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

Our ref: LTR-NRC-04-25

April 29, 2004

Subject: Submittal of WCAP-16259-P/WCAP-16259-NP, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," for NRC Review and Approval (Proprietary/Non-proprietary)

Dear Mr. Wermiel:

Enclosed is a copy of WCAP-16259-P/WCAP-16259-NP, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," submitted to the NRC for Review and Approval (Proprietary/Non-proprietary). It is requested that the above topical be approved by June 2005, in support of the Next Generation Fuel (NGF) implementation and extended power uprate submittals planned by several licensees in 2006. WCAP-16259-P describes the Westinghouse developed methodology for three-dimensional core kinetics analysis of non-LOCA transient analysis of pressurized water reactors using NRC-licensed computer codes.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-04-1829 with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit, AW-04-1829.

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

T007

Correspondence with respect to any Application for Withholding should reference AW-04-1829 and should be addressed to James A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: F. M. Akstulewicz, NRR
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Attention: J. S. Wermiel, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Our ref: AW-04-1829

April 29, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Submittal of WCAP-16259-P/WCAP-16259-NP, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," for NRC Review and Approval (Proprietary)

Reference: Letter from James A. Gresham to J. S. Wermiel, LTR-NRC-04-25, dated April 29, 2004

Dear Mr. Wermiel:

The Application for Withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-04-1829 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-04-1829 and should be addressed to James A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance and Plant Licensing

AW-04-1829

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

James A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 29th day
of April, 2004

Sharon L. Fiori
Notary Public

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

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

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

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

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked "Submittal of WCAP-16259-P/WCAP-16259-NP, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," for NRC Review and Approval (Proprietary/ Non-proprietary)," April 29, 2004, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-04-25) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is that associated with a request for NRC review and approval.

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

(a) Obtain generic NRC licensed approval for the Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis.

(b) This methodology will promote convergence between Westinghouse business units.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use its methodology capability to further enhance their licensing position over their competitors.

(b) Assist customers to obtain license changes.

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

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

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

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC. In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Copyright Notice

Westinghouse Non-Proprietary Class 3

WCAP-16259-NP
Revision 0

April 2004

Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis

Westinghouse

WCAP-16259-NP

WESTINGHOUSE METHODOLOGY FOR APPLICATION OF 3-D TRANSIENT NEUTRONICS TO NON-LOCA ACCIDENT ANALYSIS

April 2004

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The authors wish to express their sincere appreciation for the support offered by David S. Huegel, who reviewed the accident analyses presented in this document and provided invaluable comments.

Particularly noteworthy were the efforts of our project manager, Jennifer Moon, who provided efficient administration, organization and many key decisions to keep this program moving forward.

We also thank the management of both Nuclear Safety and PWR Core Technologies for their support and encouragement during the entire duration of this program. Lastly, the authors wish to thank Mr. William Slagle for his licensing support on this program.

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TABLE OF ACRONYMS

Acronym	Definition
1-D	One-Dimensional
3-D	Three-Dimensional
AC	Alternating Current
AFD	Axial Flux Difference
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	Axial Offset
ARO	All Rods Out
ASI	Axial Shape Index
ASME	American Society of Mechanical Engineers
BOC	Beginning-of-cycle Life
CE	Combustion Engineering
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CLOF	Complete Loss of Flow
COLR	Core Operating Limits Report
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EOC	End-of-cycle Life
EPRI	Electric Power Research Institute
ESF/ESFAS	Engineered Safety Features/Engineered Safety Features Actuation System
$F_{\Delta H}$	Radial Power Peaking Factor
F_Q	Total Hot Spot Peaking Factor
FSAR	Final Safety Analysis Report
HEM	Homogeneous Equilibrium Model
HFP	Hot Full Power
HZP	Hot Zero Power
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MMF	Minimum Measured Flow
MSLB	Main Steamline Break
MSSS	Main Steam Supply System
MTC	Moderator Temperature Coefficient
MWD/MTU	Megawatt Days per Metric Ton of Uranium
N-1	All Rods Inserted, Less the Worst Stuck Rod
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission

Acronym	Definition
NSC	Nuclear Science Committee
OECD	Organization for Economic Co-Operation and Development
OPΔT	Overpower Delta-T
OTΔT	Overtemperature Delta-T
PSU	Pennsylvania State University
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCCA	Reactor Control Cluster Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RSE	Reload Safety Evaluation
RTDP	Revised Thermal Design Procedure
RTP	Rated Thermal Power
SDM	Shutdown Margin
SER	Safety Evaluation Report
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SI	Safety Injection
SLB	Steamline Break
SLI	Steamline Isolation
STDP	Standard Thermal Design Procedure
TDF	Thermal Design Flow
T&H	Thermal-Hydraulic

PREFACE

This report presents the Westinghouse Electric Company developed methodology for the analysis of non-LOCA transients for pressurized water reactors using a three-dimensional core kinetics model. The report is structured into four major chapters, a list of references and three appendices. A brief overview of the content of each of these chapters follows:

1.0	Introduction	This chapter provides a brief discussion of the 3-D methodology and the current methodology used by Westinghouse.
2.0	Generic Models	This chapter describes generically the basic proposed Westinghouse methodology using 3-D kinetics, including discussion of the codes and models utilized. It also addresses the applicability to other reactor types and the safety analysis method to be used for reload cores.
3.0	Sample Applications of 3-D Methodology	This chapter presents the sample calculations performed to demonstrate the application of 3-D methods, in comparison to current methods. The calculational results are representative and are not intended for the licensing of any specific reactor unit. A concise overview of the applicability of the methodology to events not specifically analyzed is also presented in this chapter.
4.0	Summary and Conclusions	A concise overview of the methodology and continued code functionality is presented in this chapter.
5.0	References	A list of references is provided in this chapter which documents the pertinent reports and papers which are referenced throughout this report.
Appendix A	Overview of Computer Codes	Although the computer codes being used in this methodology are currently approved by the NRC, this appendix provides some background on the codes and the data interchange between the codes.
Appendix B	OECD Main Steamline Break (MSLB) Benchmark	The OECD PWR MSLB benchmark problem was analyzed using the computer codes described in Appendix A. The Westinghouse results are compared to the reference results in this appendix.
Appendix C	Sensitivity Studies	This appendix provides the results of a sensitivity study of the key parameters which impact each of the analyzed events, and defines a reference bounding analysis case for each event.

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1.0 INTRODUCTION

In order to determine the safety of a reactor with respect to reactor systems failures, a set of postulated accident events is analyzed, and the results are presented in Chapter 14 or 15 of the plant Final Safety Analysis Report (FSAR). The accidents to be addressed are specified in the Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70 (Reference 1), and are listed in Table 1.0-1. As shown in the table, the accidents are classified into generalized categories involving an Increase in Heat Removal (reactor coolant system (RCS) cooldown events), Decrease in Heat Removal (RCS heatup events), Decrease in RCS Flow, Reactivity and Power Distribution Anomalies (core-related events), and events involving an Increase or Decrease in Reactor Coolant Inventory. Within these event categories, an accident may also be classified according to its frequency of occurrence and potential consequences. In general, the more frequent occurrences must meet more limiting criteria with respect to fuel damage and radiological releases. This method of classification of events is shown in ANSI N18.2 (Reference 8), and is also listed in Table 1.0-1. The NRC review process for each of the events is presented in the NRC Standard Review Plans, NUREG-0800 Rev. 1 (Reference 2). (The NRC has issued more detailed regulatory guides for some specific events; for example, RG 1.77 for the RCCA Ejection accident.) Table 1.0-1 also lists some events which involve a significant loss of reactor coolant, i.e. the Steam Generator Tube Rupture (SGTR) and Loss of Coolant Accident (LOCA). The purpose of this report is to address the use of an updated 3-dimensional core transient analysis methodology for non-LOCA events. Loss of coolant accident events are not addressed here.

In the analysis of the non-LOCA accidents, the typical approach has been to make conservative and bounding analysis assumptions, either because of analysis expediency, or because of the simplified modeling assumptions. In some cases, this has resulted in combinations of assumptions that cannot occur in reality. For example, an accident event may have been analyzed with a beginning-of-cycle moderator temperature coefficient (MTC), an end-of-cycle Doppler feedback coefficient, excessive control rod reactivity worths, and an end-of-cycle axial power distribution. Other analysis assumptions have included overly conservative constant moderator temperature coefficients, the inconsistent use of [ ] [a, c] for calculating trip reactivity along with a top–peaked shape for Departure from Nucleate Boiling (DNB) analysis, and the use of conservative, constant (design value), core peaking factors. The consistency of the analysis assumptions can be improved by externally linking the RCS loop thermal-hydraulics calculational model to a more realistic 3-dimensional core neutronics and heat transfer model, as described in this report.

The objective of this report is to present the Westinghouse method for the application of three-dimensional core neutron kinetics to the analysis of non-LOCA FSAR accident events. This method uses the NRC-approved core neutron kinetics code SPNOVA (References 3 & 4) and the NRC-approved core thermal-hydraulics code VIPRE-01 (VIPRE) (References 5 & 6), in conjunction with the NRC-approved RCS loop thermal-hydraulics code RETRAN-02 (RETRAN) (Reference 26). See Appendix A for additional information on the computer codes and data interchange. The codes are linked using an external communication interface. No changes were made to the codes other than changes necessary to facilitate the data transfer between the codes. The linkage of the codes documented herein is based on the NRC-approved linkage of the SPNOVA and VIPRE codes for the analysis of the Control

Rod Ejection transient (Reference 7). This report demonstrates that with the additional linkage to the RETRAN computer code, the updated methodology allows a more realistic yet conservative non-LOCA analysis with respect to the current licensing acceptance criteria. The independent code limitations and uncertainties continue to be applicable when the codes are linked using an external communication interface. Although the accidents chosen for the sample applications shown in Chapter 3 were performed for a 3-loop Westinghouse plant, the methodology is not limited to this plant type. The same computer codes employed herein have been used in licensing applications for many Westinghouse-designed 2-, 3- and 4-loop plants with various fuel designs, and by Westinghouse for a CE-designed analog protection system plant. The computer codes and method of data transfer between the codes (the external communication interface) are applicable to any PWR for which a licensed model is available for the base codes (i.e., SPNOVA, VIPRE and RETRAN).

A licensed model includes plant-specific variations in the reactor core, RCS primary/secondary system design, reactor control and protection system design, accident limits and specific uncertainty allowances. These models are unaffected by the linking of the codes using the external communication interface. Therefore, the methodology demonstrated in Chapter 3 can be applied to any PWR for which licensed models exist, taking into account the plant-specific variations and uncertainty allowances. Thus, although there will be differences in the models used for different PWR configurations, these changes are clearly identified in the current licensed models and methodology for that plant. The 3-D application methodology described in this report is therefore independent of the PWR plant type.

This topical report shows sample calculations for a representative subset of the non-LOCA events. The use of an external communication interface to link the 3-D core calculations with the RCS loop model was mainly implemented to recover existing margin in the DNB limiting events. Therefore, the representative events presented in this topical report were selected based on their severity with respect to the DNB or overpressure licensing basis. However, the methodology presented in this topical report would be applicable to all of the events currently analyzed with RETRAN as listed in Table 3.6-1. The demonstration transients presented herein utilize all of the functionalities required for the remainder of the non-LOCA events.

The Nuclear Science Committee (NSC) of the Nuclear Energy Agency (NEA)/Organization for Economic Co-Operation and Development (OECD) has released a set of computational benchmark problems for a study of the accuracy of computer codes used in nuclear plants safety analysis. Recently, in a cooperative program sponsored by the OECD, the United States Nuclear Regulatory Commission (US NRC), and the Pennsylvania State University (PSU), a PWR Main Steamline Break (MSLB) benchmark problem has been defined in order to simulate the core response and the reactor coolant system response to a relatively severe steamline break accident condition. This problem was considered appropriate to test the incorporation of a full three-dimensional (3-D) modeling of the reactor core into a system transient code to allow simulations of interactions between reactor core behavior and plant dynamics. Appendix B presents the OECD PWR main steamline break benchmark problem utilizing the computer codes described in Appendix A. The benchmark was structured into three separate phases: 1) plant transient simulation with point kinetics, 2) transient simulation with 3-D neutronics/core thermal-hydraulics, and

3) plant transient simulation with 3-D core neutronics. The benchmark exercises were performed to provide additional validation of the external communication interface.

