criterion 28 as it relates to the effects of postulated reactivity insertion 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.

They are given in NUREG-0800, Standard Review Plan, as:

1. Reactivity excursions should not result in radially averaged fuel rod enthalpy greater than 280 cal/g 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 code.

3. The number of fuel rods predicted to reach assumed fuel failure and associated parameters such as the amount of fuel reaching melting conditions will be an input to a radiological evaluation. The assumed fuel failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/g at any axial location for zero or low power conditions, and fuel cladding dryout for rated power initial conditions.

5. CALCULATIONAL MODEL

An account of the ABB Atom methodology employed in Control Rod Drop Analyses is given in Reference 1.

The primary tool is the RAMONA-3B code (Reference 8), that has been linked to the standard codes for static core design, PHOENIX and POLCA (References 9 and 10).

The RAMONA-3B neutronics model employs a detailed nuclear data-base with 2-group macroscopic cross-sections tabulated vs fuel type, exposure, exposure-averaged void, actual void (moderator density), and fuel temperature. All nuclear data and dependencies are calculated using the cell data code PHOENIX. Standard control rods and high-worth control rods are described by separate data bases. History parameters (i.e. nodal distributions of exposure and exposure-averaged void) are taken from static 3-D core follow calculations with POLCA.

The calculated peak fuel enthalpy resulting from the CRDA depends on parameters such as rod worth, rod velocity, delayed neutron fraction, Doppler coefficients and inlet subcooling, and since these parameters are not sensitive to core size, fuel design and exposure, when a certain worth (Δk) of the dropped rod has been specified, it is sufficient to study the transient for a representative reference core. Thus, a typical core with standard 8x8 BWR fuel at beginning of equilibrium cycle exposure conditions has been selected. These conditions were taken from actual core follow calculations with POLCA for the quadrant core symmetric core loading.

The 532 fuel assembly D-lattice reference core is modelled in RAMONA-3B with an octant of the actual core represented (in order to reduce the problem size), with each fuel element individually represented in the neutronics model in 6x6 inch nodes (1704 in total).

The core geometry and neutronic channel enumeration are shown in figure 5.1.

1	2	3	4	5	6	7	8	9	10	11	12	13
	14	15	16	17	18	19	20	21	22	23	24	25
		26	27	28	29	30	31	32	33	34	35	36
			37	38	39	40	41	42	43	44	45	46
				47	48	49	50	51	52	53	54	55
					56	57	58	59	60	61	62	
						63	64	65	66	67		
							68	69	70			
								71				

Figure 5.1 Core geometry and neutronic channel enumeration

The fuel loading of the selected core comprises 8x8-fuel with one or two water rods, with different enrichment and gadolinia designs and with or without natural uranium blankets. Data were generated for four representative material compositions, which were selected to represent all bundle types. They are given in Table 5.1.

Type	Material Composition. Enrichment (%)	Gadolinium No of pins	Gadolinium Enrichm. (%)
1	2.66	5	2.0
2	3.03	6	2.0
3	3.01	7	3.0
4	0.71 nat.Uran	-	-

Table 5.1 RAMONA-3B Nuclear Data Type Definition

The dropped rod is assumed to be located at the center of the core. Control rod and fuel loading patterns are selected which maximize the worth of the central control rod. The drive of the dropped control rod is assumed to be at its fully withdrawn position, while the (decoupled) rod is stuck in the core at its fully inserted position.

The initial pattern of control rods includes only fully inserted or fully withdrawn control rods, figure 5.2.

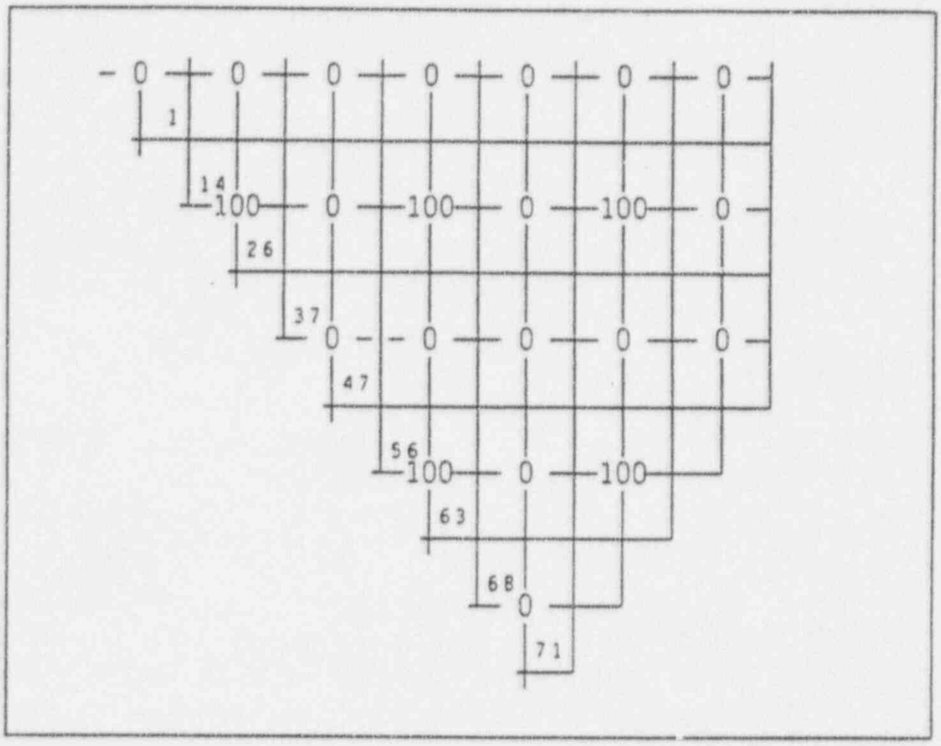

Figure 5.2 Initial control rod position (% withdrawal).
"25 %" withdrawal pattern.
(Reference neutronic channels in small digits)

The prescribed large worth of the dropped central rod has been achieved by selection of core loading at the center (neglecting possible contradiction of shut-down margin requirements). The reactivity worth of this rod is determined by static calculations with no change in fuel or moderator temperature. (The fuel in neutronic channel 1 is (8x8-2) with blankets.)

Reactivity feedback due to moderator heating is not a major feature in reactivity insertion accidents at low power - particularly at low moderator temperature (large sub-cooling). In the problem studied, significant moderator heating will occur in the central parts of the core only. Thus the thermal-hydraulics model was simplified to represent only the central channel individually and with gradually coarser (lumped) channels towards the periphery of the core. The bypass flow was neglected. In total, 6 hydraulic channels were represented, with 24 axial nodes each (144 thermal-hydraulic nodes). The hydraulic channel definition is given in figure 5.3.

1	2	2	4	4	5	5	5	5	5	5	5	6
	3	3	5	5	5	5	5	5	5	5	5	6
		3	5	5	5	5	5	5	5	5	5	6
			5	5	5	5	5	5	5	5	5	6
				5	5	5	5	5	5	5	5	6
					5	5	5	5	5	5	6	
						5	5	5	5	6		
							5	5	6			
								6				

Figure 5.3 Hydraulic channel definition.

The fuel rod model represents one average fuel rod for each node of the neutronics model. The radial heat conduction equation is solved using 4 nodes (of equal volume) in the fuel pellet and 2 nodes in the cladding, i.e. 10244 temperature nodes in total are employed.

In order to limit the amount of cell data calculations, branch-off calculations, i.e. calculation of nuclear data dependencies in moderator density, fuel temperature, and control rod presence, were made for fuel type 2 only, to represent fuel types 1-3. Base cross sections, including exposure and void history dependencies, however, were represented for the three different fuel types present in the core, and for the blanket region.

Polynomial functions describing the fuel temperature dependence in the cross-sections (the Doppler effect) are evaluated at four exposure states. The same applies to the polynomials describing the moderator density dependence. Moderator density dependence is included in all the Doppler coefficient data. Examples of evaluated Doppler coefficients are shown in Figures 5.4 and 5.5.

The neutronics model also represents, as a nodal parameter, the total effective delayed neutron fraction (β). The average importance-weighted β-value is 0.005853, i.e. a value typical for equilibrium cycle conditions.

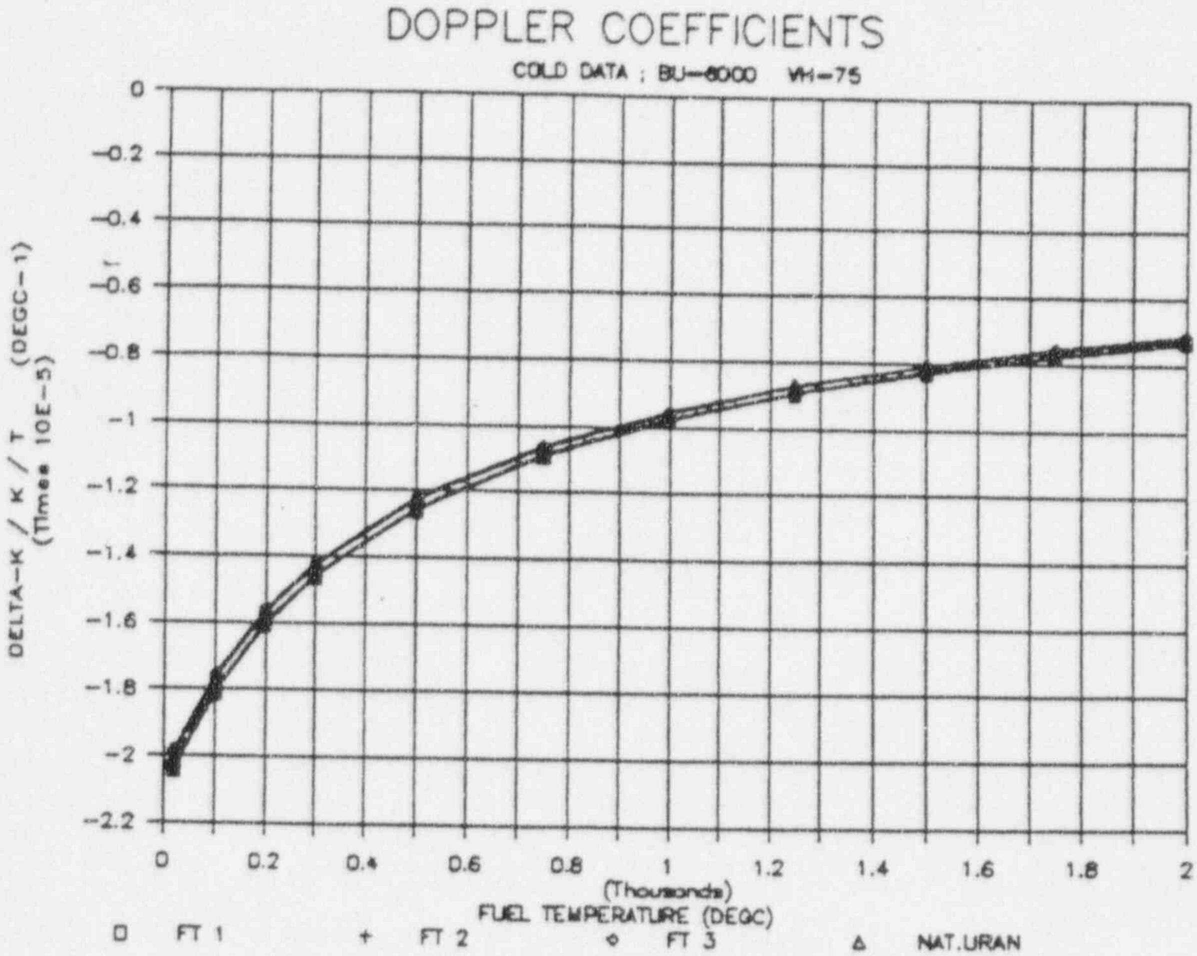

Figure 5.4 Doppler coefficients vs. fuel temperature

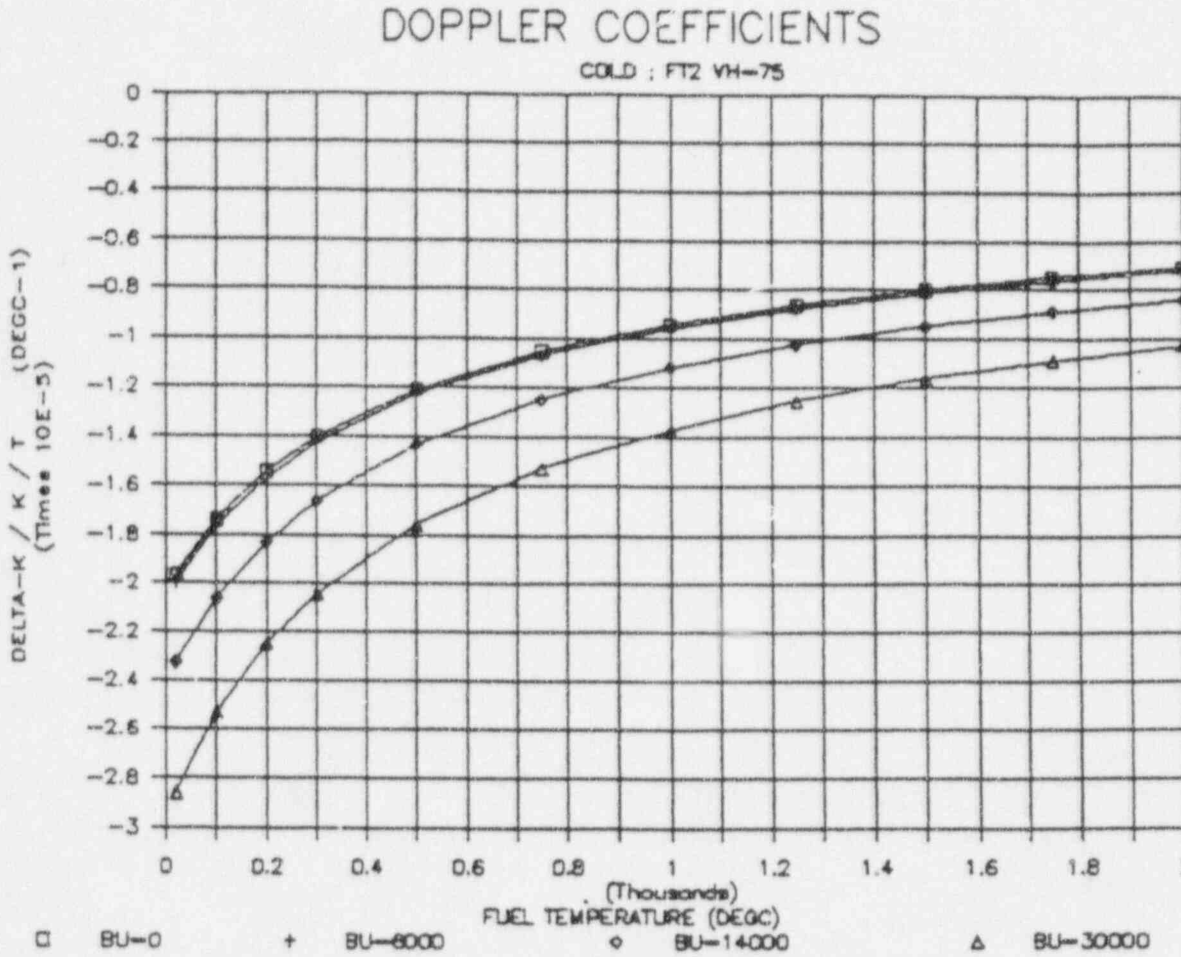

Figure 5.5 Doppler coefficients vs. burnup and fuel temperature

6. Selection of Rod Worths and Accident Initial Conditions

Calculations are carried out for standard GE control rods as a reference, and for the same core containing high-worth control rods only, HWCR.

The calculations were performed at zero power conditions (total power 10^{-8} relative to nominal) in order to minimize the reactivity feedback effects. The coolant temperature was assumed to 20°C and the reactor vessel pressure to 1 bar, i.e. giving a large subcooling of 80°C of the reactor coolant. This state minimizes the moderator density feedback effect, and is thus the most conservative state with respect to a CRDA, given a specified rod worth, Δk.

The transients are analyzed using the same assumptions on maximum rod worth for the standard control rod as in Reference 5 and an assumed rod drop velocity of 3.1 ft/s, which is the maximum drop velocity that could be achieved at normal operating conditions in a series of rod drop tests described in Reference 2. In some sensitivity calculations a drop velocity of 5 ft/s is also used. The HWCR mass and mechanical design are equivalent to that of standard control rods, and thus the drop velocity is also the same.

The full-stroke reactivity worth of the dropped rod was chosen to be about 0.012 for the standard control rod, or more exact 0.01174, to be comparable to the maximum rod worth from the generic analyses of Reference 5, 0.0083 and 0.012 respectively, for normal conditions and conditions with a maximum number of inoperable rods.

For HWCR with an assumed 15% relative higher worth (Δk_∞) the full-stroke rod reactivity increases to 0.01426.

The calculated reactivity shapes of the dropped standard and high-worth control rods are shown in figure 6.1. They were produced by a series of steady-state calculations with RAMONA-3B at the initial conditions with the center control rod at various axial positions and the reactivity worth of that rod taken as the eigenvalue difference relative to the initial eigenvalue.

The channel power distributions at the initial state of the transient for the standard and highworth rods are shown in figure 6.2. The corresponding axial power distributions, average and "hot" channel, are shown in figure 6.3. The average axial distributions are shown together with the distributions in channel 14, which is the channel with the highest power, cf. figure 5.2 (see also figure 7.1.5, page 26).