Appendix C contains the background information on the sensitivity of the key factors which impact the non-LOCA transients. Sensitivity studies were performed for each event presented to determine if the parameters selected for the base case for each event yielded the most limiting results, and to document the sensitivity of the results to variations in the parameters. The results were used to define a reference bounding analysis case for each event.

The Westinghouse methodology for application of 3-D transient neutronics to the non-LOCA analyses continues to follow the bounding analysis concept, as described in WCAP-9272-P-A (Reference 15). This concept assumes that the validity of the reference analysis is established for the reload core in question on the basis that the key safety parameters for the reload core assume values that are conservatively bounded by those used in the reference analysis. If all key safety parameters remain conservatively bounded, the reference safety analysis is assumed to apply, and no further analysis is necessary. When a reload parameter is not bounded, further analysis or evaluation is considered necessary. This may be a complete reanalysis of the accident, or a simple quantitative evaluation. Calculational uncertainties and biases in the key safety parameters continue to be accounted for both in the first time analysis and the reload analyses.

Table 1.0-1
US NRC Reg. Guide-1.70 Classification of Events
(and ANSI N18.2 Condition II, III, IV Event Classification)

1.	**Increase in Heat Removal by Secondary System**
	a. Feedwater Malfunctions Causing a Decrease in Feedwater Temperature **(II)**
	b. Feedwater Malfunction Causing an Increase in Feedwater Flow **(II)**
	c. Excessive Increase in Secondary Steam Flow **(II)**
	d. Inadvertent Opening of a SG Safety or Relief Valve **(II)**
	e. Steam System Piping Failure **(III & IV)**
2.	**Decrease in Heat Removal by Secondary System**
	a. Loss of Electrical Load and/or Turbine Trip **(II)**
	b. Loss of Non-Emergency AC Power **(II)**
	c. Loss of Normal Feedwater **(II)**
	d. Feedwater System Pipe Break **(IV)**
3.	**Decrease in Reactor Coolant Flow Rate**
	a. Partial Loss of Forced Reactor Coolant Flow **(II)**
	b. Complete Loss of Forced Reactor Coolant Flow **(III)**
	c. RCP Shaft Seizure (with & w/o Loss of AC Power) **(IV)**
	d. RCP Shaft Break **(IV)**
4.	**Reactivity and Power Distribution Anomalies**
	a. Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition **(II)**
	b. Uncontrolled RCCA Bank Withdrawal at Power **(II)**
	c. RCCA Misoperation (RCCA Misalignment, Rod Drop **(II)**, Single Rod With. **(III)**)
	d. Startup of an Inactive Reactor Coolant Loop **(II)**
	e. Uncontrolled Boron Dilution **(II)**
	f. Inadvertent Loading of a Fuel Assembly in an Improper Location **(III)**
	g. Spectrum of RCCA Ejection Accidents **(IV)**
5.	**Increase in Reactor Coolant Inventory**
	a. Inadvertent ECCS Actuation at Power **(II)**
	b. CVCS Malfunction Causing an Increase in Reactor Coolant Inventory **(II)**
6.	**Decrease in Reactor Coolant Inventory**
	a. Inadvertent Opening of a Pressurizer Safety or Relief Valve **(II)**
	b. Steam Generator Tube Failure **(IV)**
	c. Loss of Coolant Accident **(IV)**

2.0 GENERIC MODELS

The basic reactor core, reactor vessel, RCS loops, pressurizer, steam generator, reactor control and protection, and safeguards models used in the Westinghouse updated 3-dimensional core neutronics transient analysis methodology are described in this chapter. The calculational models are unchanged from the models presented in the NRC-approved computer code application reports. Only the input to the models is changed to ensure a conservative calculation for the individual transient. This is accomplished by assuming initial core conditions (e.g., time in cycle, xenon distribution, power shapes) which are conservative for the accident consequences. In addition, conservative uncertainty allowances are applied to the key parameters that affect the course of the event. For this 3-D application, the method used was to apply the uncertainties in a deterministic manner, i.e. simultaneously in the worst (i.e., most limiting) direction in the same calculation. For accidents analyzed using the NRC-approved Westinghouse revised thermal design procedure (RTDP), a statistical approach is used to take into account uncertainties in the initial thermal-hydraulic conditions (i.e., reactor power, inlet temperature, pressure, and flow rate) in determining the Departure from Nucleate Boiling Ratio (DNBR) limit. The treatment of uncertainty allowances is the same as the current FSAR non-LOCA accident analysis methods.

In this report, sample calculations were performed and presented for a 3-loop Westinghouse plant. The same analysis methodology applies to 2- and 4-loop Westinghouse plants. In addition, since the updated methodology does not result in modifications to the computer codes' calculational models, the same philosophy can be applied to any plant, Westinghouse or non-Westinghouse, for which the SPNOVA, VIPRE and RETRAN codes have been used in the conventional (non-linked) method to perform the cycle design and safety analysis.

2.1 Computer Codes

Although the updated non-LOCA 3-D core neutronics and RCS loop analysis methods described in this report are code-independent, the methods described herein use the NRC-approved SPNOVA, VIPRE and RETRAN computer codes. No changes were made to the fundamental code algorithms; the only changes were those necessary to automate the data transfer between the codes. The SPNOVA code is used to perform steady-state and transient 3-D core neutronics calculations, using the VIPRE code to calculate the transient local coolant density and fuel effective temperature (T_{eff}) for the feedback calculations. The SPNOVA code also includes static thermal-hydraulics models for steady-state design calculations. The use of the SPNOVA/VIPRE codes, and the automated data transfer method, was approved by the NRC for the 3-D transient analysis of the RCCA Ejection event in Reference 7. The VIPRE code is used to calculate the local heat flux to the coolant in the RETRAN core model described below. In the sample calculations presented in this report, the SPNOVA/VIPRE calculations were performed using a full-core 3-D model as described in Section 2.2.1. The SPNOVA and VIPRE codes are described in more detail in References 3 and 6.

The RETRAN code is used to calculate the RCS conditions versus time, including the reactor vessel, RCS loops, pressurizer and steam generators. The RETRAN code also models the reactor trips, engineered safety feature (ESF) functions, and the RCS control functions. The RCS nodal description, including the RCS loops, steam generator, reactor vessel and pressurizer models, is identical to that used in the current NRC-approved analysis method. In the core region, the number of axial nodes is increased to facilitate the data transfer from VIPRE. The core point neutron kinetics and fuel rod heat transfer models in RETRAN are not used. Instead, the pointwise local heat flux vs. time calculated by VIPRE is input to the RETRAN core nodes using the standard RETRAN non-conducting heat exchanger model. These changes do not result in any modifications to the RETRAN calculational models or numerics. The VIPRE code obtains its core inlet conditions (core inlet flow and temperature) and core exit pressure from the RETRAN calculation. The RETRAN model is described in more detail in Reference 26.

The VIPRE code is also used in a separate calculation to determine the hot rod minimum DNBR versus time and the fuel and clad temperatures versus time. The minimum DNBR vs. time is calculated using the subchannel model described in Section 2.4.1. The hot rod fuel rod and clad temperatures versus time are calculated using the model described in Section 2.4.2.

The application of the computer codes discussed above is addressed in more detail in Appendix A of this report. Because the methodology defined here is independent of the specific codes, other approved codes may be utilized in the future using the same methods and using the code-specific models that have been previously approved.

2.2 Reactor Core Model (SPNOVA/VIPRE)

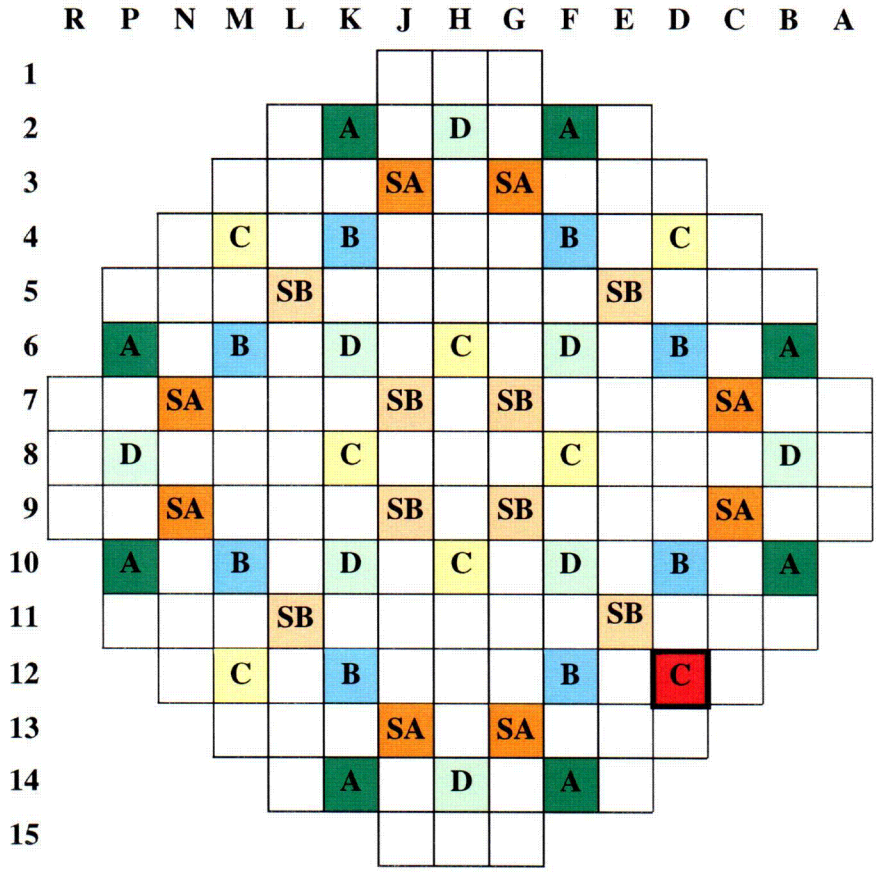
The reactor core model used in the updated 3-D core transient analysis methodology is identical to the core design model approved for use in WCAP-15806-P-A. No new models were developed for this analysis. The models used are described below.

2.2.1 Nuclear Model

The core selected for the sample application to demonstrate the methodology is a typical Westinghouse 3-loop core with 157, 17x17 fuel assemblies and an 8-cluster lead control bank (Bank D). The core geometry and control cluster locations (i.e., control banks A, B, C, D and shutdown banks SA, SB) are shown in Figure 2.2-1.

The shaded fuel assembly cluster at D-12 (or one of its symmetric counterparts) indicates the typical position of the worst stuck (non-trippable) rod at the beginning or end of the cycle.

Figure 2.2-1
Illustration of 3-Loop Control and Shutdown Rod Locations



Bank	No. of RCCAs
A	8
B	4
C	8
D	8
SDA	8
SDB	8

C01

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2.2.2 Thermal-Hydraulic Model for Feedback Calculations

The moderator densities and fuel temperatures for the neutronics feedback calculation are calculated using the VIPRE code. The VIPRE calculation uses a multi-zone fuel pellet representation for the fuel rod in each neutronics/thermal-hydraulic core node. The number of radial and axial nodes is typically mapped one-to-one between SPNOVA and VIPRE, although a more detailed axial nodalization can be used in VIPRE. The fuel rod model uses [ ]^{a,c} radial mesh points in the fuel pellet and two mesh points in the clad. The fuel pellet-to-clad gap heat transfer is calculated using the dynamic gap conductance model in VIPRE, which accounts for changes in the fuel dimensions and fill gas pressure with temperature. The resonance effective fuel temperature is generated in each SPNOVA node from the VIPRE radially-varying fuel pellet temperatures using design values of the T_{eff} weighting function. For consistency with the static nuclear design model, the VIPRE average fuel rod model is calibrated against the nominal design static fuel rod model temperatures over the power range of interest [ ]^{a,c}. This calibration is performed for the typical fuel compositions in the core, and as a function of fuel depletion.

An input multiplier on the Doppler feedback cross-section adjustment can be applied in SPNOVA to cover the uncertainties in the actual T_{eff} calculation. This results in a uniform uncertainty allowance applied to the Doppler feedback adjustments. The core parameters related to moderator feedback can be adjusted to conservatively pessimize the moderator density feedback effect. This is discussed in more detail below.

2.2.3 Static Nuclear Design Methods

The basic inputs used in the SPNOVA static nuclear model are the same cross-section sets, burnup distributions, fuel rod, fuel assembly, control rod geometry and other models used in the nuclear design model for the specific plant reload cycle design.

A potential cycle history factor is the impact at beginning-of-cycle (BOC) due to the previous cycle length. Since the safety analysis calculations may be performed prior to the shutdown of the previous cycle, the BOC evaluations need to encompass the impact of the potential variability of the previous cycle length. [ ]^{a,c}.

Fundamental in the Westinghouse methodology is the continued use of the reload safety evaluation process. Through this process, the impact of the reload cycle can be determined from static nuclear design calculations, and the transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. Key parameters for each accident are defined in Chapter 3. These were found to be consistent with the key parameters identified in the current Westinghouse reload cycle methodology presented in Reference 15.

Shown below are the typical current static calculational methods used to calculate the values of the kinetics parameters that may affect the transient accident analysis:

a. Doppler Feedback

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

b. Moderator Feedback

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

c. Delayed Neutron Fraction

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

d. Trip Reactivity Worth

The trip rod worth is dependent on the arrangement of fuel assemblies within the core, the control rod pattern, the axial and radial power distribution due to burnup and xenon effects, and the allowed insertion limits. There are two different aspects of the trip reactivity worth that are important: the total reactivity worth, which is important for shutdown margin, and the initial trip reactivity worth versus rod position, which is important to turn around the transient. If the control rods are partially inserted, the total trip rod worth decreases by the amount of the inserted rod worth, but the initial trip worth may be greater. The initial trip rod worth is a maximum for power distributions which skew the power to the top of the core.

The core power distribution is typically skewed slightly to the bottom of the core at full power due to the feedback, but could become skewed to the top of the core due to a xenon transient.

2.2.4 Reactor Core Initial Conditions

There are two key core operation parameters aside from the time of cycle and depletion model that can have a significant effect on the inserted control rod bank worths and core radial and axial power peaking factors, and can be adjusted as part of the initial conditions for the analysis. These are the axial xenon distribution and the control rod bank positions.

a. Axial Xenon Distribution

The axial xenon distribution can have a significant impact on the axial power distribution used in the DNB evaluation, and in the initial effectiveness of the reactivity insertion following a reactor trip. Xenon distributions that force the power distribution to the top of the core are more limiting for DNBR since they increase the axial power peaking factor in the top of the core where the local fluid conditions are the most limiting. However, they result in a more effective reactor trip since the trip rod reactivity is inserted into the core more quickly compared to that of a power shape in the bottom of the core.

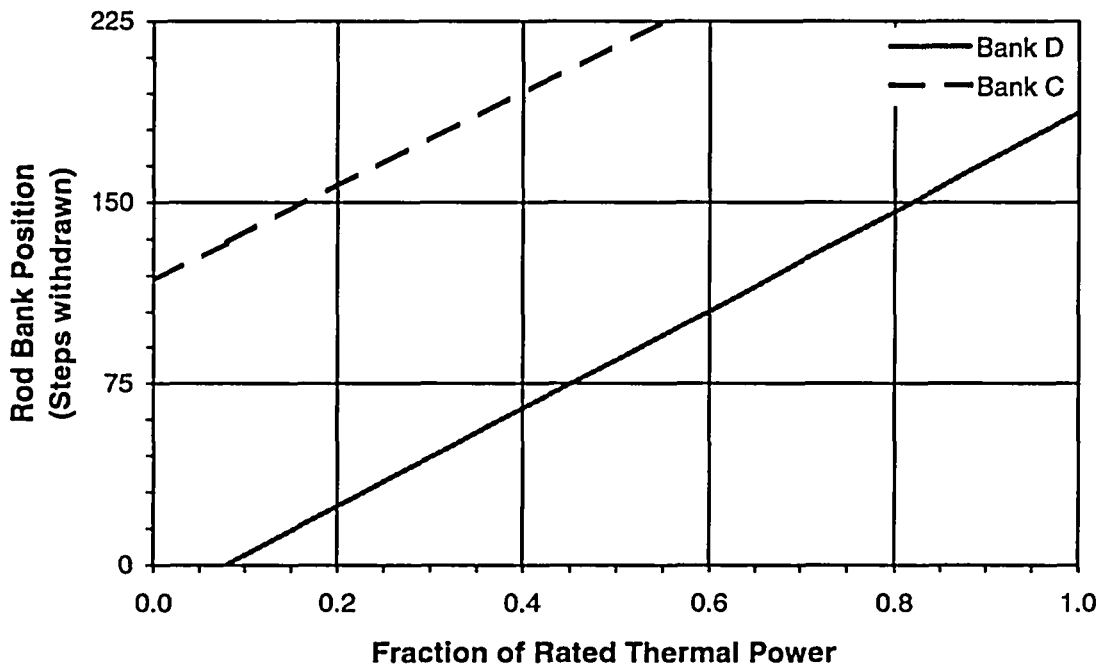
In the power operating range, there is a nominal operating range in which the reactor is allowed to operate. This band of operation is typically defined by axial flux difference (AFD) limits as a function of power level (Reference 16). Note that AFD is identical to axial offset (AO) at hot full power. The axial shape index (ASI) used for CE-designed plants is just the negative of axial offset. The AFD limits can be a band around the equilibrium value, or absolute limits. Most of the FSAR accident events are limiting at the hot full power condition. For the analysis of these events, a limiting axial xenon distribution is used in the precondition for the event. This precondition is a xenon distribution that gives an axial offset at the most positive or most negative allowed value (or any value in-between) at this power level, and may have secondary characteristics which generate significant local peaking. As shown in the sample applications presented in Chapter 3, initial axial power distributions representing several initial axial offsets/axial shape indices may have to be evaluated to find the limiting case.

b. Control Bank Positions

The allowed control bank insertion as a function of power level is confirmed during the reload cycle design process, and the control rod insertion limits are specified in the plant Technical Specifications or the Core Operating Limit Report. These limitations on the bank insertion are important to ensure sufficient shutdown margin as a function of power, and to limit the potential increase in radial and axial power peaking factors that can occur due to rod insertion. The control bank insertion limits for the core design used in the sample calculations presented in Chapter 3 of this report are presented in Figure 2.2-2. Technical Specification limits on control rod insertion, and the control rod insertion limit alarms, ensure that it is highly unlikely that the control rods will be inserted to or beyond the specified limits. For events that are caused by a malfunction of the rod control system, and are sensitive to the rate or total amount of reactivity insertion, the control rods are typically assumed to be initially inserted to the insertion limit.

For other events, where control rod movement would mitigate the event, the rod control system is assumed to not operate and the control rods are initially assumed to be fully withdrawn, since this maximizes the time to insert significant reactivity worth after a trip. In either case, the axial power distribution can be adjusted to yield a conservative power shape using the axial xenon adjustment method described above.

Figure 2.2-2
Illustration of Control Rod Insertion Limits as a Function of Power



2.3 Reactor Plant Model (RETRAN)

The reactor plant, including the RCS primary loop model and secondary steam system model, reactor control and protection system, and engineered safety features system, is modeled using the RETRAN code. RETRAN is a very flexible one-dimensional, best-estimate, thermal-hydraulic transient analysis computer code. It uses a variable nodalization with a user-selected control volumes and flow paths, and heat conductors to account for heat transfer in the primary and secondary system. The code includes various component models, including a two-region non-equilibrium pressurizer, centrifugal pumps, valves, and non-conducting heat exchangers. A flexible control system model allows the user to input a wide range of auxiliary calculations or systems. The core model allows either point-neutron kinetics or one-dimensional space-time kinetics to be used for the neutronics.

The application of this code to Westinghouse reactors, including the nodalization for the various system models, was presented to the NRC in WCAP-14882-P-A (Reference 26). This report was reviewed and approved by the NRC for application to all Westinghouse 2-, 3- and 4-loop plants. The Westinghouse model includes the use of a point neutron kinetics model for the core neutronics. The updated

3-dimensional core transient analysis methodology addressed in this report uses the same models as approved in Reference 26, except that the point-kinetics and fuel-rod heat transfer models are not used. Instead, the core kinetic behavior is calculated externally using the SPNOVA and VIPRE codes (see Section 2.2), and the calculated heat flux is automatically transferred to the RETRAN core model using the [ ] [a, c]. No new models were developed for the RETRAN calculation. The RCS primary and secondary nodalization is unchanged, except for the addition of more axial nodes in the core to facilitate the transfer of the external heat flux. Since the models are unchanged from those presented in WCAP-14882-P-A, they will be discussed only briefly below.

2.3.1 RCS Loop Model

The reactor coolant loop model of a Westinghouse reactor consists of a vertical U-tube steam generator and a vertical, single-stage, shaft-sealed reactor coolant pump in each loop, and the interconnecting piping between the steam generator, reactor coolant pump and the reactor vessel. An electrically-heated pressurizer is connected to the hot leg of one of the primary loops in order to maintain the primary side pressure above saturation, and to provide for the coolant displacement that occurs during a plant heatup or cooldown. Pressurizer relief and safety valves are modeled. Both pre-heat and feedring steam generators can be modeled. Heat is extracted from the loop through the steam generator based on the feedwater and steam flow models used for the secondary side of the steam generator. These models are described in WCAP-14882-P-A. The models are unchanged by the use of the three-dimensional core model.

2.3.2 Reactor Vessel/Core Model

Figure 2.3-1 shows the reactor vessel and core model used in WCAP-14882-P-A (Reference 26) for a Westinghouse 3-loop plant. Similar models for 2- and 4-loop plants are shown in the reference. The sample application calculations performed in this report uses the same 3-loop model, except for the addition of more axial nodes in the core to facilitate the transfer of the external heat flux.

A sample reactor vessel and core nodalization of a Combustion Engineering (CE) designed analog protection system plant used in the RETRAN analyses is depicted in Figure 2.3-2. The vessel and core nodalization is very similar to a 4-loop Westinghouse-designed plant.

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Figure 2.3-1
Reactor Pressure Vessel Nodalization – Three Loop Plant

a, c

[ ]

Figure 2.3-2
Reactor Pressure Vessel Nodalization – CE-Designed Plant

a, c

[ ]

2.3.3 Protection and Control System Models

The reactor protection system model for the sample plant includes a reactor trip signal which can be initiated on the following functions: Overtemperature and Overpower Delta-T, Steam Generator Level, Neutron Flux, Pressurizer Pressure and Level, RCS flow-related functions, and trips due to the various Safety Injection initiation signals, turbine trip, and manual trip. When the reactor trip signal is reached in the appropriate number of channels, then after the specified trip delay time, a signal to insert the control rods is sent to the SPNOVA code to begin trip rod insertion for the control and shutdown banks. The trip rod position vs. time is controlled within the SPNOVA code, and is adjusted to match the Technical Specification trip time. Selected rod clusters or banks may also be prevented from tripping within the SPNOVA model for additional conservatism. The protection system models are the same as discussed in WCAP-14882-P-A (Reference 26), except that [

] [a, c]. In addition, a high neutron flux reactor trip logic using the individual ex-core detectors can now be modeled, including the assumption of a failure of the best channel.

Reactor control system models are available for the following: Rod Control, Pressurizer Pressure Control, Feedwater Flow Control, Turbine Control, and Pressurizer Level Control. These are the same control functions as discussed in WCAP-14882-P-A. The RETRAN rod control system sends a control rod direction and rod speed (steps/min) demand signal to the SPNOVA code to control the rod motion.