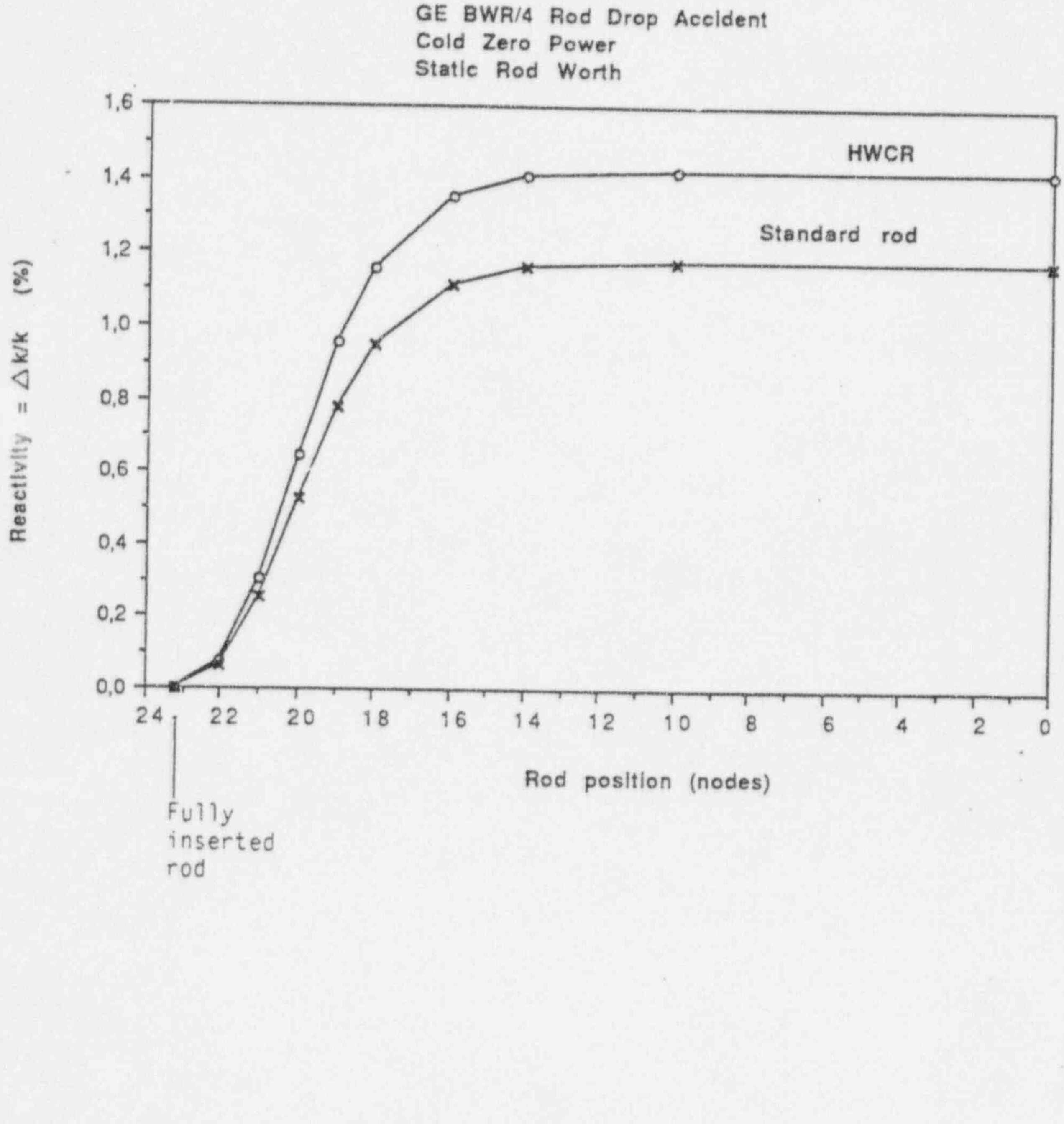

Figure 6.1 Control rod reactivity versus rod position

Standard rods

261	305	269	190	203	187	162	84	46	46	44	20	5
	549	461	228	199	301	216	99	74	81	67	26	7
		372	204	161	238	157	103	53	79	56	23	6
			166	115	103	113	80	51	48	32	13	3
				98	108	82	52	33	35	19	10	2
					127	109	43	38	50	31	12	
						107	51	36	37	19		
							34	16	15			
								6				

HWCR's

258	313	275	182	189	179	154	76	42	43	41	18	5
	609	511	231	194	312	223	95	69	82	69	25	6
		412	206	157	248	163	98	49	79	57	23	6
			160	107	99	107	73	46	45	30	12	3
				89	101	77	47	30	33	18	10	2
					127	110	41	36	51	33	12	
						110	49	34	38	21		
							31	15	15			
								5				

Figure 6.2 CRDA Initial Conditions
Channel Average Normalized Power (%)

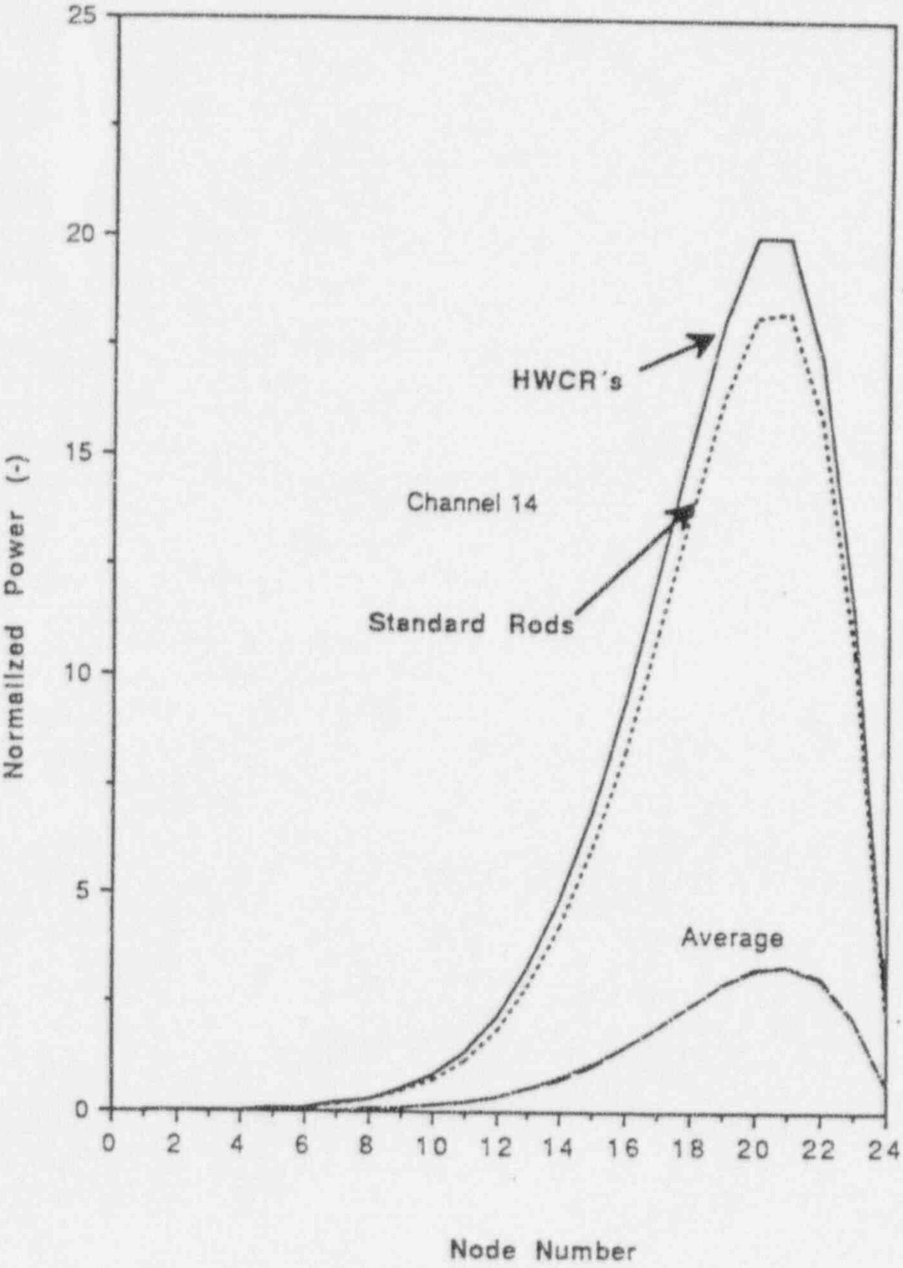

Figure 6.3 Initial conditions.
Hot channel and average axial power distributions.

7. CALCULATIONAL RESULTS

7.1 STANDARD ROD

7.1.1 Initial phase of transient

The drop of the central control rod from fully inserted to a fully withdrawn position with a constant velocity of 3.1 ft/s produces a rapid reactivity insertion to the core. The total reactivity - in units of pcm (10^{-5}) - as calculated from the total fission power using the inverse kinetics equation, is shown in figure 7.1.1 for the standard control rod. Decomposition of the total reactivity into control rods ("scram"), fuel temperature (Doppler) and moderator density ("void") contributions is also shown in figure 7.1.1. (The purpose of the reactivity calculation and its decomposition, included in the extended RAMONA-3B code version, Reference 1, is solely to get a better understanding of the phenomena involved; the fission power calculation is carried out directly from the time-dependent two-group neutron diffusion equation with no need for the "reactivity" concept.)

After about 0.6 s of the transient, when the dropped rod has fallen about 2 feet, or 4 nodes in the model, the reactivity addition exceeds the average β-value, 0.00585, and the reactor goes prompt supercritical. The power burst is interrupted at about 1.0 s by very rapid heating-up of the central parts of the core. The total reactivity drops quickly from a maximum of about 0.01000 to 0.00300. This reduction by 0.00700 can be attributed to mostly Doppler, by about 0.00600, as can be seen from figure 7.1.1, and to a much lower extent - by less than 0.00100 - by a more negative control rod reactivity worth. The latter effect is due to the reduced neutronic importance of the control rod free upper central core region when the fuel heats up and the power distribution is shifted downwards.

The reactivity contribution from moderator heating is small during the first part of the transient, and is insignificant for termination of the power burst. However, its effect is beginning to be noticeable after about 1.0s, see figure 7.1.1. After about 2.9s the coolant is heated to saturation - by direct heat absorption and some heat transfer - at the exit of central fuel channels. Figure 7.1.2, which shows the inlet and outlet flows for the hydraulic channel (channel 2) adjacent to the center channel, gives an indirect indication of the incipient boiling, and figure 7.1.3, showing core average void vs. time is a more direct verification .

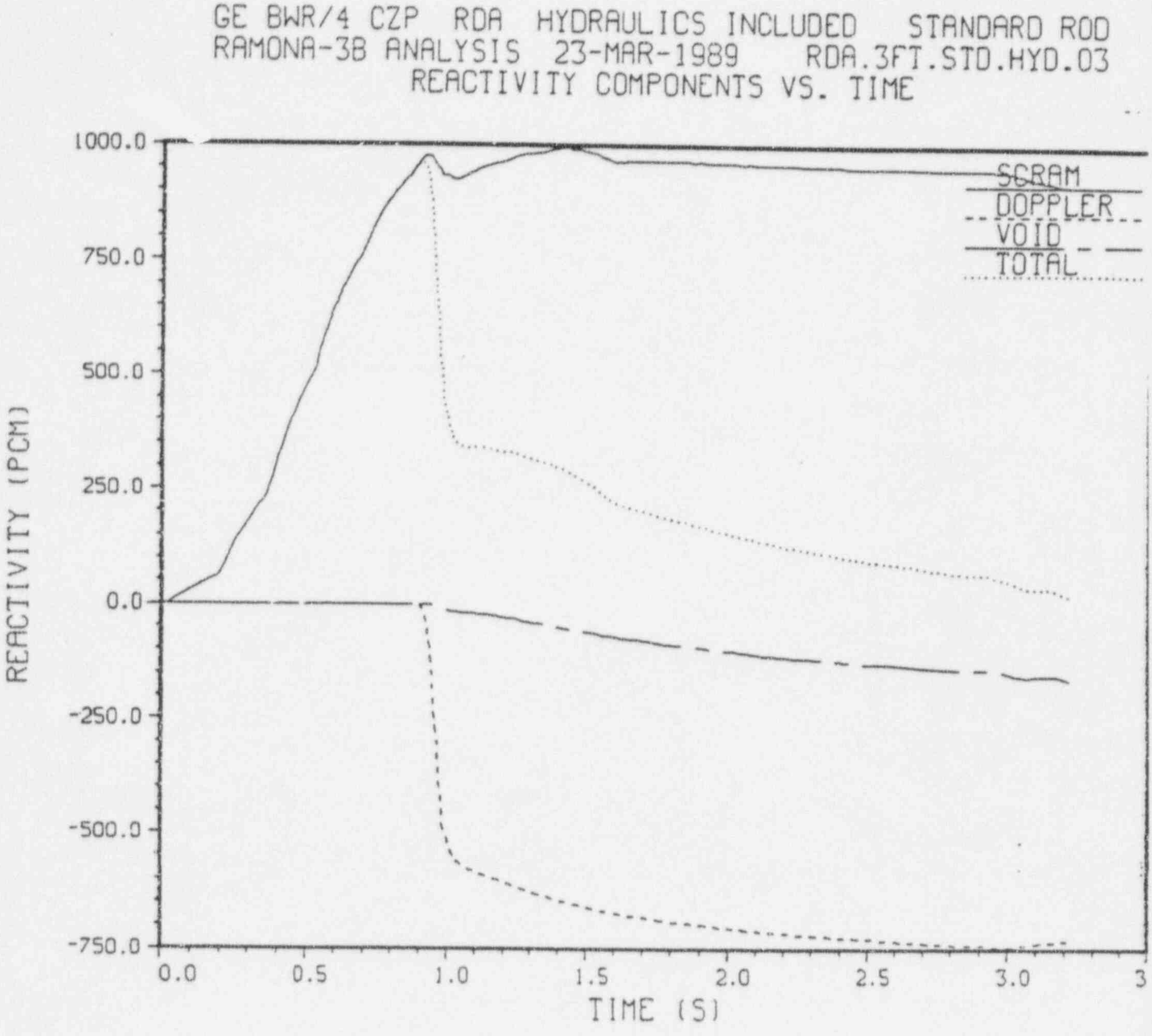

Figure 7.1.1 Standard control rods. CRDA. Rod worth 1.17%
Reactivity components vs. time.

GE BWR/4 CZP RDA HYDRAULICS INCLUDED STANDARD ROD
RAMONA-3B ANALYSIS 23-MAR-1989 RDA.3FT.STD.HYD.03
CHANNEL 2 INLET AND OUTLET FLOW VS. TIME

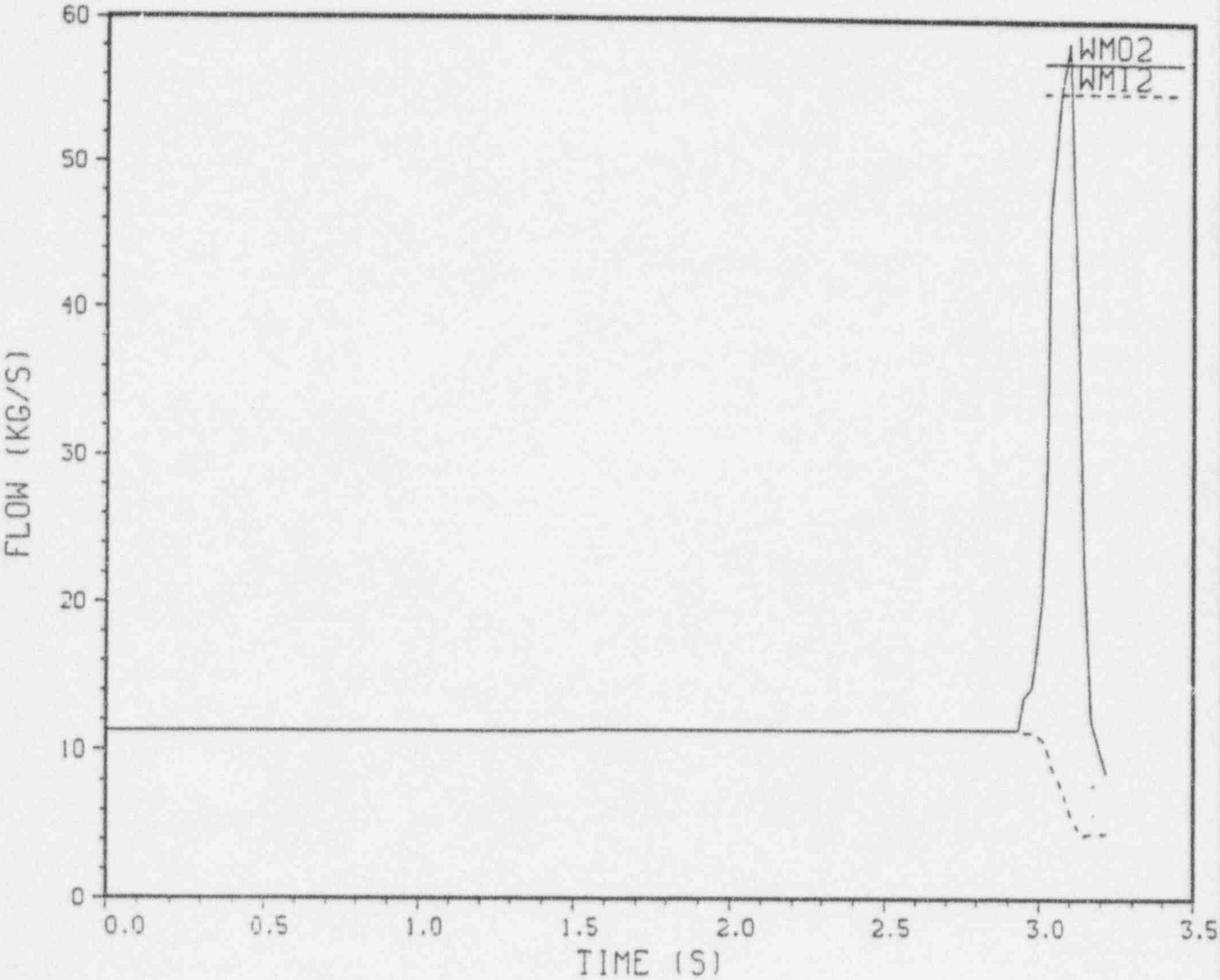

Figure 7.1.2 Standard control rods. CRDA. Rod worth 1.17%.
Hydraulic channel 2 inlet and outlet flow vs. time.
(For channel definition see figure 5.3.)