2.3.4 Engineered Safety Features System Models

These models include the Safety Injection System and actuation system models, the High and Low Steam Generator Level signals, Turbine Trip function, Auxiliary Feedwater System and various manual actuations. There is no change in these models from the description in WCAP-14882-P-A.

2.4 Hot Rod Models

In the updated 3-dimensional core transient analysis method, the "hot rod" DNBR and/or "hot rod" peak fuel/clad temperature calculations are performed in VIPRE separately from the 3-D core/RCS loop model transient calculations. The separation of the hot rod model calculation from the average rod model calculation allows separate conservatisms to be applied to the different models. This is the same approach as is used in the current FSAR methodology. The VIPRE models for the hot rod calculations are the same as those described in the NRC-approved topical reports (References 5 and 6). As input to the hot rod calculations, time-dependent core parameters are obtained from the neutron kinetic and system transient codes, including core nuclear power and changes in radial and axial power distributions, core inlet temperature, core outlet pressure and core inlet flow rate. The hot rod models are summarized below.

2.4.1 Hot Rod Model for DNB Evaluation

The DNB evaluation is performed in a separate VIPRE calculation using a subchannel model, with additional conservatism applied to the modeling and initial conditions in order to minimize the calculated DNBR. The subchannel model for the DNB evaluation is the same as that described in the NRC-approved Westinghouse VIPRE modeling topical report (Reference 5). A one-eighth core of a 3-loop PWR with the 17x17 fuel lattice can be modeled in fourteen channels comprised of [ ] [a, c], as illustrated in Figure 2.4.1. There is no change to the channel geometric modeling, heat transfer and two-phase flow correlations, turbulent mixing and flow resistance modeling, or modeling of engineering hot channel factors as compared to the approved model described in Reference 5.

Fuel rods are modeled as "conduction rods" in the VIPRE hot rod model similar to the FACTRAN code (Reference 21) and to the model described in the 3-D RCCA Ejection methodology report (Reference 7). The conduction rod model calculates transient temperature distributions in the fuel rods and heat flux at the rod surfaces, based on core power, changes in radial and axial power distributions, and local fluid conditions. The pellet-to-clad gap heat transfer is calculated using the dynamic gap model in VIPRE. The model is initialized with the bounding fuel temperature generated by a fuel performance code such as the PAD code (Reference 18) using the same calibration method as for the average rod model in the feedback calculation. The rod surface heat flux and local fluid conditions are then input to the DNBR calculation with an NRC-approved DNB correlation applicable to the fuel design.

Figure 2.4-1
VIPRE Multi-Channel Model for 1/8th Core a, c

[ ]

2.4.2 Hot Rod Model for Peak Fuel/Clad Temperature Evaluation

The VIPRE code is used in a stand-alone mode to perform the hot fuel rod thermal calculation for the peak fuel/clad temperature evaluation. It is performed with additional conservatism applied to the modeling and initial conditions in order to maximize the increase in fuel temperature and enthalpy. The hot rod calculation uses the nuclear power, core inlet flow, inlet temperature and core outlet pressure vs. time, and includes the effect of changes in the radial and axial power distribution calculated by SPNOVA.

The hot fuel rod model is based on the NRC-approved model described in the Westinghouse VIPRE modeling topical report (Reference 5), and is similar to the model used in the FACTRAN code (Reference 21). It represents the hottest fuel rod from any assembly in the core. The pellet-to-clad gap heat transfer is calculated using the dynamic gap model in VIPRE. The model is calibrated against bounding fuel rod temperatures as generated by a design fuel performance code such as the PAD program (Reference 18), using the method described above for the average rod model. As for current plant licensing applications, the heat transfer to the coolant is calculated using the Dittus-Boelter correlation for single phase forced convection and the Thom correlation for nucleate boiling. If the fuel rod is predicted to enter into DNB at any axial elevation, the Bishop-Sandberg-Tong correlation (Reference 19) is used for transition and film boiling heat transfer beyond Departure from Nucleate Boiling (DNB). In order to maximize the post-DNB fuel and clad temperature transient, and the amount of predicted clad oxidation, the hot spot is assumed to enter DNB at the beginning of the transient. The Baker-Just correlation (Reference 20) is used to account for heat generation in the cladding material due to the zirconium-water reaction. The use of these models in VIPRE is approved by the NRC (Reference 5).

2.5 Initial Conditions and Accident Assumptions

The initial conditions are the same as those used in the current FSAR analysis of each accident event, including the core-related conservatisms described below and the accident-specific analysis assumptions described in Chapter 3.

a. Initial Power, Temperature and Pressure

Most FSAR accident events which are DNB-limited are analyzed using a statistical methodology (e.g., Westinghouse Revised Thermal Design Procedure (RTDP) as described in Reference 22 or Improved Thermal Design Procedure (ITDP) as described in Reference 23). Other approved statistical methodology could be used but for the case demonstrated herein, with RTDP, the accidents are analyzed using nominal values of the initial conditions of power, temperature, pressure and RCS flow. The uncertainty allowances on these parameters are included in the limit DNBR value on a statistical basis. For accidents which are not DNB limited, or for which the RTDP is not applied, the initial conditions are obtained using the procedure commonly known as the Standard Thermal Design Procedure (STDP). With STDP, the initial conditions are obtained by applying maximum steady-state uncertainty allowances to the rated values in the limiting direction. The uncertainty values are justified on a plant to plant basis and are not affected by the use of the updated 3-D core transient accident analysis methodology.

b. Initial RCS Flow Rate

Accidents employing RTDP assume a minimum measured flow (MMF) or equivalent. An allowance for measurement uncertainty has been incorporated into the DNBR limit. Accidents employing STDP assume a conservative thermal design flow (TDF). The flow rate assumption is confirmed by a flow measurement obtained during plant startup.

c. Reactor Trip

The reactor trip is simulated by dropping any partially or fully withdrawn rod banks into the core, using a conservative control rod cluster acceleration and terminal velocity which yields a trip rod insertion time consistent with the plant Technical Specifications. Additional conservatism in the trip for full power events is added by assuming that the most reactive control rod does not trip, or by conservatively preventing additional banks from inserting.

d. Reactor Trip Point and Trip Time Delay

The reactor trip is assumed to occur when the appropriate number of protection channels reaches the trip setpoint plus the conservative uncertainty allowance. Reactor trip setpoints, uncertainty allowances, and trip time delays are given in the individual plant Technical Specifications. For a reactor trip on a high neutron flux trip signal, the trip function is based on the ex-core detector channel response as inferred from the 3-dimensional core model. For all trip functions, consistent with the single failure criterion, the channel with the "best" (maximum) response is assumed to fail to actuate, thus requiring sufficient additional channels (depending on the trip logic) to actuate in order to cause a trip.

2.6 Application of Conservative Allowances

Conservative allowances on the key analysis parameters will be applied in the calculation using a "deterministic" approach. In the "deterministic" method, the uncertainties in the key parameters are applied in the conservative direction simultaneously in the calculation. This leads to a very conservative result, since the key parameters are not all expected to be at their limiting value at the same time. A more reasonable analysis approach is a "statistical" method in which the "base case" calculation is performed without the uncertainty allowances, and then the uncertainty allowances are applied to the calculation one at a time to generate the explicit impacts on the analysis limit of interest. However, as described in this report, only the deterministic approach will be applied with the updated methodology.

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

- The Doppler feedback can be conservatively pessimized by applying a [ ]^{a,c} multiplier to the change in the fast absorption cross-section for the given change in the calculated fuel effective temperature. This multiplier applies a uniform uncertainty allowance on the Doppler feedback.

- The moderator temperature coefficient (MTC) can be pessimized by [ ] [a, c] by changing the core soluble boron concentration from the calculated critical value. For accidents typically analyzed at the beginning of a fuel cycle (BOC) where a least-negative coefficient is conservative, the boron concentration will be increased to conservatively bound the least-negative calculated value at that time in the cycle. For accident events analyzed at the end of a fuel cycle (EOC) where a most-negative MTC is conservative, a conservative minimum boron concentration will be used to bound the most negative calculated value at that time in the cycle.

- The delayed neutron fraction can be pessimized by [ ] [a, c] by applying a uniform multiplier to the node-by-node values of the delayed neutron fraction.

- The reactor trip rod worth can be pessimistically reduced by either assuming a stuck rod, or by preventing the trip of one or more shutdown banks. A plant-specific trip worth uncertainty will be applied.

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

- The hot rod DNBR calculation will use the same uncertainty allowances as for current licensing applications. The uncertainty allowances used for the thermal-hydraulic initial conditions (power, temperature, pressure, and RCS flow) are described in Section 2.5. The calculated hot rod radial power peaking factor ($F_{\Delta H}$) vs. time is used, multiplied by the current licensed uncertainty allowance. The same uncertainty factor is applied to all other hot rods investigated. The hot rod DNBR model is addressed in more detail in Section 2.4.1.

- The hot rod peak fuel/clad temperature, or maximum fuel enthalpy calculation, will apply the standard uncertainty allowances as for the current licensing applications. This includes allowances for:
  - Local peaking factor uncertainty,
  - Local engineering peaking factor penalties, and
  - Core calorimetric uncertainty for hot full power calculations

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

In addition to the conservative allowances applied to the reactor parameters mentioned above, the plant safety analysis is performed with a number of other conservatisms which are not affected by the implementation of the updated 3-D methodology. Following is a list of some of the additional conservatisms.

- Some Condition III and IV events are analyzed using Condition II criteria,
- Worst (highest worth) stuck rod assumption,
- Broken loop for a steamline break assumed in coincidence with worst stuck rod,
- Conservative rod insertion time for reactor trip,
- Conservative reactor trip setpoints,
- Conservative trip delay times,
- Conservative delay times for ESFAS,
- Best protection system channel failure assumption,
- Minimum shutdown margin at any time in life,
- Worst time in cycle life,
- Worst initial axial offset,
- Conservative reactor coolant pump coastdown characteristics.

2.7 Applicability to Various Reactor Types

Although the accidents chosen for the sample applications shown in Chapter 3 were performed for a 3-loop Westinghouse plant, the methodology is not limited to this plant type. The same computer codes employed here have been used in licensing applications for many Westinghouse-designed 2-, 3- and 4-loop plants, and by Westinghouse for a CE-designed analog protection system plant. The computer codes and method of data transfer between the codes (the external communication interface) are applicable to any PWR for which a licensed model is available for the base codes (e.g., SPNOVA, VIPRE and RETRAN).

A licensed model includes plant-specific variations in the reactor core, RCS primary/secondary system design, reactor control and protection system design, accident limits and specific uncertainty allowances. These models are unaffected by the linking of the codes using the external communication interface. Therefore, the methodology demonstrated in Chapter 3 can be applied to any PWR for which licensed models exist, taking into account the plant-specific variations and uncertainty allowances.

Thus, although there will be differences in the models used for different PWR configurations, these changes are clearly identified in the current licensed models and methodology for that plant. The 3-D application methodology described in this report is therefore independent of the PWR plant type.

2.8 Reload Safety Evaluation Method

The Westinghouse reload safety evaluation (RSE) methodology uses a bounding analysis approach in which key safety analysis parameters are identified which could affect the accident, and which could change as a result of a reload. The safety analysis is performed with reasonably bounding values for these parameters to lessen the chance that normal variations in a fuel cycle design will cause these parameters to be exceeded. In the reload design process, cycle-specific static calculations are performed to determine if the calculated values of the parameters remain bounded by the value used in the licensed safety analysis. If the reload value exceeds the value used in the analysis, an evaluation is performed to determine if the safety analysis must be repeated. This methodology is described in more detail in Reference 15.

The key parameters for the non-LOCA transients which may vary from cycle to cycle as a result of a reload, assuming no change in plant operating characteristics or fuel type, are typically:

- Moderator feedback coefficient,
- Doppler feedback coefficient,
- Delayed neutron fraction,
- Radial and axial peaking factors (power distributions),
- Axial Flux Difference (AFD) operating band,
- Control rod bank differential worths,
- Reactor trip reactivity worth.