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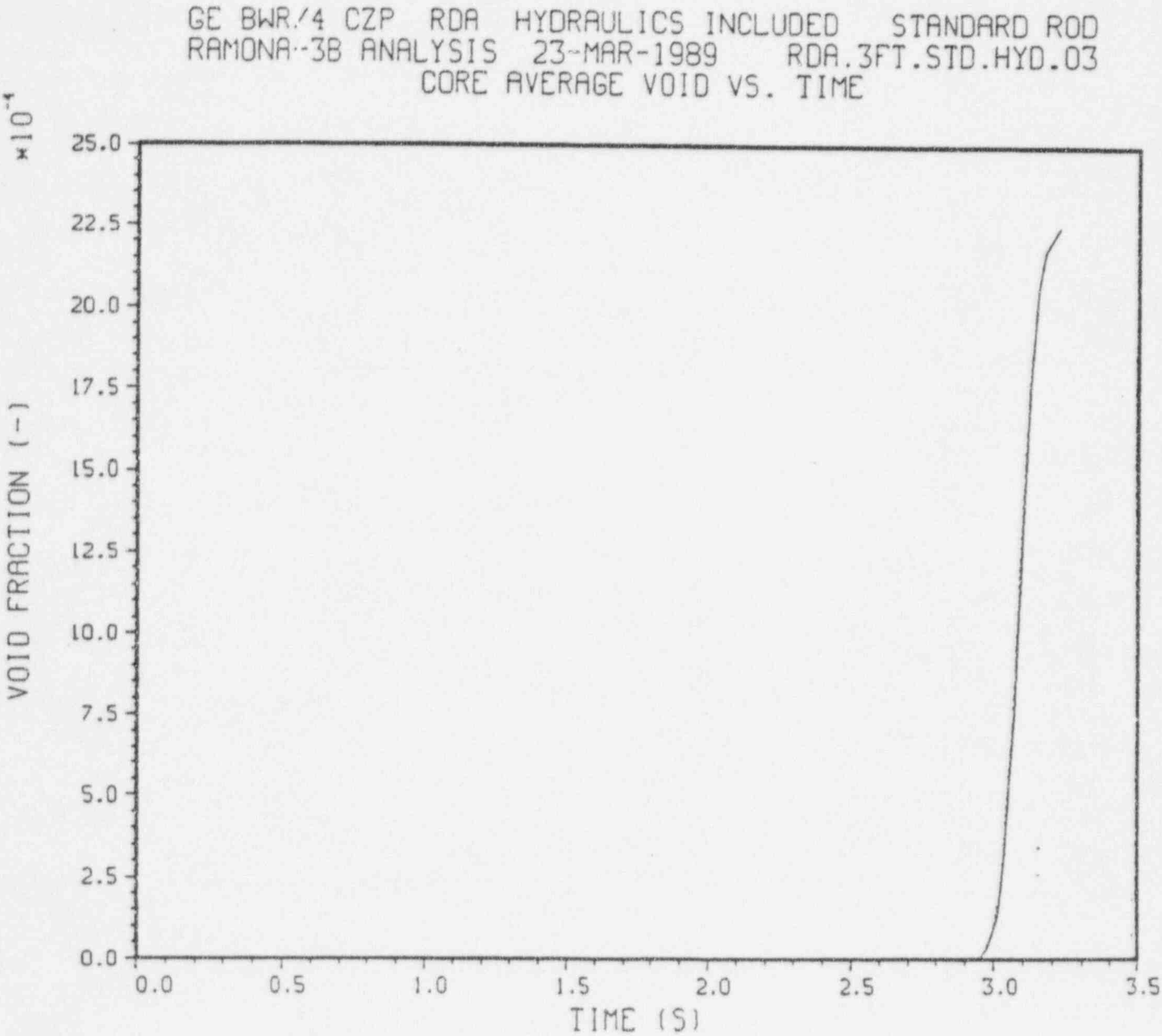


Figure 7.1.3 Standard control rods. CRDA. Rod worth 1.17%
Core average void vs. time.

ABB

The total reactor power increases from 10^{-8} of nominal at the start of the transient to a maximum of about 10 times nominal after 1.0 s of the transient, as shown in figure 7.1.4. The radial (renormalised) power redistribution reached at the time of the power peak, at the axial position where the power is at its maximum (node 21 out of 24), is shown in figure 7.1.5. The distribution is compared with the profile before the onset of the excursion. The figure shows the pronounced initial axial and radial power peaking and also displays the strongly enhanced peaking at the center as a result of the rod drop.

After the power burst, the total power levels off at about 45% of nominal and with a slight negative trend; due to further fuel heatup, and to some extent moderator heating and eventual voiding, the total reactivity - and thus also the power - decreases at the continued course of the transient. One second later the total power is about 25%.

Scram will be initiated early, by the IRM (Intermediate Range Monitor) limit on a low level and by the APRM (Average Power Range Monitor) limit in the power range close to rated power. Since the delay of the IRM signal is usually larger than that of APRM, and particularly at very low power, it may happen that the IRM signal is "superseded" by the 120% power APRM scram limit (or, the prompt flow-biased scram line at slightly lower power, if available) for scramming the reactor. 120% total reactor power is exceeded at about 0.9s of the transient. Assuming a conservative APRM scram delay of 0.5s and a linear control rod insertion of 5s, the reactor will be definitely shut down after 6 to 6.5s of the transient.

In the model used, all scram rods are initially fully withdrawn from the core. Since the power distribution in the core is shifted very much to the top of the core - an effect which is typical for an equilibrium core at start-up conditions - the negative reactivity contribution of scram will not be significant until the scram rods are inserted considerably into the core.

In an actual operating core the scram rod configuration would contain rods at intermediate axial positions. This would lead to a more favourable scram reactivity insertion rate than in the situation reported here , where all scram rods are initially fully withdrawn. Negative scram reactivity from partly inserted rods would be supplied very much earlier in the transient .

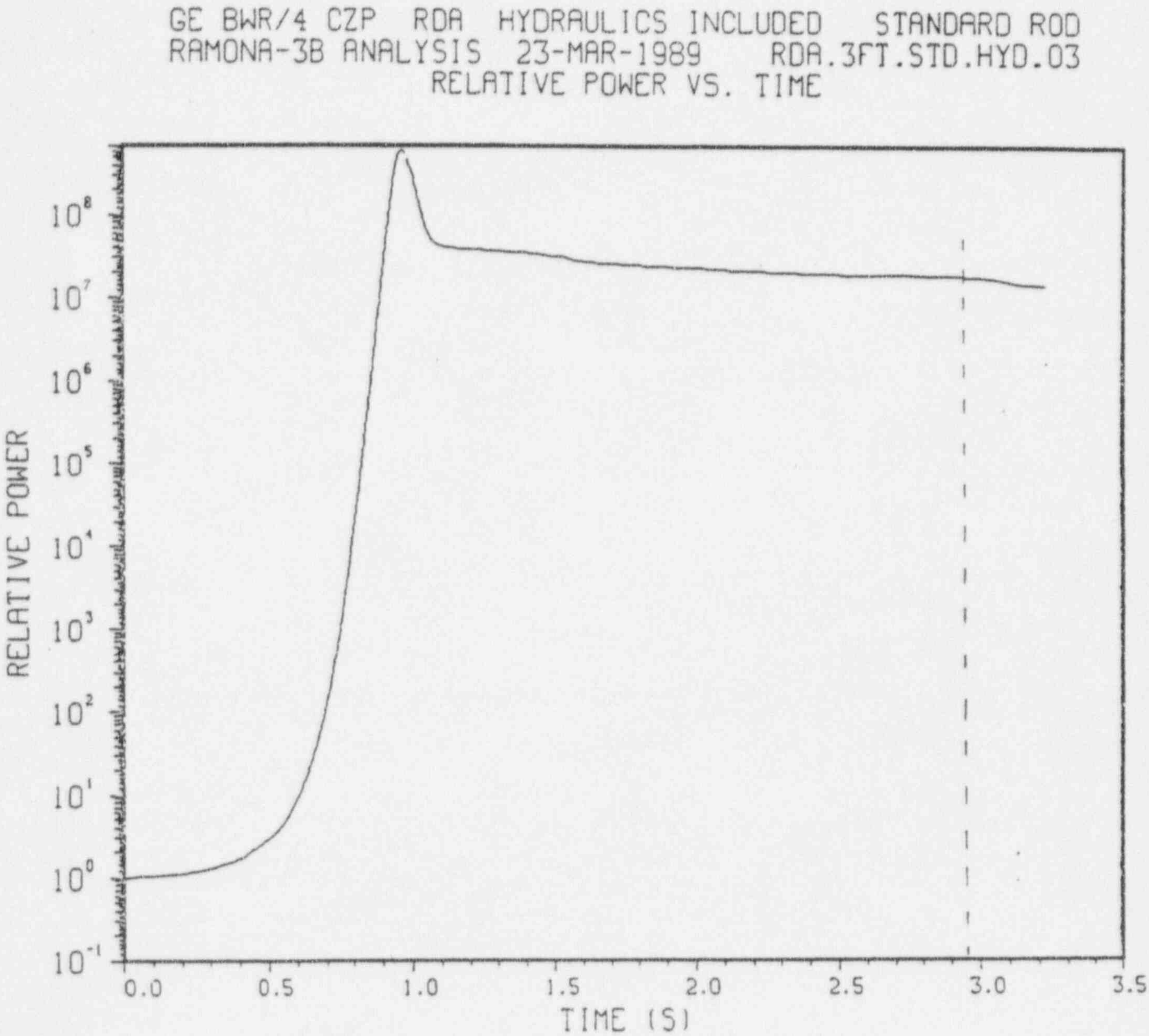
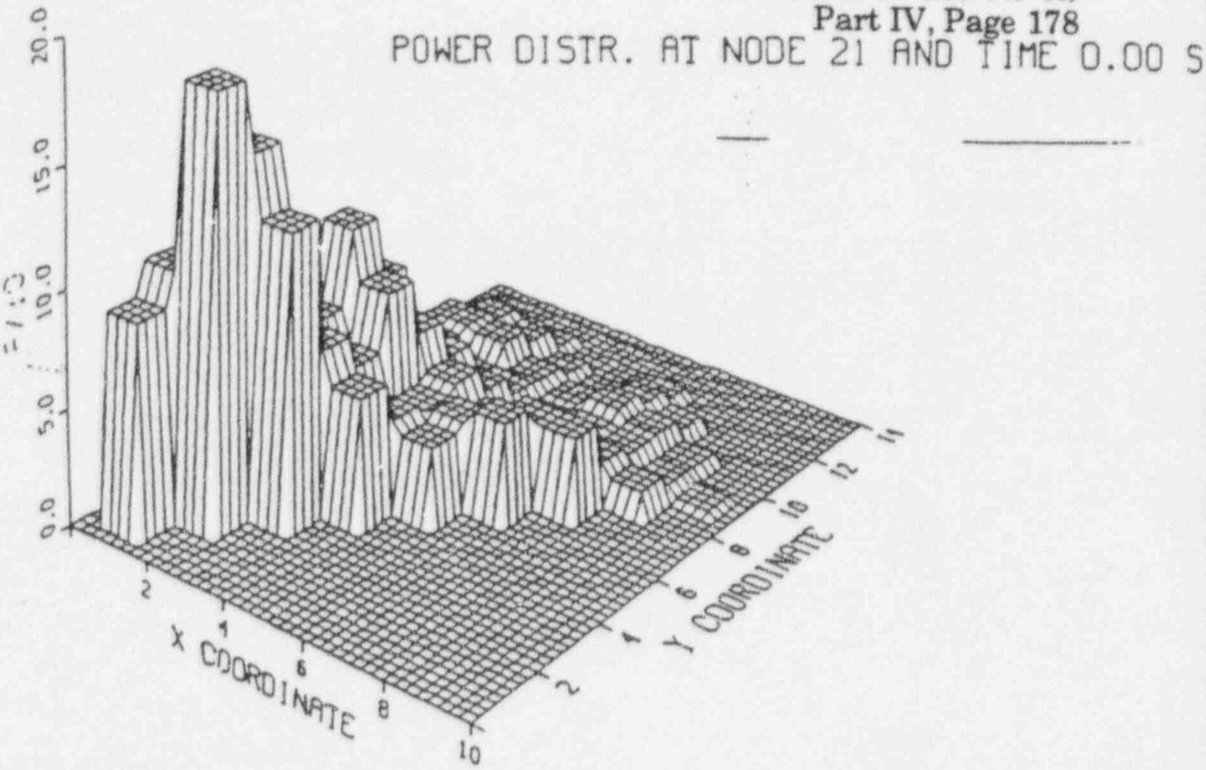


Figure 7.1.4 Standard control rods. CRDA. Rod worth 1.17%
Relative power vs. time.



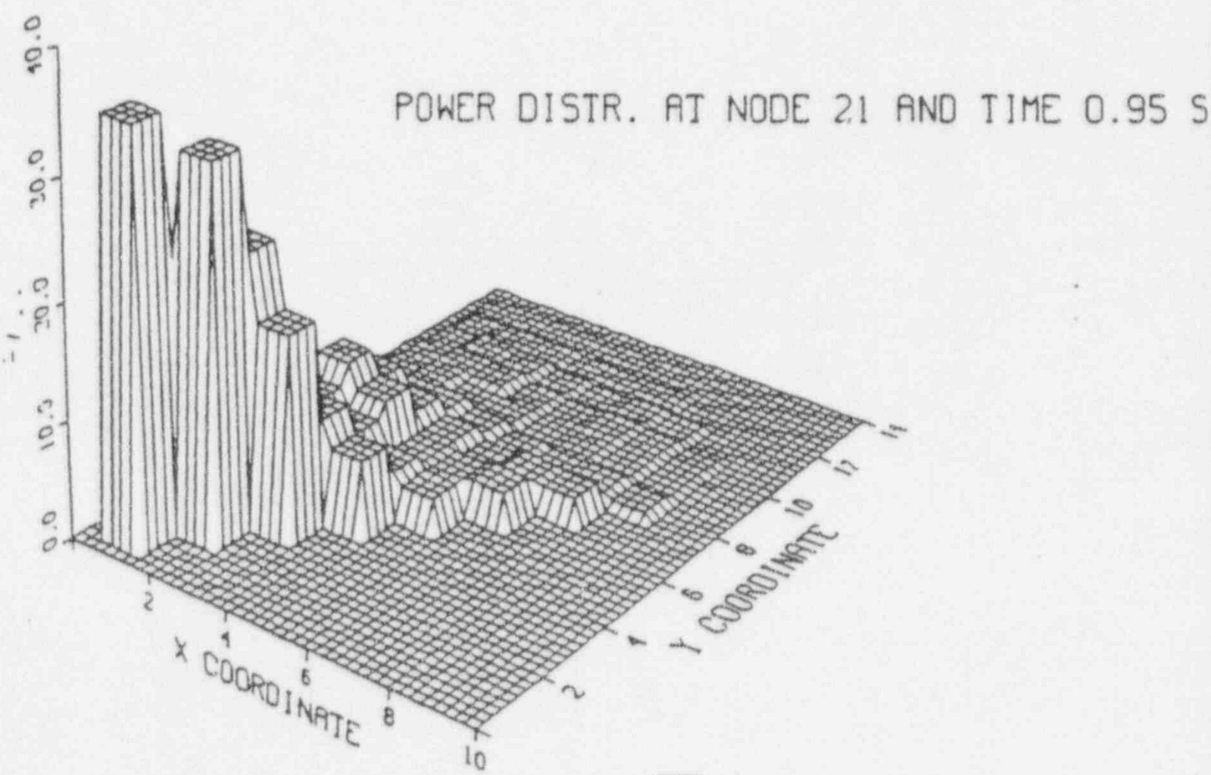


Figure 7.1.5 Standard control rods. CRDA. Rod worth 1.17%.
Power distribution in axial node 21 at time 0 and 0.95s.

ABB

The calculation was terminated at 3.2s into the transient. At that time the reactor power has decreased to roughly 10 % of nominal. The positive reactivity addition from the dropped rod has been completely off-set by the combined negative feedback effects of the fuel temperature rise and moderator density decrease, and scram rod insertion. The peak radial average fuel enthalpy at the end of the calculation is 112 cal/g, see figure 7.1.6, lower curve. The "adiabatic" fuel enthalpy, i.e. the peak enthalpy as evaluated neglecting heat transfer from the fuel (time-integrated power generation), is 150 cal/g, upper curve.