Any particular accident may be more or less sensitive to variations in the above parameters.

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3.0 SAMPLE APPLICATION OF 3-D METHODOLOGY

This chapter presents the sample application of the 3-D methodology to a representative 3-loop Westinghouse plant. The method is applied to a subset of the transients which are analyzed for a typical plant safety analysis report. The same methodology would be used in the application to other PWRs and other accident events as discussed in Section 3.6.

3.1 Complete Loss of Forced Reactor Coolant Flow (CLOF Event)

A complete loss of flow accident analysis was performed for two cases: a case using the current analysis method and a case using the updated 3-D core transient methodology, for purposes of comparison. A description of the accident, discussion of the current and updated 3-D core transient methodology, and comparison of the analysis results, are presented below.

3.1.1 Accident Description

A loss of forced reactor coolant loop flow can result from a mechanical or electrical failure in a reactor coolant pump (RCP), from an interruption in the power supplying one or more of these pumps, or from a reduction in RCP motor supply frequency. If the reactor is operating at power, the loss of forced reactor coolant flow could result in departure from nucleate boiling (DNB) in the core. The reactor protection system and reactor coolant pumps are designed to preclude the occurrence of DNB.

The Final Safety Analysis Report (FSAR) bounds a number of flow transients, postulating the loss of power to one or more pumps or reduction in frequency of the power supply. The most limiting event is a complete loss of forced coolant flow, which can occur from an interruption of power to all RCP electrical buses, or a frequency decay event affecting all buses.

3.1.2 Reactor Protection

Several functions are provided to detect the occurrence of a loss of flow and to subsequently trip the reactor. Plant specific protective functions include a subset of the following reactor trips. These include:

- Low Primary Coolant Flow,
- RCP Breaker Opening on one or more loops,
- RCP Undervoltage on the electrical buses supplying two or more RCPs, and
- RCP Underfrequency on the electrical buses supplying two or more RCPs.

For a complete loss of flow accident, depending on the cause of the event, a reactor trip will be actuated on either the undervoltage or underfrequency reactor trip functions.

3.1.3 Accident Limits

Based on its expected frequency of occurrence, the complete loss of flow transient is considered to be a Condition III event, an Infrequent Incident, as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants". However, as presented in the Final Safety Analysis Report (FSAR), the event is analyzed to meet the criteria for Condition II events, Incidents of Moderate Frequency. Per ANSI N18.2-1973, the design criteria for Condition II events are:

- Pressure in the RCS and MSS (Main Steam Supply) system shall be maintained below 110% of the design values.
- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the limit value.
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

3.1.4 Current Analysis Method

The current analysis method case uses the RETRAN computer code (Reference 26) to calculate the loop and core flow during the transient, the time of reactor trip, the nuclear power transient, and the primary and secondary system pressure and temperature transients. The VIPRE computer code (Reference 5) is then used to calculate the heat flux and DNBR transient based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. The momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics. Also, it is based on conservative estimates of system pressure losses.

Since the RCS and MSS pressure rise is not limiting, the event is analyzed to show that the integrity of the core is maintained by showing that the DNBR remains above the safety analysis limit value. The DNBR calculation is performed using the WRB-2 DNB correlation (Reference 13).

This event is analyzed with RTDP (Reference 22). Therefore the initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values, and the uncertainties are included in the DNBR limit. Minimum measured flow is also assumed, with the flow uncertainty included in the DNBR limit.

A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive or least-negative MTC allowed by the plant Technical Specifications for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

A conservatively low trip reactivity value [ ] [a,c] is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux used in the DNB evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A

conservative trip reactivity worth versus rod position is modeled in addition to a conservative rod drop time (e.g., 2.7 seconds to dashpot).

The reactor rod control system is not simulated, since it would act to reduce the reactor power which would lessen the severity of the event. The pressurizer power-operated relief valves and spray are simulated since this minimizes the RCS pressure rise. However, for the DNBR analysis, the hot channel is analyzed using the initial pressure, with no credit for the increase in RCS pressure. This is conservative since the pressure rise results in a DNB benefit.

A reference DNB axial power shape that bounds the cycle operation is assumed in VIPRE for the calculation of DNBR. This shape, in combination with a cycle bounding [

]^{a,c}.

3.1.5 Updated 3-D Transient Neutronics Method and Sample Calculation

a) Computer Codes

The analysis was performed using the NRC-approved SPNOVA, VIPRE and RETRAN computer codes and models, linked by an external communication interface. The computer codes are described in Section 2.1.

The VIPRE code is also used in a separate calculation to determine the hot rod minimum DNBR vs. time. The minimum DNBR is calculated using the subchannel model described in Section 2.4.1.

b) Assumptions Used in the Reactor Core Calculation

The following assumptions are applicable to the reactor core calculations performed for the Complete Loss of Flow event using the SPNOVA/VIPRE computer codes:

Initial Core Conditions: The Complete Loss of Flow calculation was performed at Beginning-of-cycle (BOC) Hot Full Power (HFP) conditions with equilibrium xenon. [

]^{a,c}.

Reactivity Feedback: The analysis used minimum moderator temperature feedback and maximum Doppler feedback, consistent with the current analysis method. [

]a,c.

Delayed Neutron Fraction: The analysis assumed a maximum (bounding) value of the delayed neutron fraction of 0.0072, which is the same as used in the current analysis method.

Trip Reactivity: The control rods were initially assumed to be at their fully withdrawn position to minimize the initial rate of reactivity insertion following a reactor trip. A conservative rod position vs. time curve was assumed, resulting in a drop time of 2.7 seconds to dashpot. (These assumptions are the same as used in the current analysis method for this event.) A conservative value of trip reactivity was obtained by [

]a,c.

c) Assumptions Used in the Reactor Coolant System Calculation

The following assumptions are applicable to the reactor coolant system calculations performed for the Complete Loss of Flow event using the RETRAN computer code:

Initial RCS Conditions: Since the Loss of Flow event is analyzed using the Revised Thermal Design Procedure (RTDP), the analysis was performed using nominal HFP conditions (no uncertainties) for reactor power, RCS average temperature, and pressurizer pressure (Reference 22). The RCS flow rate was set to the Minimum Measured Flow (MMF). All other RCS initial conditions (pressurizer water volume, steam generator level, etc.) were also set to nominal conditions. (These assumptions are the same as for the current methodology for this event.)

Accident Initiation: The accident was initiated by causing a linear decrease in the RCP speed consistent with a frequency decay rate of 5 Hz/s. (This is identical to the current analysis method.)

Reactor Protection: The accident was assumed to trip on the underfrequency reactor trip function at a setpoint of 56.8 Hz including uncertainties, with a trip delay time of 0.6 seconds. (The reactor trip setpoints, delay times and uncertainties assumed are identical to the current analysis method.)

d) DNB Evaluation

The VIPRE code was used in a separate time-dependent calculation to determine the minimum DNBR, based on the core average power, power distribution, inlet temperature, core inlet flow, and core exit pressure vs. time. The core average power and power distribution were obtained from SPNOVA, including the time-dependent changes in radial enthalpy rise hot channel factor ($F_{\Delta H}$) and the axial power distribution. The current methodology pin-by-pin design power distribution (Reference 5), with the peak rod power at the limit allowed by the plant Technical Specifications or the Core Operating Limits Report (COLR), was used as the initial value for the DNBR calculations. The reactor coolant conditions (inlet temperature, core inlet flow and core exit pressure vs. time) were obtained from RETRAN. The same uncertainty allowances in core power, hot channel factors, and coolant conditions were applied in the VIPRE DNB evaluation as in the current methodology. The analysis method is described in more detail in Section 2.4.1. The results are presented in Section 3.1.6 below.

3.1.6 Results and Comparison with Current Method

The complete loss-of-flow event was analyzed for a loss of three RCPs with three loops in operation using both the current analysis method and the updated 3-D core transient analysis method. [

] [a,c]. The minimum DNBR obtained with the two analysis methods is given in Table 3.1-1. The sequence of events is supplied in Table 3.1-2. The results are compared in Figures 3.1-1 to 3.1-6.

[

] [a,c].

[

] [a,c].

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

3.1.7 Summary

The complete loss of flow event was analyzed with the updated 3-D core transient methodology, using conservative core initial conditions indicative of hot full power operation at the beginning of a fuel cycle. The results were compared to the results of the same transient analyzed with the current point-kinetics analysis method. The comparison shows that the updated 3-D core transient methodology results in an increase in the minimum DNBR due to a more realistic prediction of DNB margin. This is attributed primarily to the following factors:

1) [

]a,c.

2) [

]a,c.

3) [

]a,c.

3.1.8 Conclusions

A sensitivity study was performed for the updated 3-D transient neutronics method, which addresses the effect of variations in the initial conditions and assumptions used in the analysis. The sensitivity study is presented in Section C.1 of Appendix C. As a result of the sensitivity study, it is concluded that the analysis assumptions chosen for the base case in Section 3.1.5 define a conservative 3-D methodology for this event, provided that:

1) [

]a,c.

2) [

]a,c.

3) [

]a,c.

This is defined as the Reference Bounding Analysis Case for this event as discussed in Section C.1.4 of Appendix C.

3.1.9 Reload Safety Evaluation

For a reload core using a safety evaluation performed with the updated 3-D transient neutronics methodology, the continued applicability of the safety evaluation will be confirmed as part of the Reload Safety Evaluation (RSE) process described in Section 2.8. For the Complete Loss of Flow event, the core neutronics parameters assumed in the analysis that may vary from cycle-to-cycle as a result of a reload are:

- Moderator feedback coefficient*
- Doppler feedback coefficient
- Delayed neutron fraction
- Radial and axial peaking factors (power distributions)*
- Axial Flux Difference (AFD) operating band*
- Reactor trip reactivity worth*

* **Key parameters – see below.**

Based on the sensitivity study presented in Section C.1 of Appendix C, the transient is not sensitive to [

] [a,c]. These key parameters are not expected to change significantly from cycle-to-cycle unless there is a significant change in the fuel loading pattern. In the reload design process, cycle-specific static calculations are performed to determine if the calculated values of the parameters remain bounded by the values used in the licensed safety analysis. The transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. This methodology is described in more detail in Reference 15.

Table 3.1-1
Complete Loss of Forced Reactor Coolant Flow
(Underfrequency) Analysis Results

Analysis Method	Minimum DNBR	Time of Min. DNBR (sec.)*
Current Point-Kinetic Methodology	[ ] a, c	[ ] a, c
Updated 3-D Core Transient Method	[ ] a, c	[ ] a, c

* *From the start of the event. (Includes a 1-second delay to the initiation of the flow coastdown.)*

Table 3.1-2
Complete Loss of Forced Reactor Coolant Flow
(Underfrequency) Sequence of Events
(Updated 3-D Core Transient Method)

Event	Time (seconds)
Transient Begins	[ ] a, c
Frequency Decay Initiated	[ ] a, c
RCP Underfrequency Trip Setpoint Reached	[ ] a, c
Rods Begin to Drop	[ ] a, c
Minimum DNBR Occurs	[ ] a, c

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Figure 3.1-1
Complete Loss of Forced Reactor Coolant Flow
Nuclear Power vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.1-2
Complete Loss of Forced Reactor Coolant Flow
Core Average Heat Flux vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.1-3
Complete Loss of Forced Reactor Coolant Flow
Reactor Coolant Flow vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.1-4
Complete Loss of Forced Reactor Coolant Flow
Pressurizer Pressure vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.1-5
Complete Loss of Forced Reactor Coolant Flow
$F_{\Delta H}$ and Axial Offset vs. Time
Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.1-6
Complete Loss of Forced Reactor Coolant Flow
Minimum DNBR vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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3.2 Reactor Coolant Pump Locked Rotor (Rods-in-DNB Evaluation)

A RCP locked rotor accident analysis (rods-in-DNB evaluation) was performed for two cases, a case using the current analysis method and a case using the updated 3-D transient neutronics methodology, for purposes of comparison. A description of the accident, discussion of the current and updated 3-D transient neutronics methodology, and a comparison of the analysis results, are presented below.