During the continued course of the transient increasingly more negative reactivity will be created from the feedback effects and inserted by scram and the reactor power will decrease accordingly. The enthalpy curve, figure 7.1.6, lower curve, will therefore exhibit values not greater than that at the end of the calculation, 112 cal/g, with a substantial margin to the threshold for fuel failure, 170 cal/g.

7.1.2
Calculation with moderator density feedback effects excluded.
Final phase of transient.

The result of the calculation described above can be compared with that of an analysis of exactly the same rod drop situation but where the effect of changing moderator density was neglected. This analysis was, due to its reduced degree of computational complexity, followed until scram rod insertion is completed.

Figures 7.1.7-7.1.9 show the comparisons for peak fuel enthalpy, "adiabatic" enthalpy and relative reactor power, respectively. The maximum enthalpy increases from 112 cal/g to 133 cal/g when moderator density feedback is neglected, figure 7.1.7.

For the "adiabatic" enthalpy no maximum is, of course, reached. As long as there is a trace of fission or decay power released, its time integral will continue to increase. However, the value at the end of the transient, at 6.5s, when all scram rods are inserted, is 215 cal/g. This result is consistent with the reported value in Reference 5 of 232 cal/g, applying to a rod worth of 0.012 and a case with moderator feedback excluded.

Figure 7.1.7 supports the statement that the peak fuel enthalpy value, 112 cal/g, will not increase during the continued course of the transient. Extrapolation of the enthalpy curve from the fist case run, with the curve for the case with moderator feedback excluded as a guide-line,

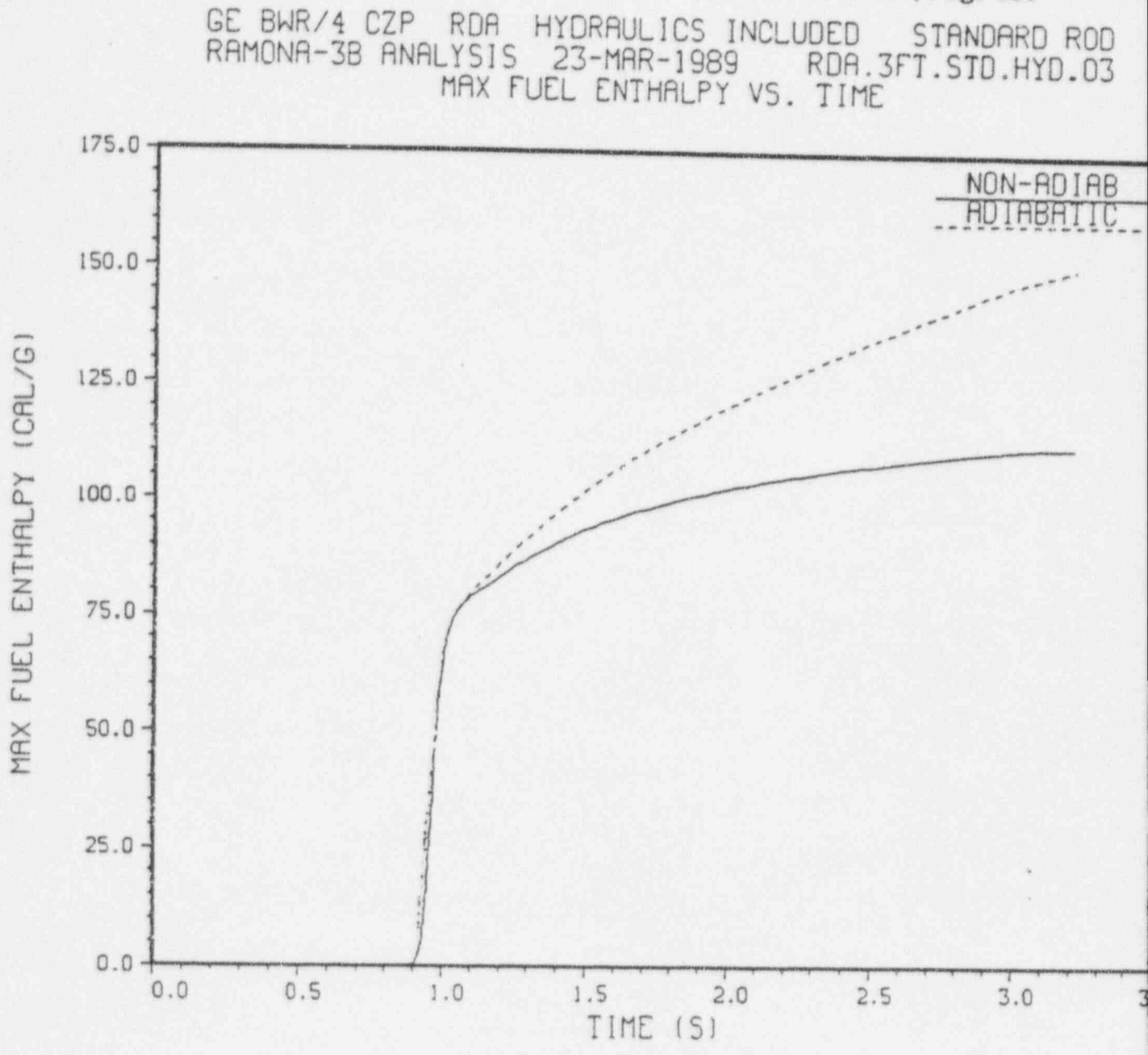


Figure 7.1.6 Standard control rods. CRDA. Rod worth 1.17%.
Peak fuel enthalpy vs. time.

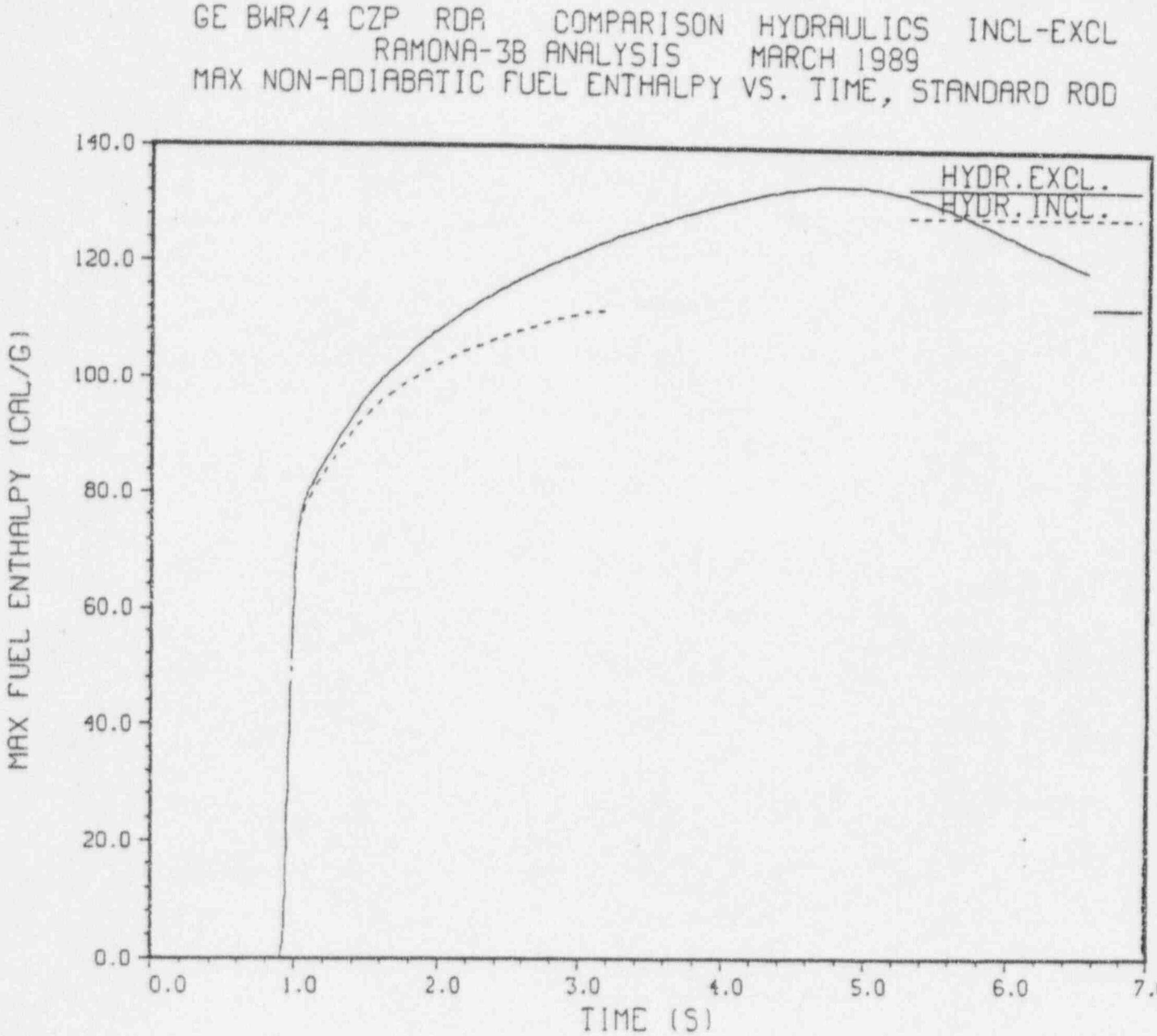


Figure 7.1.7 Standard control rods. CRDA. Rod worth 1.17%.
Comparison: Hydraulics included - excluded.
Peak fuel enthalpy vs. time.

leads to an estimation for the upper limit of the maximum enthalpy not exceeding 112 cal/g. This conclusion is aided by the fact that the incipient boiling in central channels and the subsequent release of negative void reactivity causes the maximum in the enthalpy curve to be reached sooner in the former transient than in the case where the moderator density feedback is not accounted for.

Figure 7.1.8 shows the corresponding comparison of the two cases for the "adiabatic" peak fuel enthalpies. As stated above, the nature of the adiabatic approximation puts no limit on the adiabatic enthalpy during the calculation. However, as discussed in the preceding paragraph, the reduction in the slope of the enthalpy curve will be stronger when moderator density feedback is accounted for and this allows us to suggest an extrapolated upper limit for the adiabatic enthalpy of 180 cal/g.

The results of the calculations are summarized in Table 7.3.1 (page 50), and also in figures 7.2.10 (page 43) and 7.3.1 (page45).

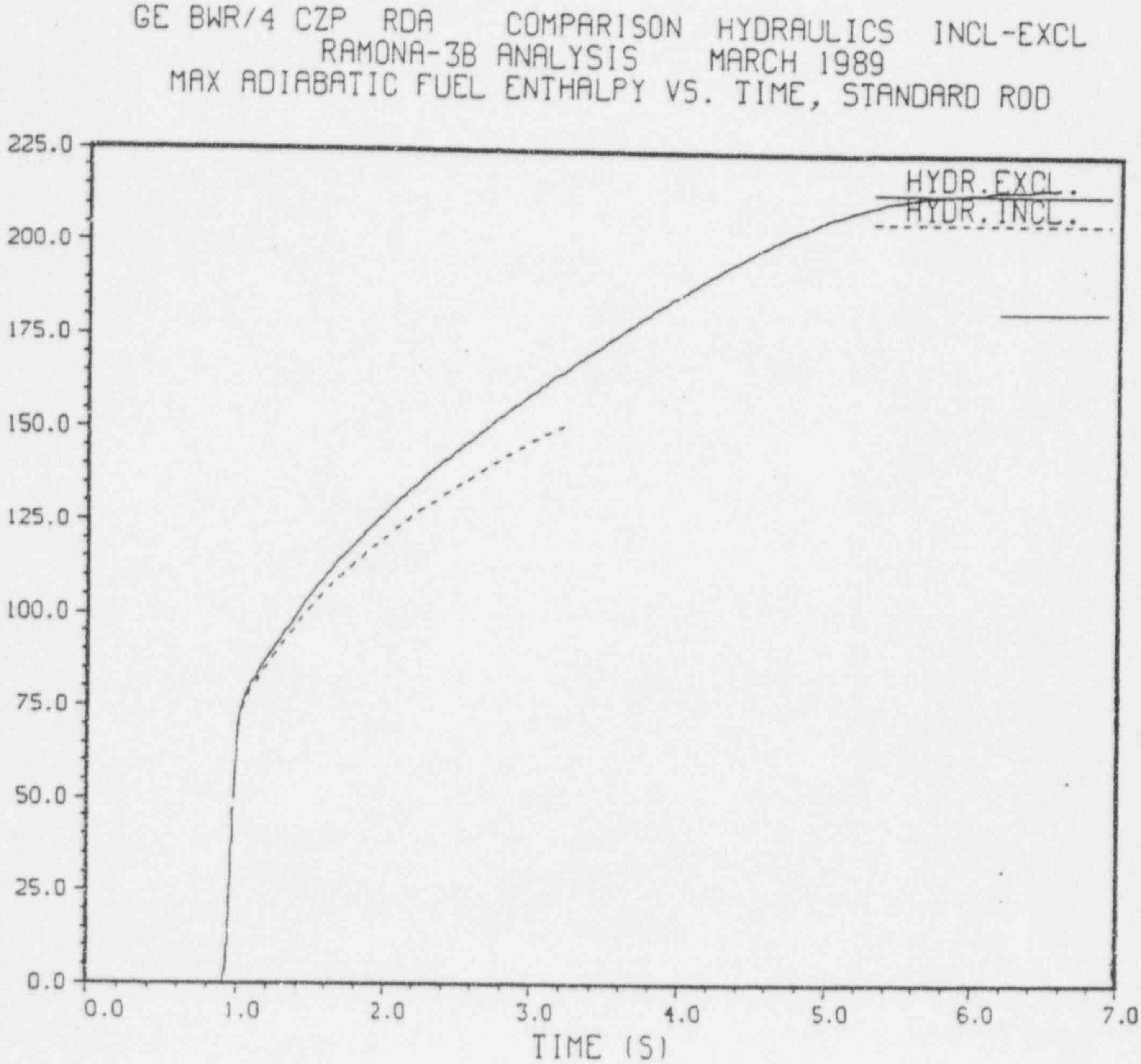


Figure 7.1.8 Standard control rods. CRDA. Rod worth 1.17%.
Comparison: Hydraulics included - excluded.
Peak adiabatic fuel enthalpy vs.time.

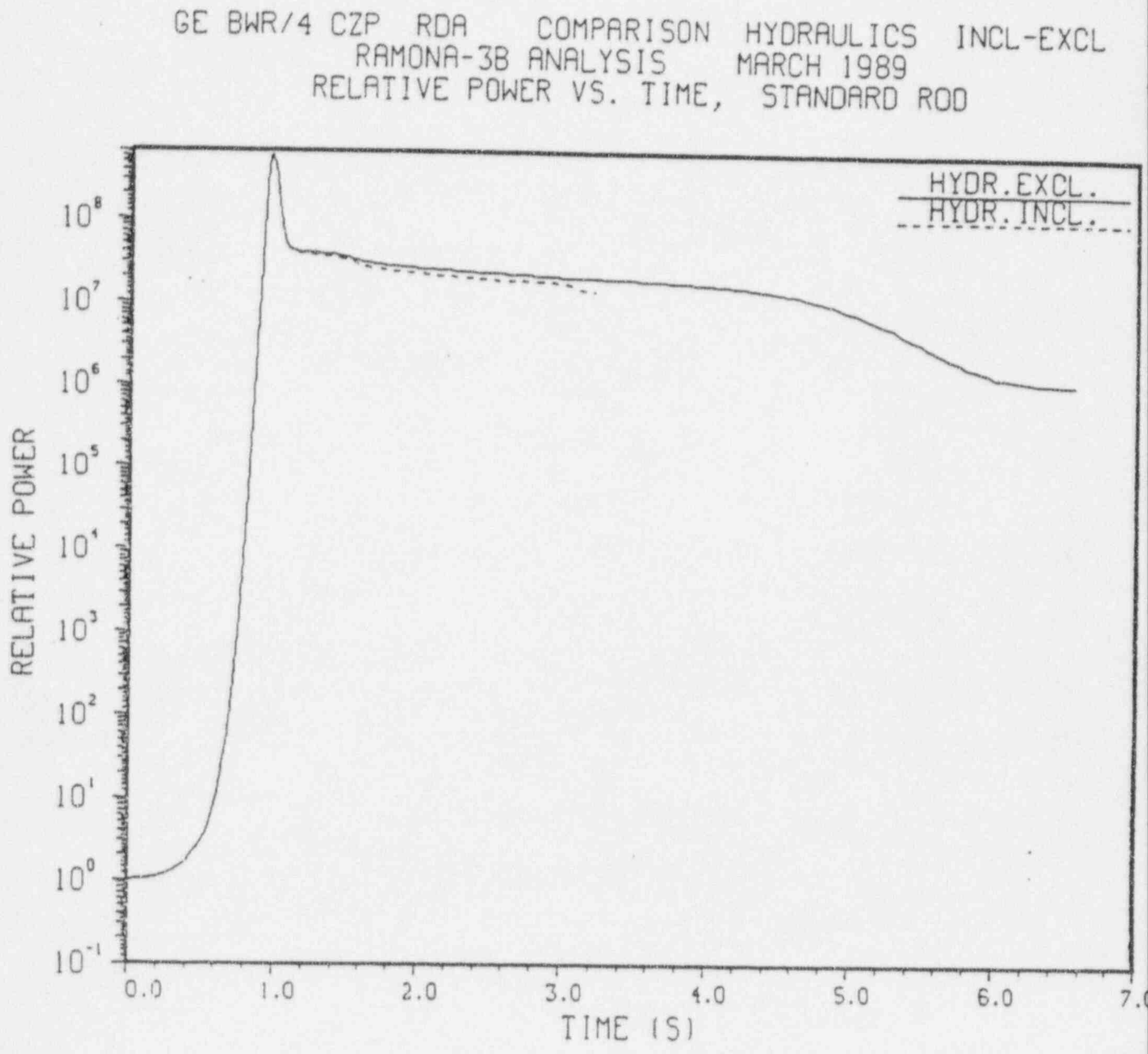


Figure 7.1.9 Standard control rods. CRDA. Rod worth 1.17%.
Comparison: Hydraulics included - excluded.
Relative power vs. time.

7.2 HIGH WORTH CONTROL RODS

7.2.1 Initial phase of transient

The rod drop analysis was repeated for the HWCR core. Here the calculation, with moderator density feedback included, was terminated after 2.3s. Again the positive reactivity from the dropped rod is balanced by Doppler- and moderator density feedback reactivity. Results from this analysis are shown in figures 7.2.1-7.2.5.

At the end of the calculation the peak fuel enthalpy is 140 cal/g. The "adiabatic" enthalpy is 175 cal/g, figure 7.2.5.

7.2.2 Final phase of transient

A calculation with no credit taken for moderator density feedback was also made for the HWCR, see figure 7.2.6-7.2.9 for comparison of results. In analogy with the analysis for the standard control rod core and with reference to figure 7.2.6, a value of 150 cal/g can be assigned to the maximum peak fuel enthalpy for the HWCR core. The maximum peak fuel enthalpy when moderator feedback is excluded is 182 cal/g.

Figure 7.2.7 shows the corresponding comparison for the "abiabatic" peak fuel enthalpies. From this figure, employing the same arguments as for the standard rod core, an upper limit of 230 cal/g is placed on the adiabatic peak fuel enthalpy for the HWCR core.

For convenience, the results from the calculations for peak fuel enthalpy in the comparison between standard rod and HWCR are summarized in figure 7.2.10 , showing the influence of moderator density feedback ("hydraulics" included/excluded) and fuel heat transfer ("adiabatic/non-adiabatic").

The results of the calculations are also summarized in Table 7.3.2 (page 51), and in figure 7.3.1 (page 45).

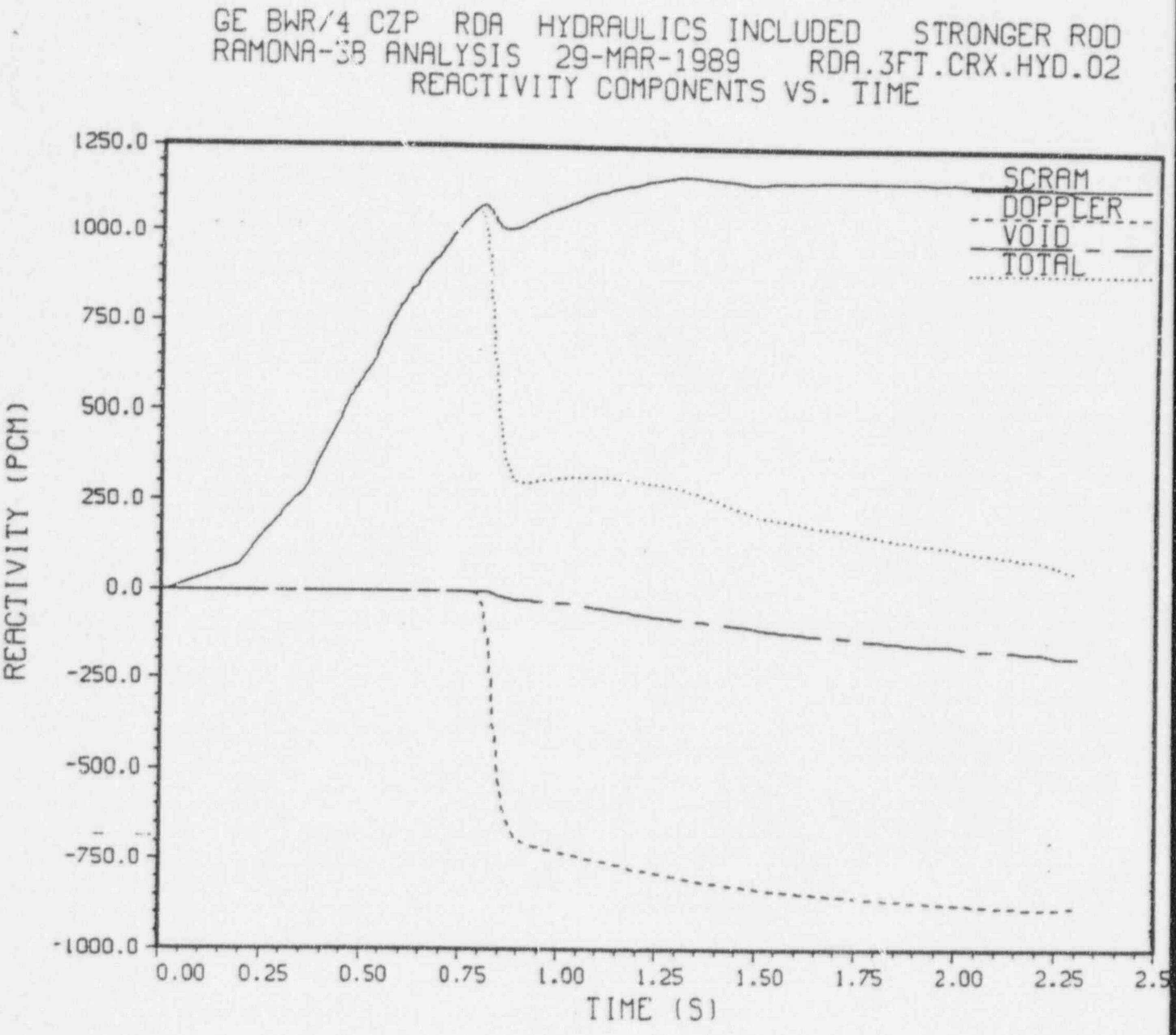


Figure 7.2.1 HWCR's. CRDA. Rod worth 1.43%.
Reactivity components vs. time.

ABB

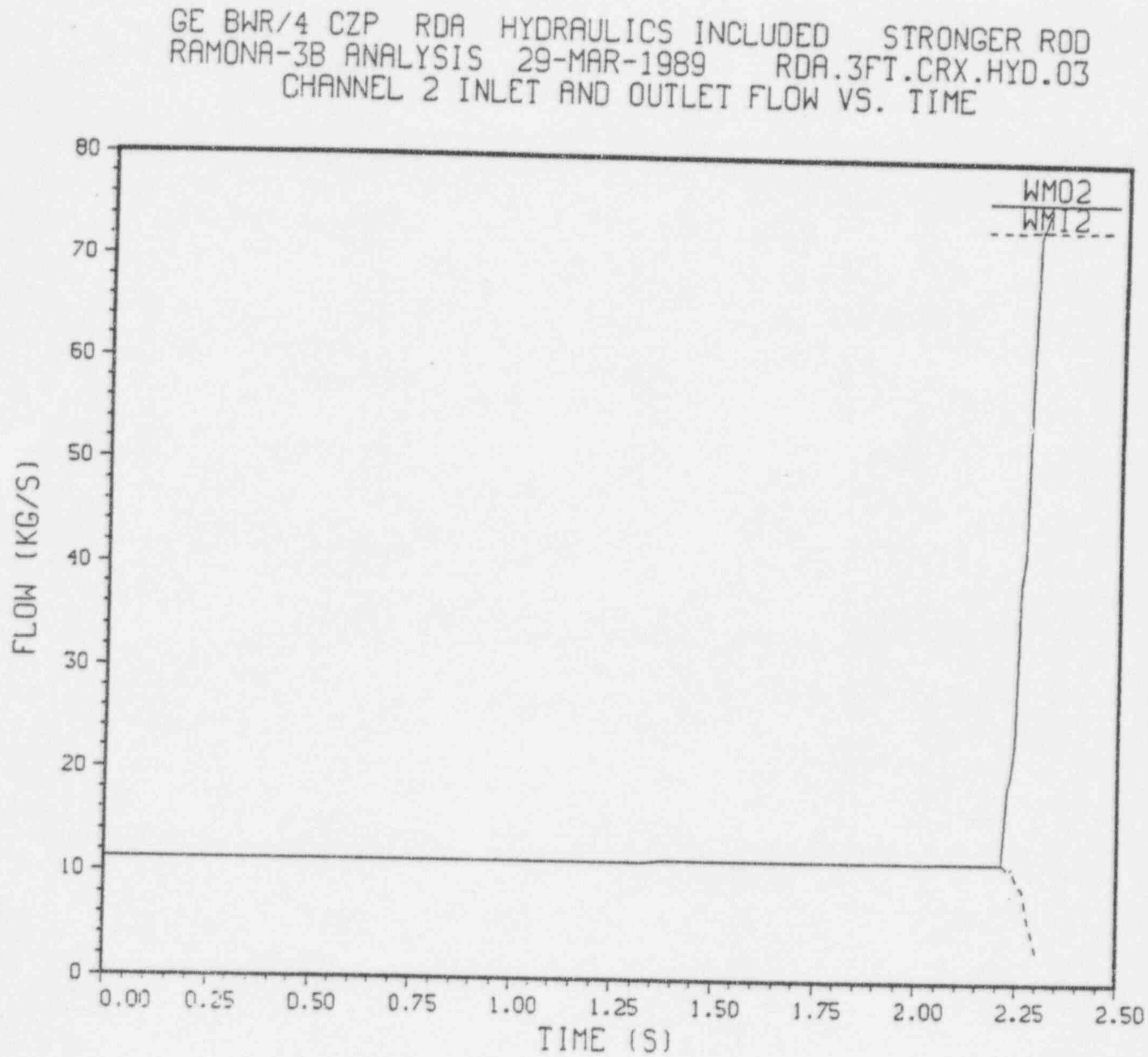


Figure 7.2.2 HWCR's. CRDA. Rod worth 1.43%.
Hydraulic channel 2 inlet and outlet flow.
(For channel definition see figure 5.3.)

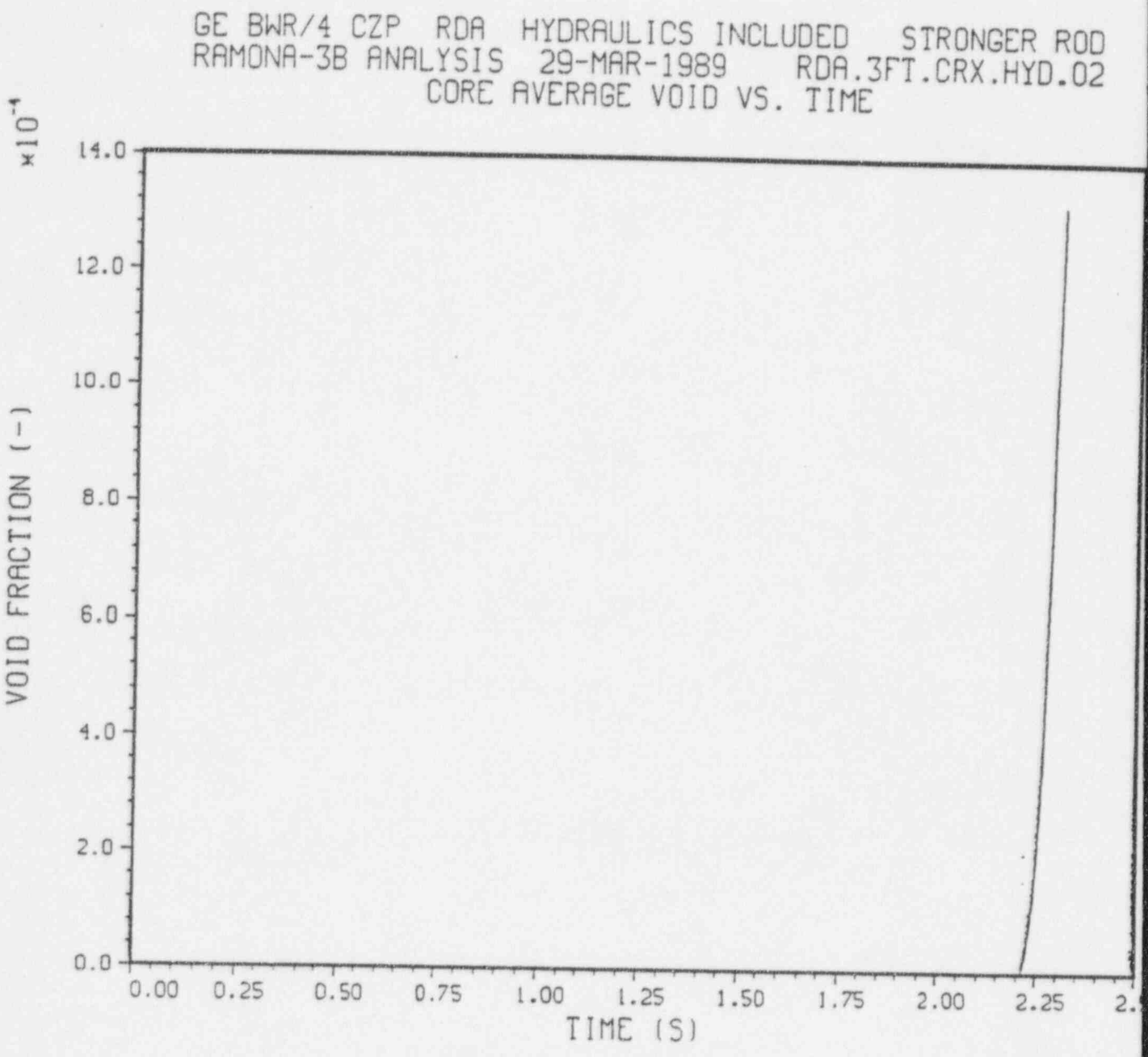


Figure 7.2.3 HWCR's. CRDA. Rod worth 1.43%.
Core average void vs. time.

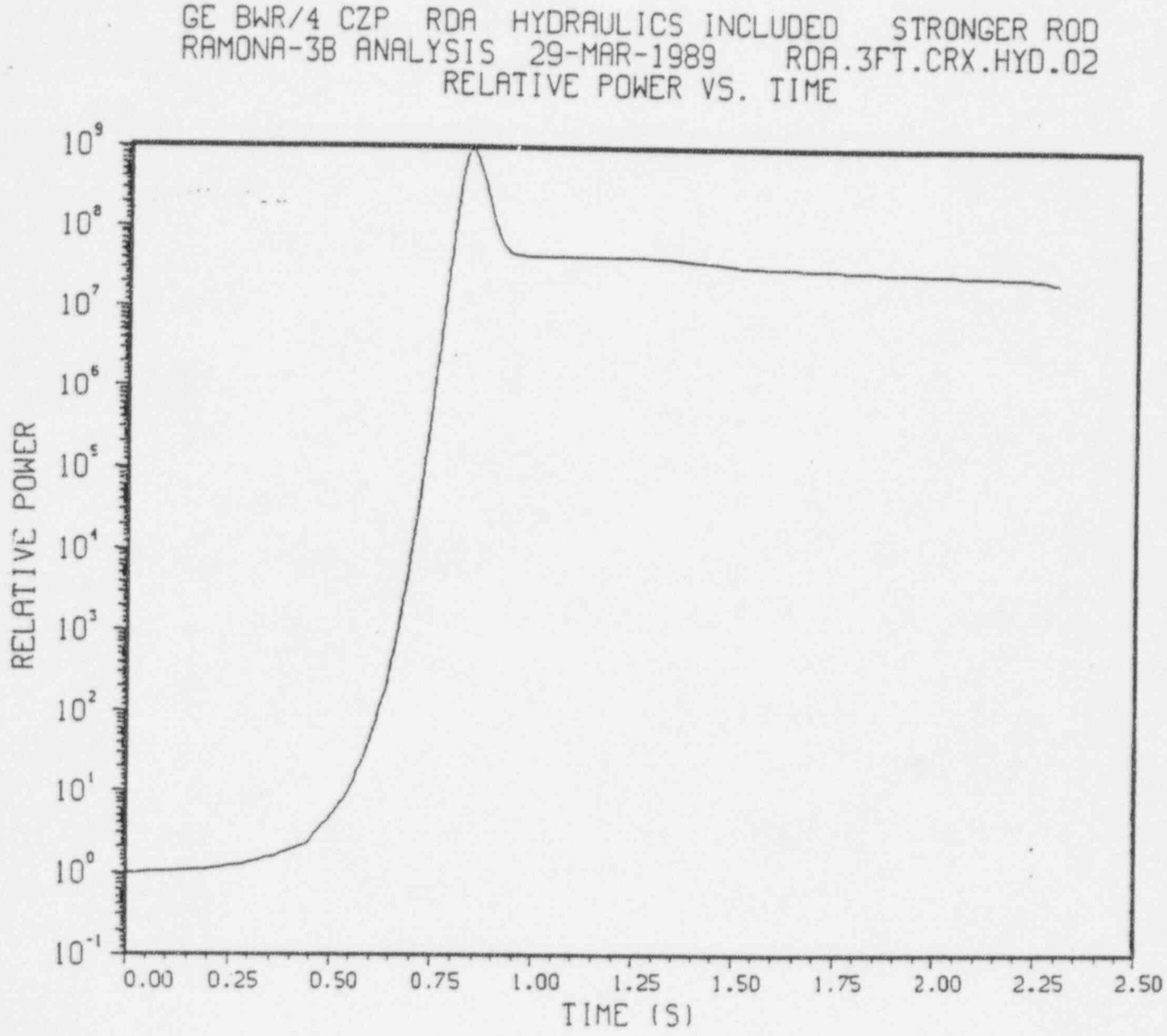


Figure 7.2.4 HWCR's. CRDA. Rod worth 1.43%.
Relative power vs. time.