3.2.1 Accident Description

The postulated locked rotor accident is an instantaneous seizure of a reactor coolant pump (RCP) rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal. However, the reactor trip may not occur sufficiently fast to prevent DNB from occurring. If DNB occurs, the number of rods entering DNB is assessed for the radiological release evaluation, and a hot spot fuel and clad temperature evaluation is performed to ensure continued core coolability. The flow reduction also causes a rapid heatup of the coolant in the reactor core, and a reduction in heat removal in the steam generators, resulting in an increase in RCS pressure.

The consequences of a postulated pump shaft break accident are similar to the locked rotor event. With a broken shaft, the impeller is assumed to be free to spin, as opposed to it being fixed in position for a locked-rotor event. Therefore, the initial rate of reduction in core flow is greater during a locked-rotor event than in a pump shaft break event because the fixed shaft causes greater resistance than a free-spinning impeller early in the transient, when flow through the affected loop is in the positive direction. As the transient continues, the flow direction through the affected loop is reversed. If the impeller is free to spin as the flow reverses in the affected loop, this would result in a larger reverse flow in the loop, and the net core inlet flow rate will be less than if the RCP impeller is assumed to be seized. Because peak pressure, cladding temperature, and DNB occur very early in the transient, the reduction in core flow during the period of forward flow in the affected loop dominates the severity of the results. Consequently, the bounding results for the locked-rotor transients also are applicable to the RCP shaft break.

A loss of offsite power is conservatively assumed to occur at the time of reactor trip, causing the unaffected RCPs to lose power and coast down freely.

The calculation of the number of rods in DNB and the hot spot fuel and clad temperature calculation will be addressed in this section. The peak RCS pressure will be addressed in Section 3.3.

3.2.2 Reactor Protection

The locked rotor event results in a rapid loss of flow in one of the operating loops. At high power levels, a reactor trip will occur when measured RCS flow rate falls below the reactor trip setpoint.

3.2.3 Accident Limits

The locked-rotor event is classified as an ANS Condition IV "Limiting Fault" as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants." Limiting faults are not expected to occur, but are postulated because their consequences would include the potential release of significant amounts of radioactive material. The event is conservatively analyzed to demonstrate that the following acceptance criteria are met:

- Pressure in the primary and secondary RCS must be maintained below that which would cause the stresses to exceed the faulted condition stress limits, which translates to Service Level D of the ASME code. For ease of interpretation, the more restrictive criterion of Service Level C (equivalent to emergency condition stress limits) is applied. Some plants assume more restrictive criteria.
- Coolable core geometry is ensured by showing that the peak cladding temperature and maximum oxidation level for the hot spot do not exceed their respective limits.
- Activity release is such that the calculated doses meet 10 CFR Part 100 guidelines.

For the locked-rotor event, a primary RCS overpressure analysis is performed to demonstrate that the first criterion is met (see Section 3.3). Since the loss-of-load analysis bounds the locked rotor due to a conservative analysis method, a specific MSSS overpressurization analysis is not performed for the locked-rotor event. Additionally, a hot-spot evaluation is performed to calculate the peak cladding temperature and maximum oxidation level to address the second criterion. Finally, a calculation of the "rods-in-DNB" is performed for input to the radiological dose analysis to address the third criterion.

The "rods-in-DNB" criterion and the hot-spot fuel and clad temperature calculation are being addressed in this section. The RCS overpressure is addressed in Section 3.3.

3.2.4 Current Analysis Method

The current analysis method case uses the RETRAN computer code (Reference 26) to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary and secondary system pressure and temperature transients. The VIPRE computer code (Reference 5) is then used to calculate the heat flux and DNBR transient based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. The momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics. Also, it is based on conservative estimates of system pressure losses.

A loss-of-offsite-power is assumed to occur at the time of reactor trip resulting in a coastdown of the unaffected RCPs. No delay is conservatively assumed between the time of loss-of-offsite-power and the time that the coastdown of the unaffected RCPs begins. The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. The momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics. Also, it is based on conservative estimates of system pressure losses.

The locked rotor rods-in-DNB event is analyzed with the RTDP (Reference 22). Therefore the initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values, and the uncertainties are included in the DNBR limit. Minimum measured flow is also assumed, with the flow uncertainty incorporated into the DNBR limit. DNBR is predicted using the WRB-2 DNB correlation (Reference 13).

A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive or least-negative MTC allowed by the plant Technical Specifications for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

A conservatively low trip reactivity value [ ] [a,c] is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux used in the DNB evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position is modeled in addition to a conservative rod drop time (e.g., 2.7 seconds to dashpot).

The reactor rod control system is not simulated, since it would act to reduce the reactor power which would lessen the severity of the event. The pressurizer power-operated relief valves and spray are simulated since this minimizes the RCS pressure rise. However, for the rods-in-DNB analysis, the hot channel is analyzed using the initial conditions, with no credit for the increase in RCS pressure. This is conservative since the pressure rise would result in a DNB benefit.

A limiting [ ] [a,c] DNB axial power shape is assumed in VIPRE for the calculation of DNBR, similar to the Complete Loss of Flow event. This shape in combination with a [ ] [a,c] provides the most limiting minimum DNBR for the locked rotor event using the current methodology.

If the minimum DNBR decreases below the DNB limit, an additional hot rod calculation is performed with VIPRE to obtain the post-DNB fuel and clad temperature transient and amount of predicted clad oxidation. The same system transient input vs. time is used, except that the analysis is performed using the Standard Thermal Design Procedure (STDP). Therefore, appropriate uncertainty allowances are added to the initial power, coolant temperature, RCS pressure and flow rate. The analysis uses the conservative assumptions that the hot rod enters into DNB at the beginning of the locked rotor transient, and that the rod power at the hot-spot is at the Technical Specification F_Q limit for full power operation. The peaking factor is assumed to remain constant for the duration of the transient. The results represent the upper limit with respect to cladding temperature and zirconium water reaction. The method of calculation used is described in more detail in WCAP-14565-P-A (Reference 5).

3.2.5 Updated 3-D Transient Neutronics Method and Sample Calculation

a) Computer Codes

The analysis was performed using the NRC-approved SPNOVA, VIPRE and RETRAN computer codes and models, linked by an external communication interface. The computer codes are described in Section 2.1.

The VIPRE code is also used in a separate calculation to determine the hot rod minimum DNBR vs. time. The minimum DNBR is calculated using the subchannel model described in Section 2.4.1.

If the minimum DNBR decreases below the DNB limit, an additional hot rod calculation is performed with VIPRE to obtain the post-DNB fuel and clad temperature transient and amount of predicted clad oxidation. The method of analysis is the same as the current analysis method, except that the [

] [a,c]. The hot rod model for calculating the fuel rod and clad temperatures vs. time is described in Section 2.4.2. The calculations use the approved methods described in WCAP-14565-P-A (Reference 5) with the input changes described above as a result of the 3-D methodology.

b) Assumptions Used in the Reactor Core Calculation

The following assumptions are applicable to the reactor core calculations performed for the Locked Rotor event using the SPNOVA/VIPRE computer codes:

Initial Core Conditions: The Locked Rotor calculation was performed at Beginning-of-Cycle (BOC) Hot Full Power (HFP) conditions with equilibrium xenon. [

] [a,c].

Reactivity Feedback: The analysis used minimum moderator temperature feedback and maximum Doppler feedback, consistent with the current analysis method. [

]^{a,c}.

Delayed Neutron Fraction: The analysis assumed a maximum value of the delayed neutron fraction of 0.0072, which is the same as used in the current analysis method.

Trip Reactivity: The control rods were initially assumed to be at their fully withdrawn position to minimize the initial rate of reactivity insertion following a reactor trip. A conservative rod position vs. time curve was assumed, resulting in a drop time of 2.7 seconds to dashpot. (These assumptions are the same as used in the current analysis method for this event.) A conservative value of trip reactivity was obtained by [

]^{a,c}.

c) Assumptions Used in the Reactor Coolant System Calculation

The following assumptions are applicable to the reactor coolant system calculations performed for the Locked Rotor event using the RETRAN computer code:

Initial RCS Conditions: Since the Locked Rotor-DNB evaluation is analyzed using the Revised Thermal Design Procedure (RTDP), the analysis was performed using nominal HFP conditions (no uncertainties) for reactor power, RCS average temperature, and pressurizer pressure (Reference 22). The RCS flow rate was set to the Minimum Measured Flow (MMF). All other RCS initial conditions (pressurizer water volume, steam generator level, etc.) were also set to nominal conditions. (These assumptions are the same as for the current methodology for this event.)

Accident Initiation: The accident was initiated by causing an immediate halt in the rotational speed of one RCP. A loss of offsite power was conservatively assumed to occur at the time of reactor trip, causing the unaffected RCPs to lose power and coast down freely. (This is identical to the current analysis method.)

Reactor Protection: The accident was assumed to trip on the low flow reactor trip function at a setpoint of 85% flow including uncertainties, with a trip delay time of 1.0 seconds. (The reactor trip setpoints, delay times and uncertainties assumed are identical to the current analysis method.)

d) DNB Evaluation

The VIPRE code was used in a separate time-dependent calculation to determine the minimum DNBR in a number of different assemblies to calculate the percentage of the core reaching DNB. The DNBR calculation was based on the core average power, power distribution, inlet temperature, core inlet flow, and core exit pressure vs. time. The core average power and power distribution were obtained from

SPNOVA, including the time-dependent changes in radial enthalpy rise hot channel factor ($F_{\Delta H}$) and the axial power distribution. The current methodology pin-by-pin design power distribution (Reference 5), with the peak rod power raised to the limit allowed by the plant Technical Specifications or the Core Operating Limits Report (COLR), was used as the initial value for the DNBR calculations. The same peak rod normalization factor was applied to any other assemblies for which the DNB evaluation was performed. The reactor coolant conditions (inlet temperature, core inlet flow and core exit pressure vs. time) were obtained from RETRAN. The same uncertainty allowances in core power, hot channel factors, and coolant conditions were applied in the VIPRE DNB evaluation as in the current methodology. The analysis method is described in more detail in Section 2.4.1. The results are presented in Section 3.2.6 below.

e) Peak Fuel and Clad Temperature Evaluation

The method of analysis is the same as described in Section 3.2.5 d) above, except that the analysis was performed using the Standard Thermal Design Procedure (STDP). Therefore, appropriate uncertainty allowances were added to the initial power, coolant temperature, RCS pressure and flow rate. The analysis used the conservative assumptions that the hot rod enters into DNB at the beginning of the locked rotor transient, and that the initial rod power at the hot-spot is at the Technical Specification F_Q limit for full power operation. These assumptions are the same as for the current methodology, except that the peaking factors were not held constant during the transient. Since the results presented below indicate that the DNBR remained above the limit value for this transient using the updated 3-D core transient methodology, the post-DNB hot-spot fuel/clad temperature and amount of predicted clad oxidation transient was not performed for this study. It should be noted that this result is not expected to occur for all plant designs. If DNB should be predicted to occur, a hot rod post-DNB calculation will be performed using the assumptions addressed in this section. The analysis method is described in more detail in Section 2.4.2.

3.2.6 Results and Comparison with Current Method

The locked rotor event was analyzed assuming an instantaneous seizure of the rotor of one RCP with three loops in operation, using both the current analysis method and the updated 3-D core transient analysis method. [

]^{a,c}. The minimum DNBR for the two cases is given in Table 3.2-1. Table 3.2-2 shows the sequence of events for this transient. The transient results are shown in Figures 3.2-1 to 3.2-6.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

3.2.7 Summary

The Locked Rotor, Rods-in-DNB event was analyzed with the updated 3-D core transient methodology, using conservative core initial conditions indicative of hot full power operation at the beginning of a fuel cycle. The results were compared to the results of the same transient analyzed with the current point-kinetics analysis method. The updated 3-D core transient methodology resulted [

]a,c.