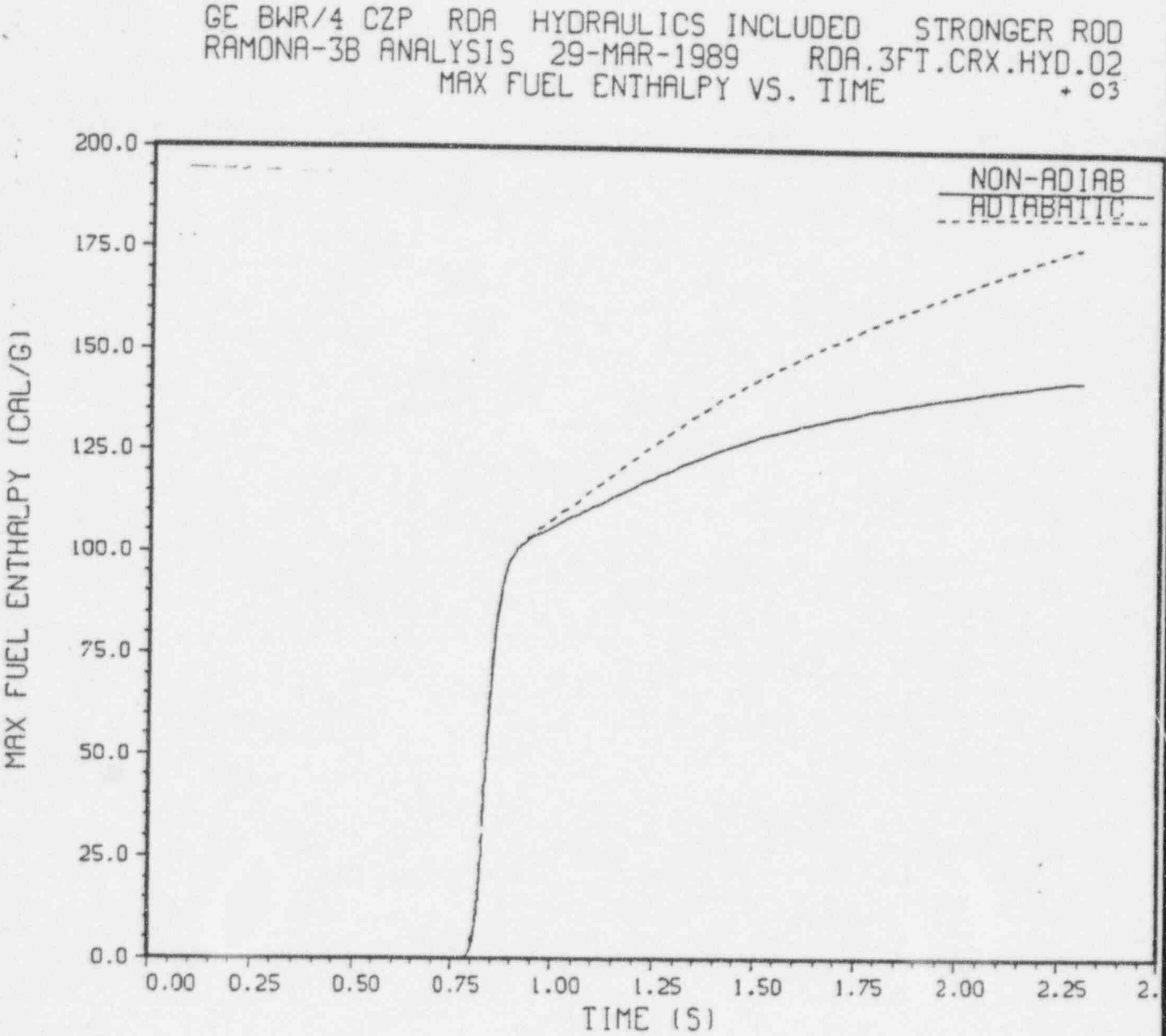


Figure 7.2.5 HWCR's. CRDA. Rod worth 1.43%.
Peak fuel enthalpy vs. time.

GE BWR/4 CZP RDA HWCR COMPARISON HYDR. INCL - EXCL
RAMONA-3B ANALYSIS MARCH 1989
MAX NON-ADIABATIC FUEL ENTHALPY VS. TIME

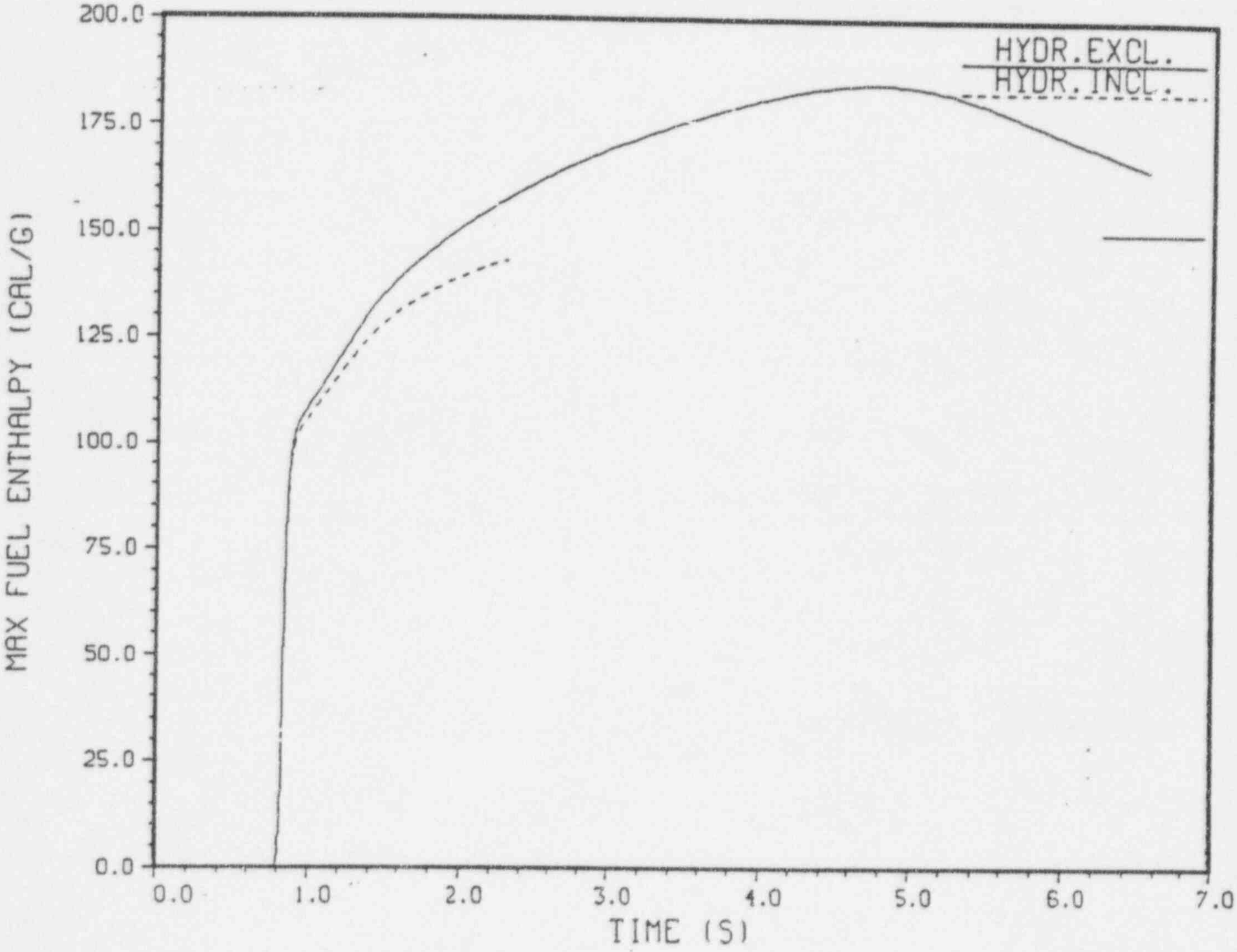


Figure 7.2.6 HWCR's. CRDA. Rod worth 1.43%.
Comparison: Hydraulics included - excluded.
Peak fuel enthalpy vs. time.

ABB

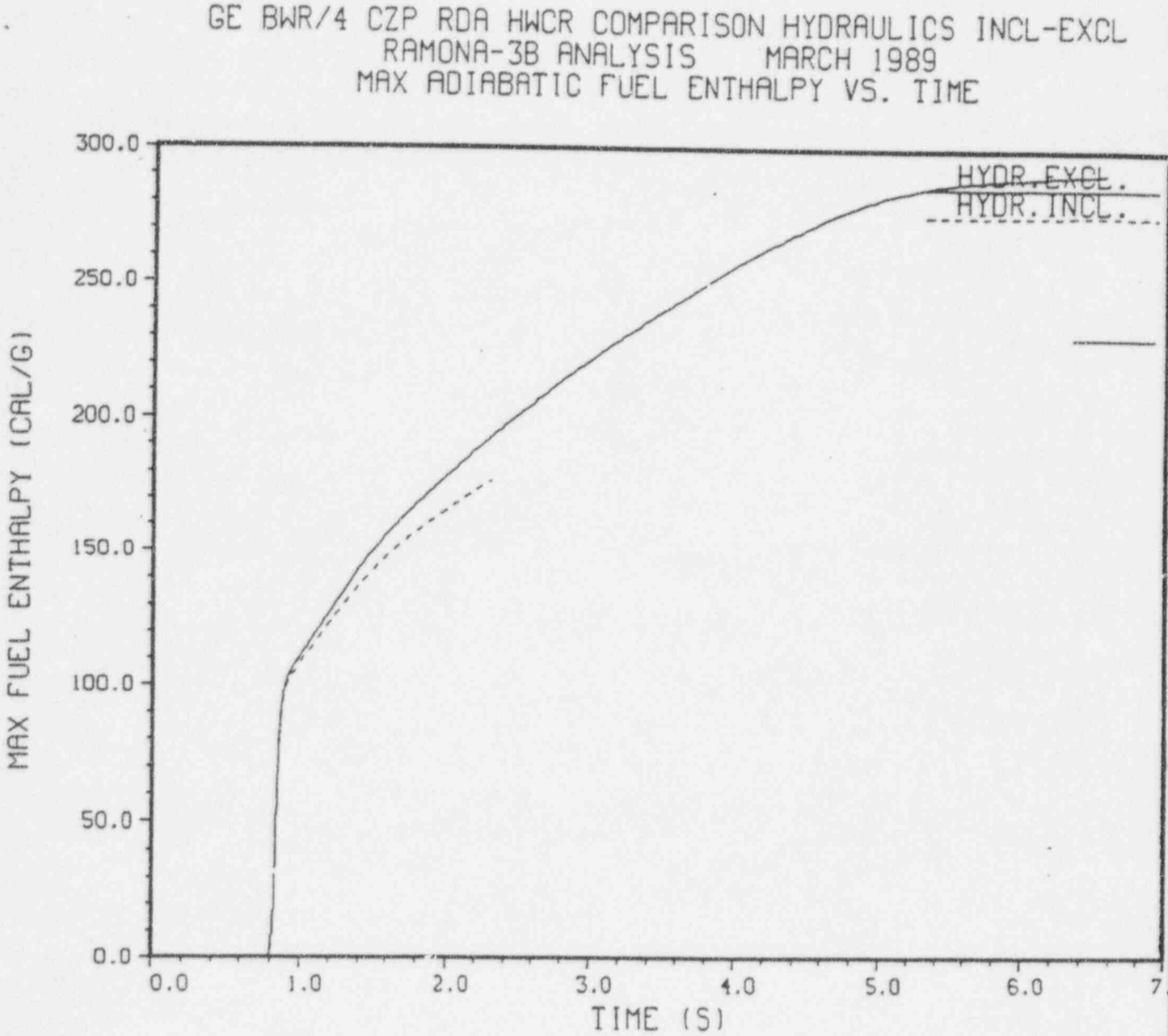


Figure 7.2.7 HWCR's. CRDA. Rod worth 1.43%.
Comparison: Hydraulics included - excluded.
Peak adiabatic fuel enthalpy vs. time.

ABB

GE BWR/4 CZP RDA HWCR COMPARISON HYDRAULICS INCL-EXCL
RAMONA-3B ANALYSIS MARCH 1989
RELATIVE POWER VS. TIME

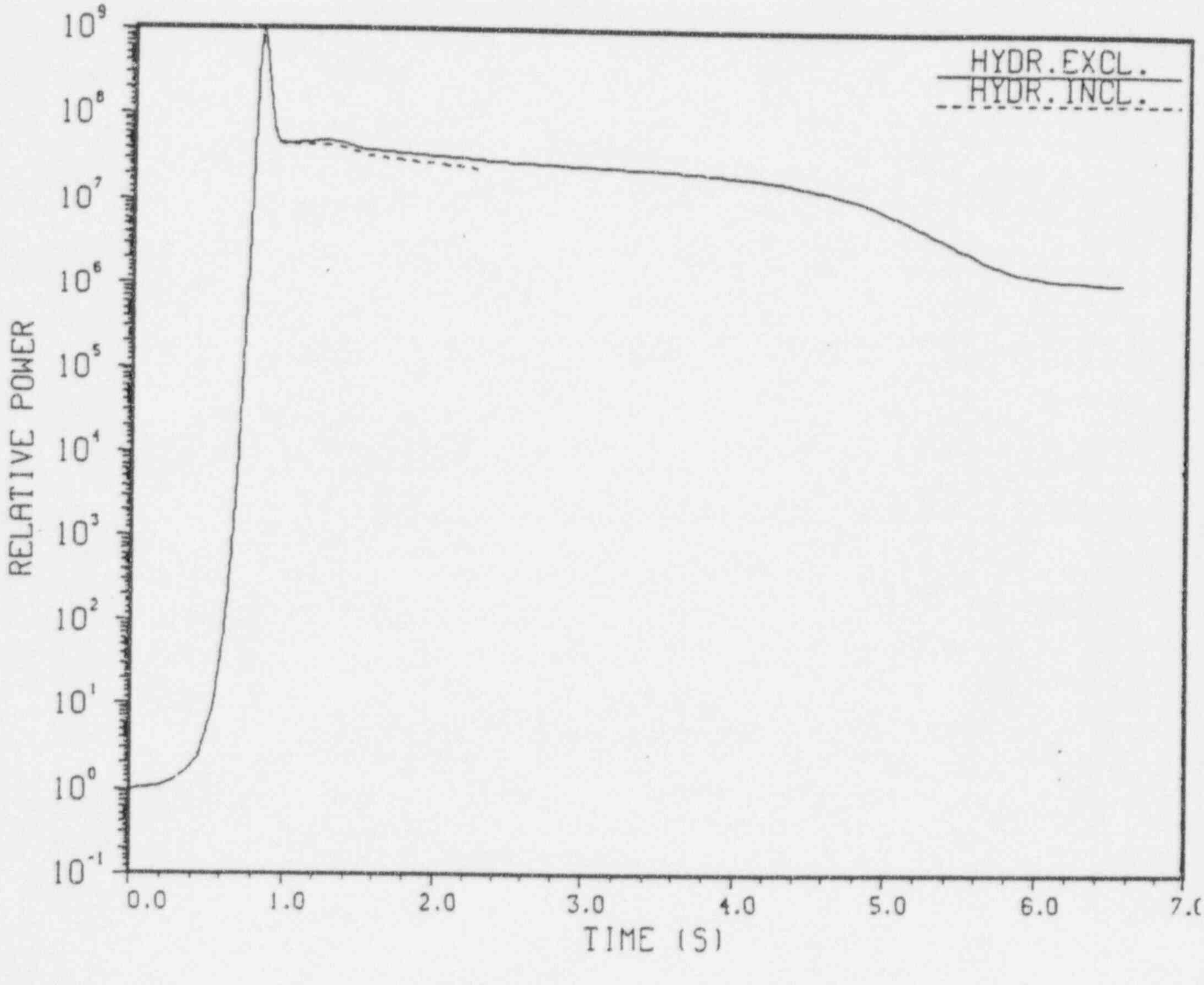


Figure 7.2.8 HWCR's. CRDA. Rod worth 1.43%.
Comparison: Hydraulics included - excluded.
Relative power (logarithmic) vs. time.

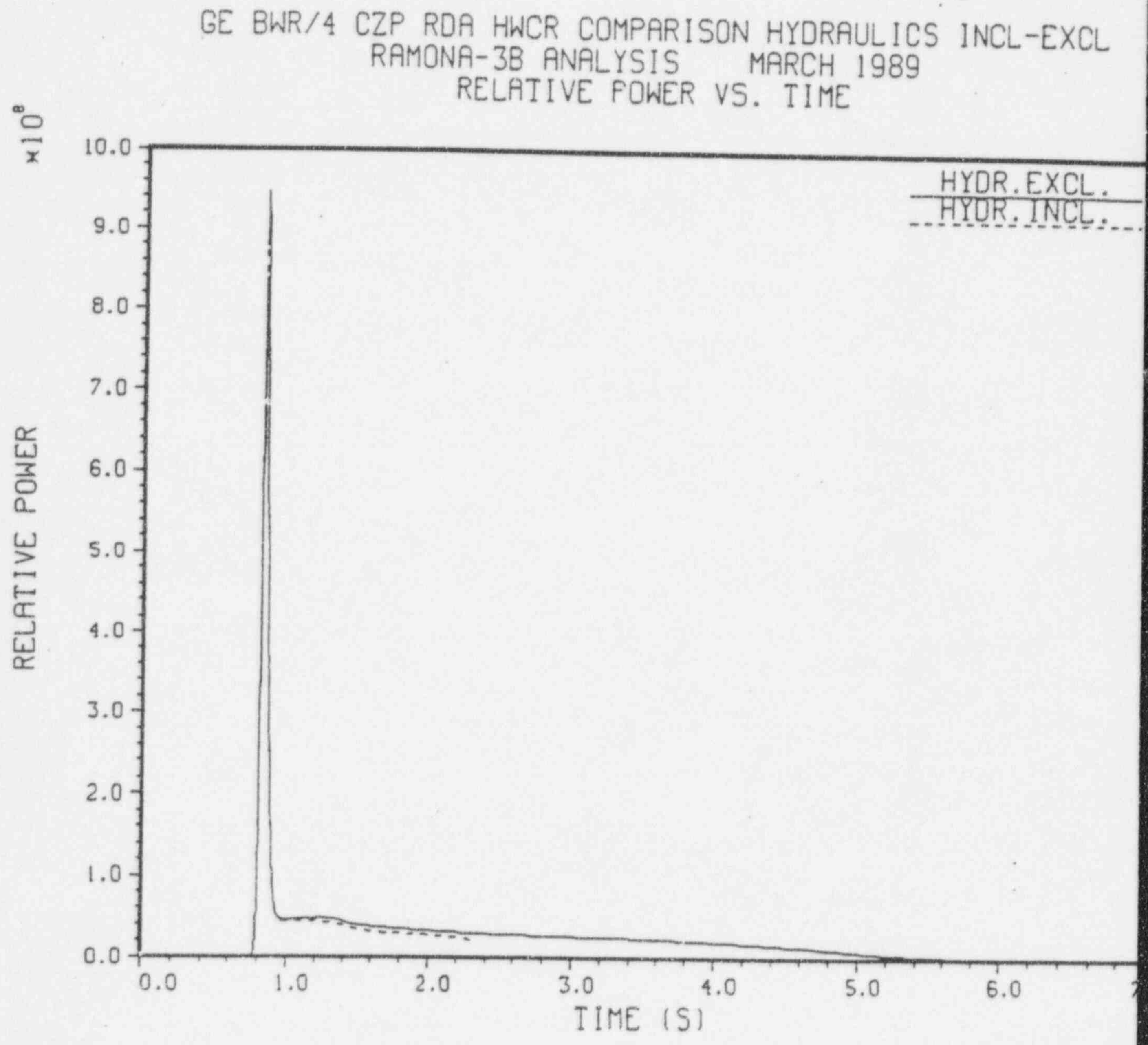


Figure 7.2.9 HWCR's. CKDA. Rod worth 1.43%.
Comparison: Hydraulics included - excluded.
Relative power vs. time.

ABB

GE BWR/4 CZP RDA RAMONA-3B ANALYSIS MARCH 1989
MAX. FUEL ENTHALPY VS. TIME , STANDARD ROD AND HWCR
MODERATOR DENSITY FEEDBACK AND HEAT TRANSFER EFFECTS

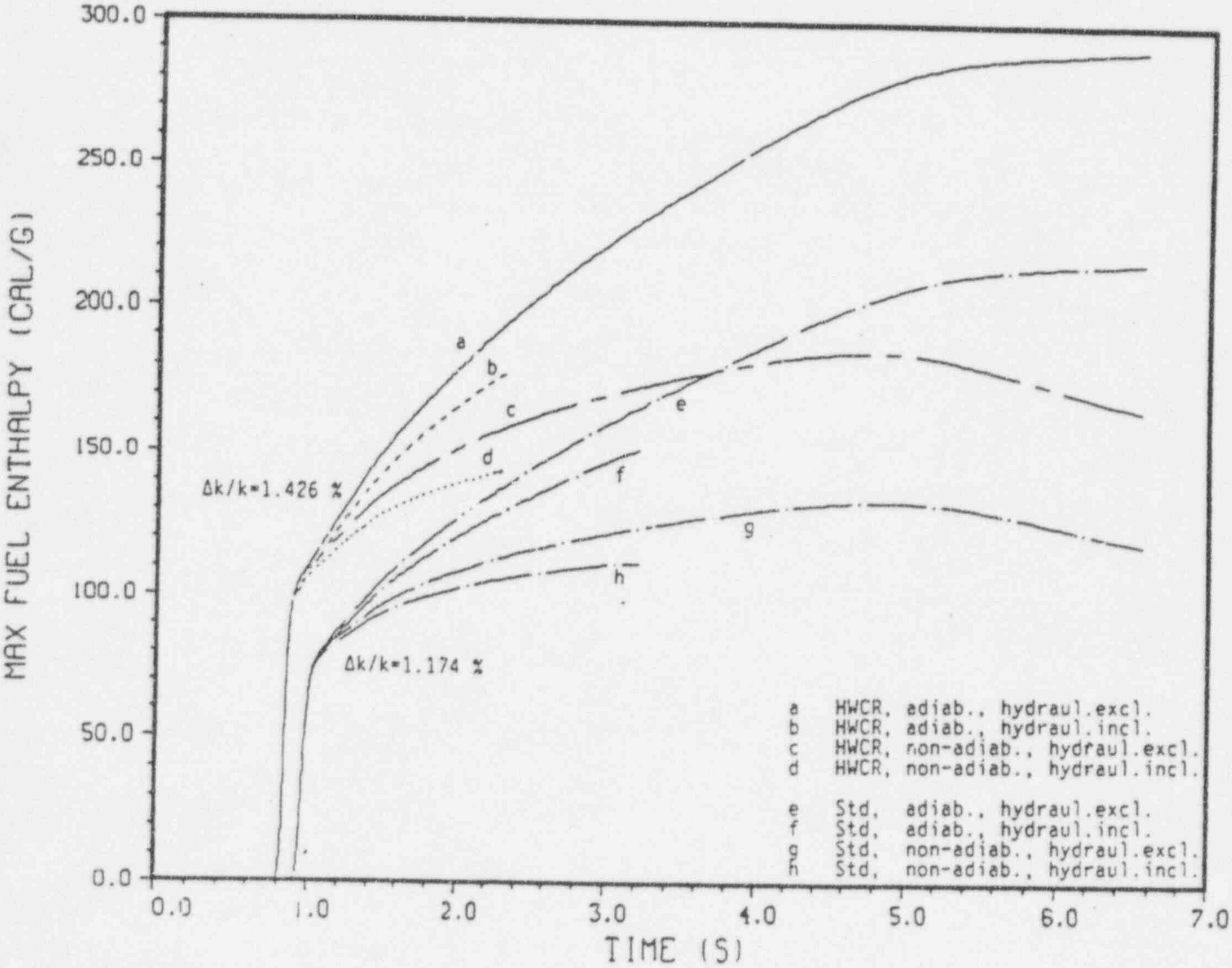


Figure 7.2.10 Standard control rods and HWCR´s.
Moderator density feedback and heat transfer effects.

7.3 FURTHER CRDA CALCULATIONS
Sensitivity and consistency studies

As a check of the consistency of rod drop calculations figure 7.3.1 is a display of results from the calculations discussed above and from a number of RAMONA 3B - calculations not specifically discussed otherwise in this report. These calculations differ in various prerequisites, briefly accounted for in figure 7.3.1, which mainly aims at showing the correlation between rod worth and peak fuel enthalpy. The calculations marked 1 through 6 were all done with moderator density feedback neglected and the reported maximum enthalpy refers in each case to the "adiabatic" value (i.e no heat transfer) at the end of the transient when all scram rods have been fully inserted and the increase rate in integrated power generation is low.

As an example, some results from a typical calculation, calculation number 1 in figure 7.3.1, are shown in figures 7.3.2-7.3.4. Here, the control rod density was a 50% pattern (which in practice could never be used at cold critical conditions), see figure 7.3.5, and the rod drop velocity was 5 ft/s . The static rod worth was 0.01166. The peak non-adiabatic enthalpy following the CRDA was 144 cal/g and the peak adiabatic enthalpy was 230 cal/g.

For comparison with the RAMONA-3B results, two results from the calculations in Reference 2-5 (rod worth 0.0083 and 0.012, marked GE), where the void feedback was also neglected, have been included in figure 7.3.1.

The peak enthalpies from this group of calculations, including the referenced values, constitute, as expected, a fairly linear correlation with dropped rod worth.

In the diagram are also included approximate maximum "adiabatic" enthalpy values for the calculations (marked 7 and 8) of sections 7.1 and 7.2, where the effect of moderator density feedback is accounted for. The values presented are extrapolations of the curves of figures 7.1.8 and 7.2.7, respectively.

It can be noted that among the different rod drop cases analyzed in Reference 6 there is one which can be compared with case 7 of this report, namely a hot zero power case with an imposed large inlet subcooling of 56ºC (case 13). The rod worth in that case is 1.192% and the calculation is terminated at 2.0s. Including the effect of differing delayed neutron fractions and rod drop velocities, quantified in Reference 6, a comparison of the peak enthalpy values at the relevant instance of the transient, 1.55s after the onset of the power excursion, yields a good agreement.

ABB

PEAK ENTHALPY vs. ROD WORTH
Adiabatic values; no heat transfer credited

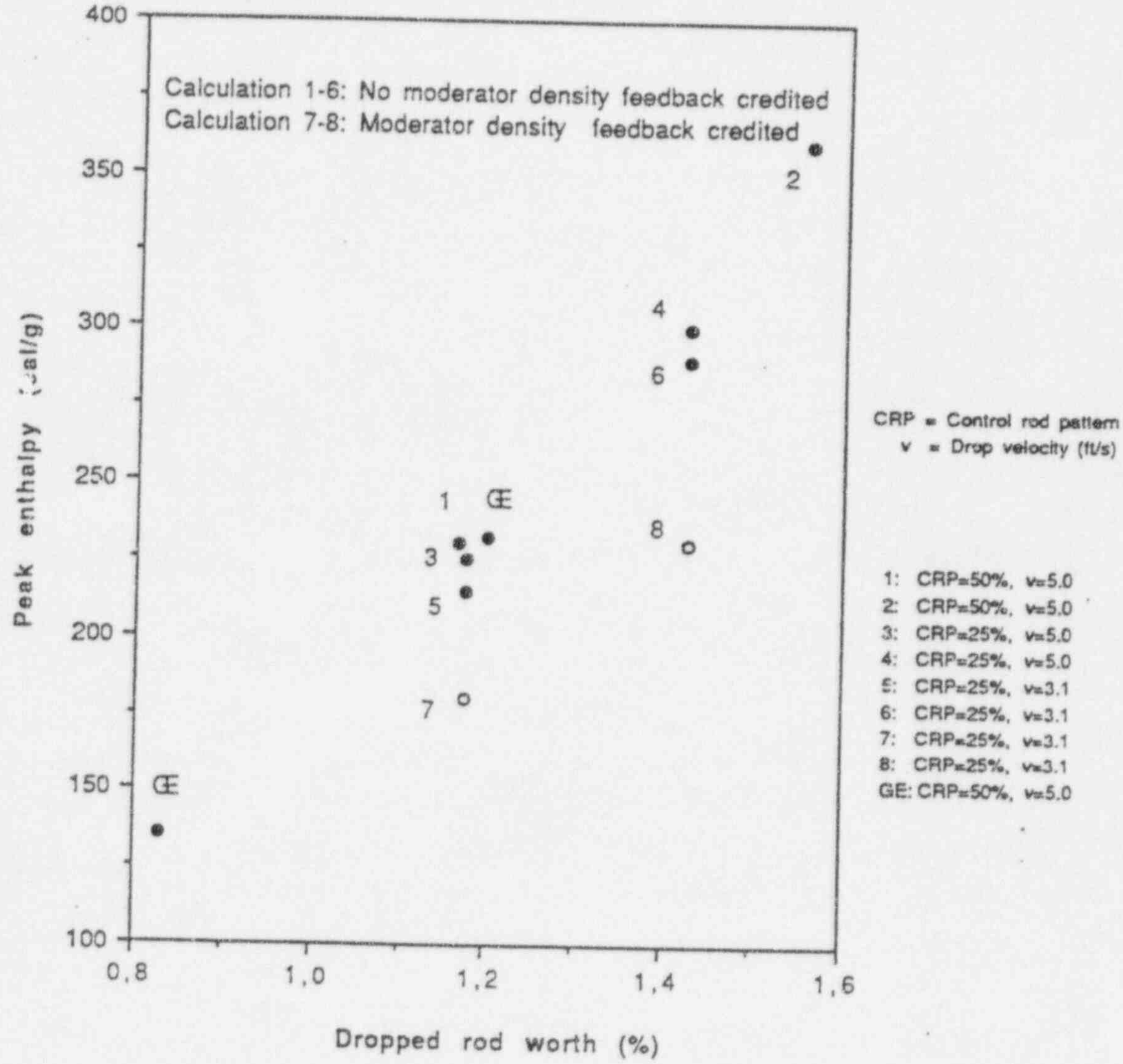


Figure 7.3.1 Peak adiabatic fuel enthalpy vs. dropped rod worth

GE BWR/4 CZP RDA APRM-SCRAM NO HYDR. STANDARD ROD
RAMONA-3B ANALYSIS
REACTIVITY COMPONENTS VS. TIME

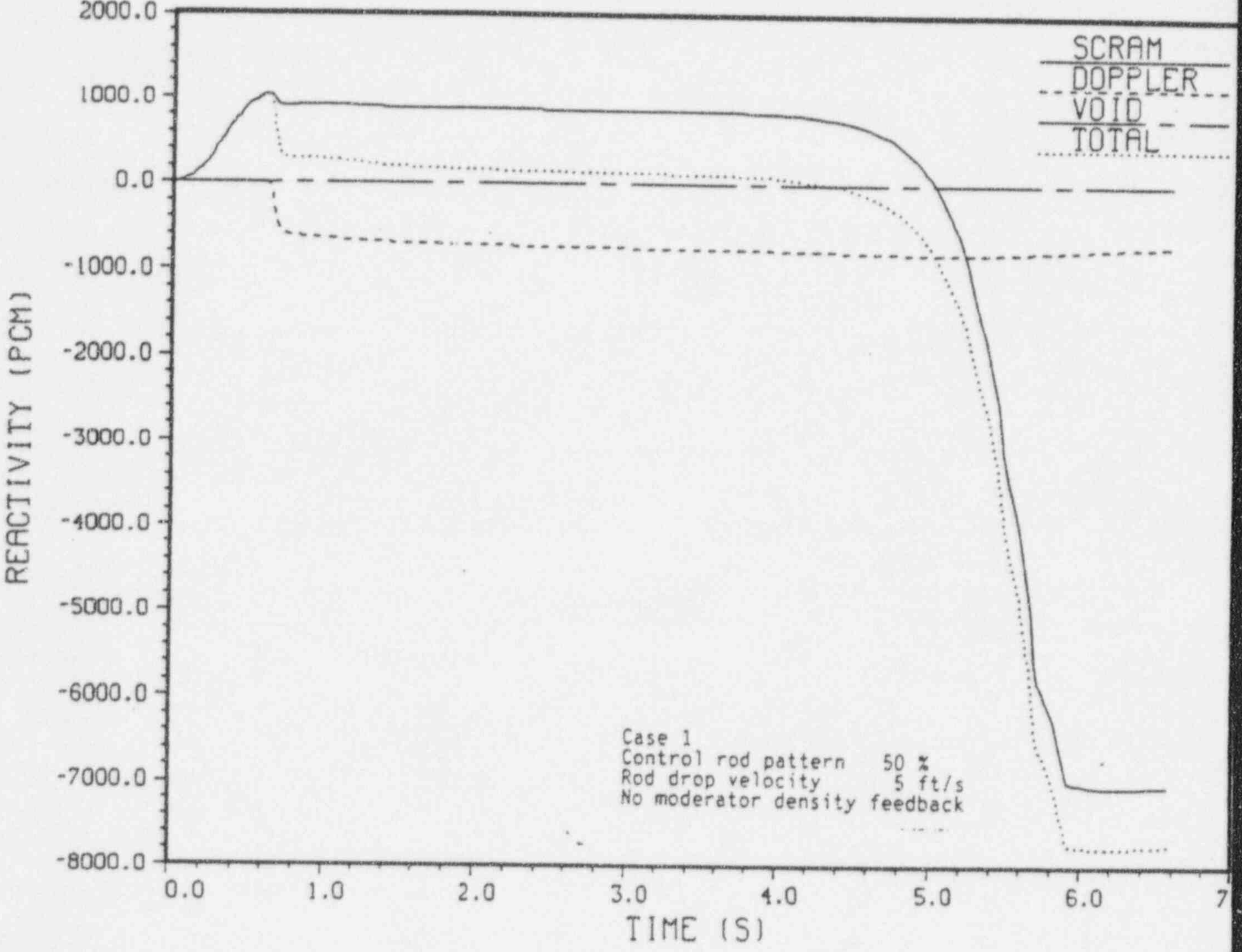


Figure 7.3.2 Standard control rods. CRDA. Rod worth 1.17%.
Sensitivity study. Sample case.
Reactivity components vs. time.