These results show that there can be a substantial decrease in the predicted number of fuel rods entering DNB if the transient is analyzed using the 3-D analysis method. This is attributed primarily to the following factors:

1) [

]a,c.

2) [

]a,c.

3) [

]a,c.

3.2.8 Conclusions

A sensitivity study was performed for the updated 3-D core transient method, which addresses the effect of variations in the initial conditions and assumptions used in the analysis. The sensitivity study is presented in Section C.2 of Appendix C. As result of the sensitivity study it is concluded that the analysis assumptions chosen for the base case in Section 3.2.5 define a conservative 3-D methodology for this event, provided that:

1) [

]a,c.

2) [

]a,c.

3) [

]a,c.

This is defined as the Reference Bounding Analysis Case for this event as discussed in Section C.2.4 of Appendix C.

3.2.9 Reload Safety Evaluation

For a reload core using a safety evaluation performed with the updated 3-D core transient methodology, the continued applicability of the safety evaluation will be confirmed as part of the Reload Safety Evaluation (RSE) process described in Section 2.8. For the Locked Rotor, Rods-in-DNB event, the core neutronics parameters assumed in the analysis that may vary from cycle to cycle as a result of a reload are:

- Moderator feedback coefficient*
- Doppler feedback coefficient
- Delayed neutron fraction
- Radial and axial peaking factors (power distributions)*
- Axial Flux Difference (AFD) operating band*
- Reactor trip reactivity worth*

* **Key parameters – see below.**

Based on the sensitivity study presented in Section C.2 of Appendix C, the transient is not sensitive to [

] [a, c]. These key parameters are not expected to change significantly from cycle to cycle unless there is a significant change in the fuel loading pattern. In the reload design process, cycle-specific static calculations are performed to determine if the calculated values of the parameters remain bounded by the values used in the licensed safety analysis. The transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. This methodology is described in more detail in Reference 15.

Table 3.2-1
Reactor Coolant Pump Locked Rotor
(Rods-in-DNB Evaluation) Analysis Results

Analysis Method	Minimum DNBR	Time of Min. DNBR (sec.)*
Current Point-Kinetic Methodology	[ ] a,c	[ ] a,c
Updated 3-D Core Transient Method	[ ] a,c	[ ] a,c

* From the start of the event. (Includes a 1-second delay to the initiation of the locked rotor.)
[ ] a,c.

Table 3.2-2
Reactor Coolant Pump Locked Rotor
(Rods in DNB Evaluation) Sequence of Events
(Updated 3-D Core Transient Method)

Event	Time (seconds)
Transient Begins	[ ] a,c
Rotor on One Pump Locks	[ ] a,c
Low Flow Reactor Trip Setpoint Reached	[ ] a,c
Rods Begin to Drop	[ ] a,c
Remaining RCPs Begin to Coast Down	[ ] a,c
Minimum DNBR Occurs	[ ] a,c

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Figure 3.2-1
RCP Locked Rotor (Rods-in-DNB Evaluation)
Nuclear Power vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.2-2
RCP Locked Rotor (Rods-in-DNB Evaluation)
Core Average Heat Flux vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.2-3
RCP Locked Rotor (Rods-in-DNB Evaluation)
Reactor Vessel Inlet Flow vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.2-4
RCP Locked Rotor (Rods-in-DNB Evaluation)
Pressurizer Pressure vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.2-5
RCP Locked Rotor (Rods-in-DNB Evaluation)
$F_{\Delta H}$ and Axial Offset vs. Time
Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.2-6
RCP Locked Rotor (Rods-in-DNB Evaluation)
Minimum DNBR vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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3.3 Reactor Coolant Pump Locked Rotor (Peak RCS Pressure Evaluation)

A RCP locked rotor accident analysis (peak RCS pressure evaluation) was performed for two cases, a case using the current analysis method and a case using the updated 3-D core transient methodology, for purposes of comparison. A description of the accident, discussion of the current and updated 3-D core methodology, and comparison of the analysis results, are presented below.

3.3.1 Accident Description

The postulated locked rotor accident is an instantaneous seizure of a reactor coolant pump (RCP) rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal. The flow reduction causes a rapid heatup of the coolant in the reactor core, and a reduction in heat removal in the steam generators, resulting in an increase in RCS pressure. The RCS pressure transient is addressed in this section.

As explained in Section 3.2.1, the results for the locked-rotor transients also are applicable to the RCP shaft break.

A loss of offsite power is conservatively assumed to occur at the time of reactor trip, causing the unaffected RCPs to lose power and coast down freely.

3.3.2 Reactor Protection

The locked rotor event results in a rapid loss of flow in one of the operating loops. At high power levels, a reactor trip will occur when measured RCS flow rate falls below the reactor trip setpoint.

3.3.3 Accident Limits

The locked-rotor event is classified as an ANS Condition IV "Limiting Fault" as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants". Limiting faults are not expected to occur, but are postulated because their consequences would include the potential release of significant amounts of radioactive material. The event is conservatively analyzed to demonstrate that the following acceptance criteria are met:

- Pressure in the primary and secondary RCS should be maintained below that which would cause the stresses to exceed the faulted condition stress limits, which translates to Service Level D of the ASME code. For ease of interpretation, the more restrictive criterion of Service Level C (equivalent to faulted condition stress limits) is applied. (Some plants assume more restrictive criteria.)
- Coolable core geometry is ensured by showing that the peak cladding temperature and maximum oxidation level for the hot spot do not exceed their respective limits.
- Activity release is such that the calculated doses meet 10 CFR Part 100 guidelines.

Only the peak RCS pressure criterion is being addressed in this section. The rods-in-DNB and peak cladding temperature/maximum oxidation level aspect of the event is addressed in Section 3.2.

3.3.4 Current Analysis Method

The current analysis method for the Locked Rotor, Peak RCS pressure case uses the same computer codes and calculational methods as for the Locked Rotor, Rods-in-DNB case (Section 3.2.4) except that the Standard Thermal Design Procedure (STDP) is used to obtain the initial conditions, and the pressurizer pressure control system is made inactive to enhance the RCS pressure rise. This is addressed below.

Since the Standard Thermal Design Procedure (STDP) is used, the event was analyzed with a +2% uncertainty in the initial reactor power, a +6 °F uncertainty in RCS temperature, and a +50 psi uncertainty in pressurizer pressure to allow for uncertainties in the pressurizer pressure measurement and control channels. (These uncertainties are plant-dependent.) The Thermal Design Flow is also assumed. This results in calculating the highest possible rise in the coolant pressure during the transient, which occurs in the lower plenum of the reactor vessel.

The same conservative Doppler power coefficient, moderator temperature coefficient, trip reactivity worth and trip rod position vs. time as used for the locked rotor, rods-in-DNB case (Section 3.2.4) are used for the locked rotor, peak RCS pressure case.

The reactor rod control system is not simulated, since it would act to reduce the reactor power which would lessen the severity of the event. The pressurizer pressure control system (power-operated relief valves and spray) are not simulated in order to maximize the RCS pressure rise.

3.3.5 Updated 3-D Transient Neutronics Method and Sample Calculation

a) Computer Codes

The locked rotor, peak RCS pressure calculation uses the same SPNOVA, VIPRE and RETRAN computer codes as for the locked rotor, rods-in-DNB evaluation. Refer to Section 2.1 for a description of the computer codes used.

b) Assumptions Used in the Reactor Core Calculation

The following assumptions are applicable to the reactor core calculations performed for the Locked Rotor event using the SPNOVA/VIPRE computer codes:

<u>Initial Core Conditions</u>: The locked rotor, peak RCS pressure calculation was performed at Beginning-of-Cycle (BOC) Hot Full Power (HFP) conditions with equilibrium xenon. [

]a,c.

Reactivity Feedback, Delayed Neutron Fraction, Trip Reactivity: The moderator and Doppler reactivity feedback, delayed neutron fraction and trip reactivity assumptions are identical to those used in the locked rotor, rods-in-DNB evaluation.

c) Assumptions Used in the Reactor Coolant System Calculation

The following assumptions are applicable to the reactor coolant system calculations performed for the Locked Rotor event using the RETRAN computer code:

Initial RCS Conditions: Since the Locked Rotor, peak RCS pressure case is analyzed using the Standard Thermal Design Procedure (STDP), the analysis was performed using a +2% uncertainty in the initial reactor power, a +6 °F uncertainty in RCS temperature, and a +50 psi uncertainty in pressurizer pressure. These uncertainties are typical values, and may change from plant-to-plant. The RCS flow rate was set to the Thermal Design Flow (TDF). All other RCS initial conditions (pressurizer water volume, steam generator level, etc.) were set to nominal conditions. These assumptions are the same as for the current methodology for this event.

Accident Initiation and Reactor Protection: The same accident initiation and reactor protection functions were assumed as in the locked rotor, rods-in-DNB case (Section 3.2.5). These are also identical to the current methodology.

3.3.6 Results and Comparison with Current Method

The locked rotor, peak RCS pressure event was analyzed assuming an instantaneous seizure of the rotor of one RCP with three loops in operation, using both the current analysis method and the updated 3-D core transient analysis method. [

]a,c.

The peak RCS pressure results for the two cases are given in Table 3.3-1. Table 3.3-2 shows the sequence of events for this transient. The transient results are compared in Figures 3.3-1 to 3.3-5.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

3.3.7 Summary

The locked rotor, peak RCS pressure event was analyzed with the updated 3-D core transient methodology, using conservative core initial conditions indicative of hot full power operation at the beginning of a fuel cycle. The results were compared to the results of the same transient analyzed with the current point-kinetics analysis method. The peak RCS pressure results are shown in Table 3.3-1, and the sequence of events for the 3-D analysis case are presented in Table 3.3-2. [

]a,c.

These results show that there is a substantial gain in RCS pressure margin in analyzing this transient with the 3-D core transient analysis methodology. This is attributed primarily to the following factors:

1) [

]a,c.

2) [

]a,c.

3.3.8 Conclusions

A sensitivity study was performed for the updated 3-D core transient method, which addresses the effect of variations in the initial conditions and assumptions used in the analysis. The sensitivity study is presented in Section C.3 of Appendix C. As result of the sensitivity study, it is concluded that the analysis assumptions chosen for the base case in Section 3.3.5 define a conservative 3-D methodology for this event, provided that, to [

]a,c.

This case therefore represents the Reference Bounding Analysis Case for this event as discussed in Section C.3.4 of Appendix C.

3.3.9 Reload Safety Evaluation

For a reload core using a safety evaluation performed with the updated 3-D core transient methodology, the continued applicability of the safety evaluation will be confirmed as part of the Reload Safety Evaluation (RSE) process described in Section 2.8. For the Locked Rotor, Peak RCS Pressure or Peak Fuel and Clad Temperature evaluation, the core neutronics parameters assumed in the analysis which may vary from cycle-to-cycle as a result of a reload are:

- Moderator feedback coefficient*
- Doppler feedback coefficient
- Delayed neutron fraction
- Radial and axial power peaking factors
- Axial Flux Difference (AFD) operating band*
- Reactor trip reactivity worth*

* **Key parameters – see below.**

Based on the sensitivity study presented in Section C.3 of Appendix C, the transient is not sensitive to [

]a,c. These key parameters are not expected to

change significantly from cycle-to-cycle unless there is a significant change in the fuel loading pattern. In the reload design process, cycle-specific static calculations are performed to determine if the calculated values of the parameters remain bounded by the value used in the licensed safety analysis. The transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. This methodology is described in more detail in Reference 15.