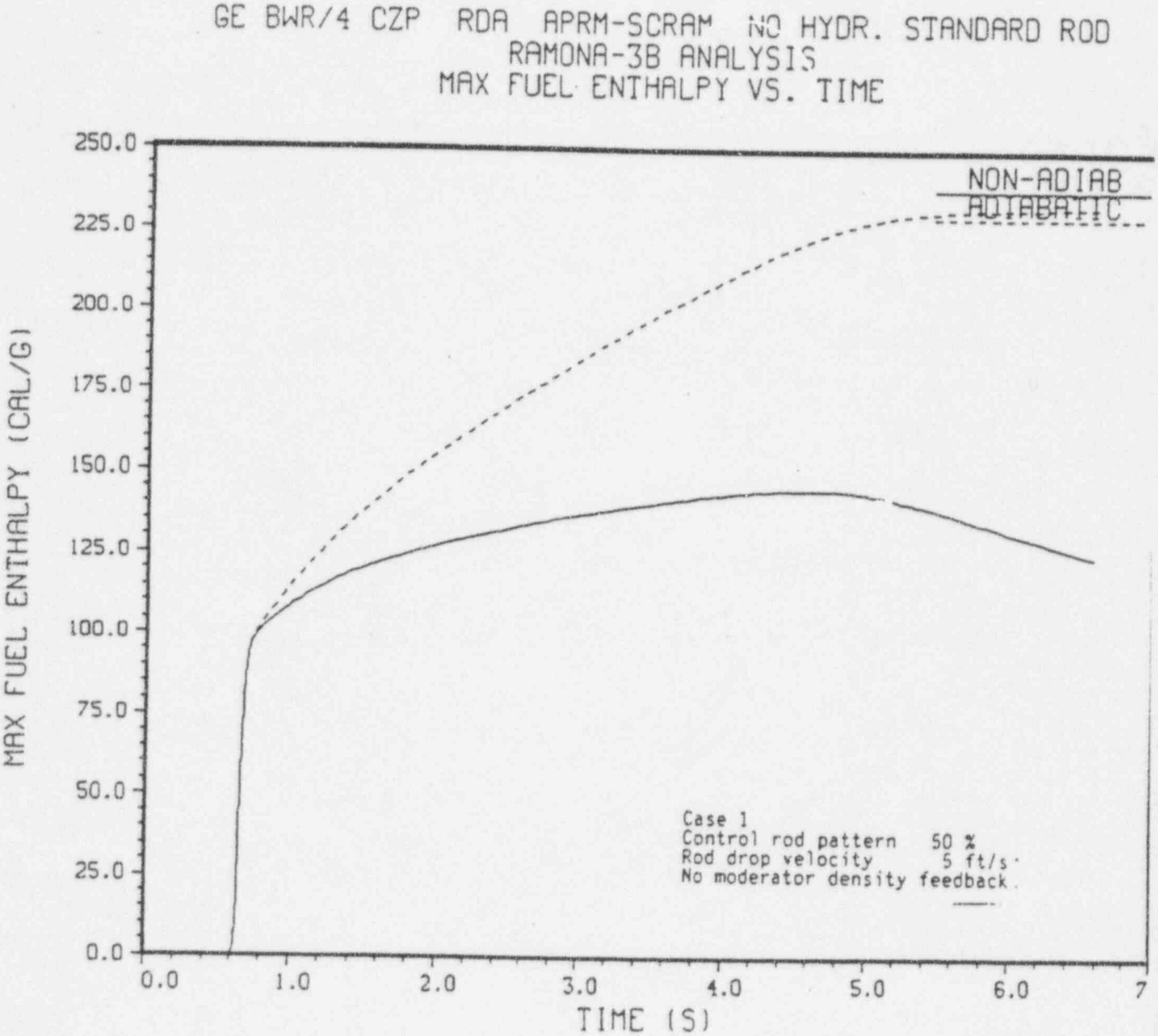


Figure 7.3.3 Standard control rods. CRDA. Rod worth 1.17%.
Sensitivity study. Sample case.
Peak fuel enthalpy vs. time.

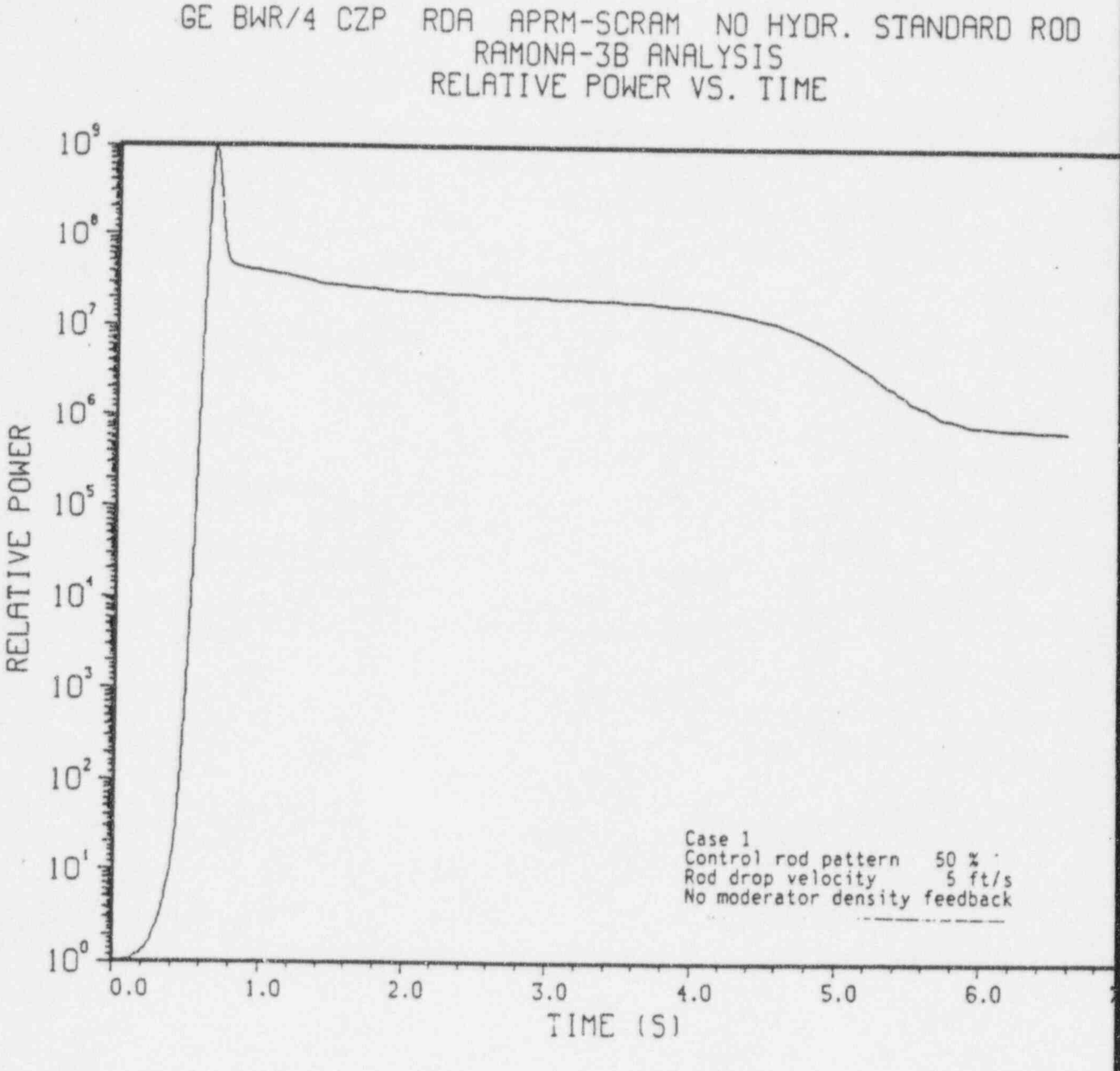


Figure 7.3.4 Standard control rods. CRDA. Rod worth 1.17%.
Sensitivity study. Sample case.
Relative power vs. time.

The influence of various analysis assumptions can be inferred from figure 7.3.1. For instance, the calculation marked 1 has a 50 % control rod density pattern whereas calculation 2 has 25 % .The difference in maximum adiabatic enthalpy is only 5 cal/g. (Also , there is a slight difference in the static rod worth, 0.01166 as opposed to 0.01174, which stems from the method of constructing a core with a central control rod with a rod worth of roughly 0.012.)

The control rod pattern used in calculation 1 is given below.

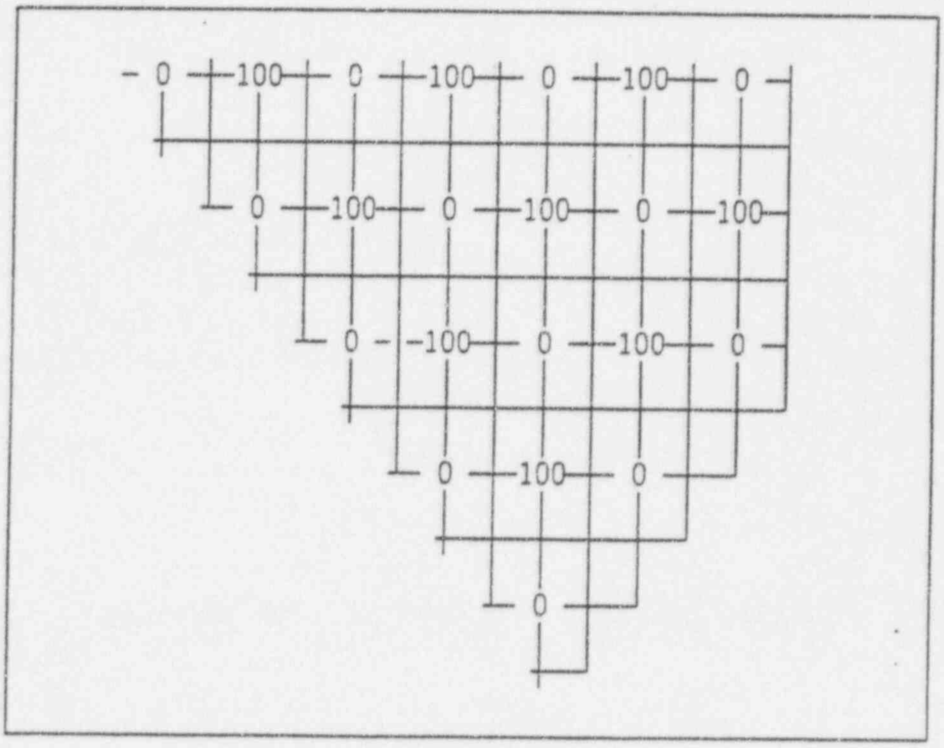


Figure 7.3.5 Control rod position (% withdrawal).
"50%" withdrawal pattern.

In calculation 3 the scram rod insertion velocity was assumed to be 5 ft/s, in calculation 5 it was 3.1 ft/s, all other parameters being unchanged. The difference in maximum enthalpy is 10 cal/g.

Table 7.3.1 summarizes the results for the CRDA analyses applying to the standard rod, and Table 7.3.2 gives the corresponding information regarding the High Worth Control Rod.

Calculation Number		Non-adiab. peak enth. (Heat transfer incl.)	Adiabatic peak enth. (No heat transfer)
1	CRP 50% 5ft/s Moderator feedb.excl	144	230
3	CRP 25% 5ft/s Moderator feedb.excl	136	225
5	CRP 25% 3.1ft/s Moderator feedb.excl	133	215
7	CRP 25% 3.1ft/s Moderator feedb.incl	112 *	180 *

* extrapolated value, see text, section 7.1.2
CRP = Control Rod Pattern

Table 7.3.1 Summary of calculated peak enthalpies (cal/g) in a postulated center rod drop accident.
Standard rod.
Static rod worth 1.174 % . (Case 1: 1.166 %)

Calculation Number		Non-adiab. peak enth. (Heat transfer incl.)	Adiabatic peak enth. (No heat transfer)
2	CRP 50% 5ft/s Moderator feedb.excl	232	360
4	CRP 25% 5ft/s Moderator feedb.excl	190	300
6	CRP 25% 3.1ft/s Moderator feedb.excl	182	290
8	CRP 25% 3.1ft/s Moderator feedb.incl	150 *	230 *

* extrapolated value, see text, section 7.2.2
CRP = Control Rod Pattern

Table 7.3.2 Summary of calculated peak enthalpies (cal/g) in a postulated center rod drop accident.
High Worth Control Rod.
Static rod worth 1.426 % . (Case 2: 1.559 %)

ABB

8. DISCUSSION

The modelled core is a representative reference core for the analysis of HWCR relative to standard GE control rods. The selected control rod density pattern (25% fully withdrawn) corresponds to a realistic configuration at cold critical conditions . At hot zero power conditions a 50% rod density would be more appropriate and this would allow a central control rod to be selected with higher rod worth. However, with a prescribed rod worth of about 1.2% for the standard rod, the postulated rod drop accident was analyzed at cold zero power initial conditions at which the favourable effect of moderator density feedback is much less pronounced. It is shown that, for a central control rod with a given rod worth, the maximum fuel enthalpy following a CRDA is not strongly dependent on global rod density.

It must be noted, also, that the selected rod patterns are conservative in the sense that all scram rods are either in the fully withdrawn or fully inserted position. In a more realistic rod pattern some rods would be banked in a partly inserted position. This would lead to the addition of more negative scram reactivity in the early part of the scram stroke, which would considerably improve the mitigation of the consequences of the CRDA.

The rod worth value selected for the CRDA analysis is close to the highest incremental rod worth within the constraints of the Banked Position Withdrawal Sequence in the generic analysis in Reference 2-5, which was achieved in a reactor core in which the maximum allowable number of control rods are bypassed and positioned in a "worst case" distribution.

It is known from other studies (see for instance Reference 6) that rod drop accident consequences are fairly insensitive to the delayed neutron fraction, fuel design and exposure. For a given (large) rod worth, a prompt critical excursion would occur somewhat earlier and be potentially more severe in exposed and reload cores than in low exposure cores, since the delayed neutron fraction is reduced (from approximately 0.00700 to approximately 0.00500) due to plutonium buildup in the fuel with exposure. However, Reference 5 indicates that the incremental reactivity worth associated with a given control rod tends to decrease with exposure. Therefore, a decrease in the delayed neutron fraction with exposure is generally accompanied by a decrease in the incremental reactivity worth associated with a given control rod. At any rate, the impact of the delayed neutron fraction is small.

The rod drop velocity also plays a minor role as can be seen from the study here. The parameter of most significance is rod worth. The results reported here fit well with those of Reference 5, where applicable, see figure 7.3.1, page 45. In this figure, the enthalpy values denoted "GE" should be increased by approximately 10 cal/g to account for a difference of roughly 0.00150 in delayed neutron fraction between these calculations and the rest in the diagram.

From the inspection of, for instance, figure 7.2.10, page 43, it is clear that there is an element of arbitrariness in the determination of the extrapolated peak enthalpies reported here. However, this is not a serious shortcoming since it is also clear that there is a substantial margin to the acceptance limit of 280 cal/g in these values. In view of this large margin the absolute values of the extrapolated peak enthalpies are not of major importance. This applies especially to the asymptotic "adiabatic" enthalpy for which the neglect of heat transfer to the coolant is an obvious conservatism. An alternative way to present results from the rod drop calculations, which has been practised in other studies, would be to report all peak enthalpy values at a specified point in time after the power peak, say t=2s. These values could be referred to as "power burst enthalpy". They would be lower than the values actually reported here.

The analysis reported here was specifically applied to a reactor core model of the GE BWR/4 design. The compatibility with the results of the generic study in Reference 2-5 (cf. figure 7.3.1), and with those of other studies using different methodologies (Reference 6), has been demonstrated. Because the BPWS maintains incremental rod worths to low values and because the analysis reported here applies to a rod worth (for the standard rod) corresponding to the highest rod worth attainable in BPWS cores (assuming a maximum allowable number of inoperable control rods) the results reported here can be considered to be generically applicable to all US reactors employing the BPWS, i.e. all BWR/4-6, and for all fuel types in these reactors.

The analyses presented in this report demonstrate that the stronger control rod (1.43 %) does not introduce more than a limited increase in maximum fuel enthalpy in a postulated rod drop accident as compared to the standard rod (1.17 %), and that in both cases there is an adequate margin to the acceptance limit of 280 cal/g.

9. SUMMARY

The control rod drop accident has been analysed using the ABB Atom CRDA analysis methods including the RAMONA-3B computer code. The influence on the outcome of the accident of replacing a standard control rod with a HWCR having 15 % higher rod worth has been determined by performing calculations for both types of control rods and comparing the results. Heat transfer to the coolant as well as reactivity feedback due to density changes in the coolant have been accounted for. Calculations have also been performed where these effects have been neglected. Sensitivity studies have been performed to show the influence of the control rod worth, rod drop velocity, and initial control rod density pattern.

The results show that the replacement of the standard control rod by a HWCR will increase the maximum fuel enthalpy following the CRDA by approximately 40 cal/g when accounting for moderator density feedback and heat transfer from the fuel rods during the accident. At the high control rod worths assumed in this work, the peak fuel enthalpies are 112 cal/g and 150 cal/g for the standard control rod and the HWCR respectively.

The sensitivity studies that have been performed show that the effect of neglecting the moderator density feedback compared to including it is to increase the peak fuel enthalpy with 20 to 25 cal/g . Neglecting the heat transfer from the fuel rods leads to an additional conservatism of 70 to 80 cal/g.

The conclusions from the analyses are that even though the peak fuel enthalpy is increased by replacing the standard control rod by a HWCR, there is still ample margin to the limitations set forth in the Standard Review Plan, in particular to the peak fuel enthalpy limit of 280 cal/g.

10. REFERENCES

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