Table 3.3-1
Reactor Coolant Pump Locked Rotor
(Peak RCS Pressure Evaluation) Analysis Results

Analysis Method	Peak RCS Pressure, psia	Time of Peak Pressure (sec.)*
Current Point-Kinetic Methodology	[ ]a,c	[ ]a,c
Updated 3-D Core Transient Method	[ ]a,c	[ ]a,c

* **From the start of the event. (Includes a 1-second delay to the initiation of the locked rotor.)**

Table 3.3-2
Reactor Coolant Pump Locked Rotor
(Peak RCS Pressure Evaluation) Sequence of Events
(Updated 3-D Core Transient Method)

Event	Time (seconds)
Transient Begins	[ ]a,c
Rotor on One Pump Locks	[ ]a,c
Low Flow Reactor Trip Setpoint Reached	[ ]a,c
Rods Begin to Drop	[ ]a,c
Remaining RCPs Begin to Coast Down	[ ]a,c
Maximum RCS Pressure Occurs	[ ]a,c

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Figure 3.3-1
RCP Locked Rotor (Peak RCS Pressure Evaluation)
Nuclear Power vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.3-2
RCP Locked Rotor (Peak RCS Pressure Evaluation)
Core Average Heat Flux vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.3-3
RCP Locked Rotor (Peak RCS Pressure Evaluation)
RCS Loop Flows vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.3-4
RCP Locked Rotor (Peak RCS Pressure Evaluation)
RCS Pressure vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.3-5
RCP Locked Rotor (Peak RCS Pressure Evaluation)
Pressurizer Surge vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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3.4 Steamline Break at Hot Full Power (HFP)

A steamline break at hot full power (HFP) accident analysis was performed for two cases, a case using the current analysis method and a case using the updated 3-D core transient methodology, for purposes of comparison. A description of the accident, discussion of the current and updated 3-D core methodology, and comparison of the analysis results, are presented below.

3.4.1 Accident Description

A rupture in the main steam system piping from an at-power condition creates an increased steam load, which extracts an increased amount of heat from the RCS via the steam generators, resulting in a reduction in RCS temperature and pressure. In the presence of a strong negative moderator temperature coefficient, typical of end-of–cycle conditions, the colder core inlet coolant temperature causes the core power to increase from its initial level due to the positive reactivity insertion. The power approaches a level equal to the total steam flow. Depending on the break size, the reactor may trip due to an over-power condition, or as a result of a steamline break protection function actuation.

3.4.2 Reactor Protection

The reactor protection for a steamline break at power involves in the short term the initiation of a reactor trip. In the long term, reactor protection is provided by the initiation of safety injection, isolation of main feedwater, steamline isolation, and initiation of auxiliary feedwater. The protective functions vary from plant-to-plant, and are addressed in Reference 26. The specific protection functions assumed for this analysis were:

Reactor Trip: A reactor trip signal is provided for overpower protection by the Overpower Delta-T (OPΔT) trip function. This function is actuated on receipt of the signal in two-out-of-three loops for a three-loop plant. (The trip logic is two-out-of-four for a 2- or 4-loop plant). A reactor trip may also occur on a low steamline pressure signal (see SIS actuation discussed below.) Whether the reactor trip occurs on an OPΔT or Low Steamline Pressure signal depends on the break size.

Safety Injection System Actuation: The SI system is assumed to be actuated by a low steamline pressure signal in two-out-of-three steamlines. This results in steamline isolation, feed line isolation and auxiliary feedwater start. SI actuation also causes a reactor trip.

3.4.3 Accident Limits

Depending on the size of the break, this event may be classified as either an ANS Condition III "Infrequent Fault" or Condition IV "Limiting Fault". For all break sizes, current Westinghouse practice is to analyze the event to meet the more conservative Condition II "Incidents of Moderate Frequency" criteria. The acceptance criteria associated with this event typically include the following:

- The dose limit for activity release shall not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

- Pressures in the reactor coolant system and main steam supply system shall be maintained below 110% of the design pressures.

- The fuel temperature and clad strain limits shall not be exceeded. This is ensured by limiting the peak linear heat generation rate to a value below which would cause fuel centerline melting.

The limiting conditions that may be challenged during this event are the peak critical heat flux and peak linear heat generation rate. The evaluation is performed to show that the above criteria are met by ensuring that the minimum DNBR remains above the limit value and that the peak linear heat rate (kW/ft) does not exceed the value which would cause fuel centerline melt.

3.4.4 Current Analysis Method

In the current analysis method, the RETRAN computer code (Reference 26) is used to determine the reactor conditions resulting from a steamline break at hot full power. The code models the core neutron kinetics, RCS loops, pressurizer, steam generators, safety injection system and the main and auxiliary feedwater system. The code also models the reactor protection system, engineered safeguards features actuation system, and control systems. The code computes the pertinent variables, including the core nuclear power and heat flux transients and the RCS temperature and pressure vs. time. The resulting reactor conditions are then used as input to the detailed thermal-hydraulic digital computer code, VIPRE (Reference 5), to determine if DNB occurs. The radial and axial power distributions needed by VIPRE are provided from a detailed 3-dimensional static core model using the ANC code (Reference 10).

The analysis is performed using the Revised Thermal Design Procedure (RTDP, Reference 22) wherein uncertainties on RCS initial conditions (power, temperature, pressure and flow) are included in the development of the DNBR limit value. The minimum measured flow is also used. DNBR is predicted using the WRB-2 DNB correlation (Reference 13).

A conservative minimum Doppler-only power coefficient is used, along with a conservative most-negative end-of-cycle moderator temperature coefficient. These assumptions maximize the core power increase during the transient.

A conservatively low trip reactivity value [ ] [a, c] is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux used in the DNB evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position is modeled in addition to a conservative rod drop time (2.7 seconds to dashpot).

The transient is analyzed assuming no automatic rod control, since automatic control would initially act to insert the rods due to the primary to secondary power imbalance and minimize the core power rise. No pressurizer pressure control (pressurizer heaters) is assumed.

The analysis examines a spectrum of break sizes to determine the limiting break. The limiting break is a break that results in the closest approach to the DNBR limit, and is expected to be an intermediate-size break which trips on the overpower delta-T (OPΔT) signal, just avoiding a trip on low steamline pressure. (Larger breaks trip more rapidly on low steamline pressure before there is a significant reduction in the minimum DNBR.)

3.4.5 Updated 3-D Transient Neutronics Method and Sample Calculation

a) Computer Codes

The analysis was performed using the NRC-approved SPNOVA, VIPRE and RETRAN computer codes and models, linked by an external communication interface. The computer codes are described in Section 2.1.

The VIPRE code is also used in a separate calculation to determine the hot rod minimum DNBR vs. time. The minimum DNBR is calculated using the subchannel model described in Section 2.4.1.

The fuel centerline melt criterion is checked by ensuring that the maximum calculated peak linear heat rate from the SPNOVA code remains below the fuel centerline melt limit. Alternatively, a separate VIPRE calculation could be performed using the hot rod model for the peak fuel/clad temperature evaluation as described in Section 2.4.2.

b) Assumptions Used in the Reactor Core Calculation

The following assumptions are applicable to the reactor core calculations performed for the steamline break at HFP event using the SPNOVA/VIPRE computer codes:

Initial Core Conditions: The steamline break at hot full power calculation was performed at end-of-cycle (EOC) conditions with the reactor at hot full power (HFP). [

]^{a,c}.

Reactivity Feedback: The analysis used maximum moderator temperature feedback and minimum Doppler feedback, consistent with the current analysis method. [

]a,c.

Delayed Neutron Fraction: The analysis assumed a minimum (bounding) value of the delayed neutron fraction of 0.0045, consistent with the EOC condition. This is the same value as was used in the current analysis method.

Trip Reactivity: The control rods were initially assumed to be at their fully withdrawn position to minimize the initial rate of reactivity insertion following a reactor trip. A conservative rod position vs. time curve was assumed, resulting in a drop time of 2.7 seconds to dashpot. (These assumptions are the same as used in the current analysis method for this event.) A conservative value of trip reactivity was obtained by [

]a,c.

c) Assumptions Used in the Reactor Coolant System Calculation

The following assumptions are applicable to the reactor coolant system calculations performed for the steamline break at HFP event using the RETRAN computer code:

Initial RCS Conditions: Since the steamline break at hot full power event is analyzed using the Revised Thermal Design Procedure (RTDP), the analysis was performed using nominal HFP conditions (no uncertainties) for reactor power, RCS average temperature, and pressurizer pressure (Reference 22). The RCS flow rate was set to the Minimum Measured Flow (MMF). All other RCS initial conditions (pressurizer water volume, steam generator level, etc.) were also set to nominal conditions. No automatic rod control is assumed. (These assumptions are the same as for the current methodology for this event.)

Accident Initiation: The accident was initiated by assuming an instantaneous break in one of the steamlines. The break size was varied to find the limiting break resulting in the minimum DNBR. (These assumptions are the same as for the current methodology for this event.)

Reactor Protection: The accident resulted in a reactor trip on the overpower delta-T (OPΔT) function. The trip delay time (to the start of rod motion) was 2.0 seconds. (The trip setpoint, uncertainty, and delay time are the same as for the current analysis method.)

d) DNB Evaluation

The VIPRE code was used in a separate time-dependent calculation to determine the minimum DNBR, based on the core average power, power distribution, inlet temperature, core inlet flow, and core exit pressure vs. time. The core average power and power distribution were obtained from SPNOVA, including the time-dependent changes in radial enthalpy rise hot channel factor ($F_{\Delta H}$) and the axial power distribution. The current methodology pin-by-pin design power distribution (Reference 5), with the peak rod power at the limit allowed by the plant Technical Specifications or the Core Operating Limits Report (COLR), was used as the initial value for the DNBR calculations. The reactor coolant conditions (inlet temperature, core inlet flow and core exit pressure vs. time) were obtained from RETRAN. The same uncertainty allowances in core power, hot channel factors, and coolant conditions were applied in the VIPRE DNB evaluation as in the current methodology. The results are presented in Section 3.4.6 below.

3.4.6 Results and Comparison with Current Method

Case With No Automatic Rod Control

The steamline break at hot full power event was analyzed without automatic rod control for both the current analysis method and the updated 3-D core transient analysis method, assuming several break sizes between 0.3 to 0.8 ft^2. [

]^{a,c}. The minimum DNBR obtained with the two methods is shown in Table 3.4-1. The sequence of events is supplied in Table 3.4-2. The results for the two methods are compared in Figures 3.4-1 to 3.4-4.

[

]^{a,c}.

[

]^{a,c}.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

Case with Automatic Rod Control

The analysis for the base case was repeated assuming the rod control system is in the automatic control mode. The control system model in RETRAN is unaffected by the updated 3-D core transient method. [

]a,c.

[

]a,c.

3.4.7 Summary

The steamline break event from hot full power event was analyzed with the updated 3-D core transient methodology. The selection of parameter values assumed for the 3-D analysis case was consistent with those used in the current point-kinetics methodology. The results were compared to the results of the same transient analyzed with the current point-kinetics analysis method. The minimum DNBR obtained with the two methods is shown in Table 3.4-1. The 3-D method resulted in [

]^{a,c}.

3.4.8 Conclusions

A sensitivity study was performed for the updated 3-D core transient method, which addresses the effect of variations in the initial conditions and assumptions used in the analysis. The study showed that the most limiting nuclear and thermal power transient is associated with [

]^{a,c}.

This case therefore represents the Reference Bounding Analysis Case for DNB evaluation for this event as discussed in Section C.4.4 of Appendix C. [

]^{a,c}.

It should be noted that the analysis of this event for a small number of Westinghouse plants assumes that as a result of environmental effects, the control rods move outward in an uncontrolled manner coincident with the event. For these plants, the 3-D method analysis would be performed [

]^{a,c}.

3.4.9 Reload Safety Evaluation

For a reload core using a safety evaluation performed with the updated 3-D core transient methodology, the continued applicability of the safety evaluation will be confirmed as part of the Reload Safety Evaluation (RSE) process described in Section 2.8. For the Steamline Break at Hot Full Power event, the

core neutronics parameters assumed in the analysis that may vary from cycle-to-cycle as a result of a reload are:

- Moderator feedback coefficient*
- Doppler feedback coefficient
- Delayed neutron fraction
- Radial and axial peaking factors (power distributions)*
- Axial flux difference (AFD) operating band*
- Reactor trip reactivity worth

* **Key parameters – see below.**

Based on the sensitivity study presented in Section C.4 of Appendix C, the transient is not sensitive to significant variations in the [

] [a, c]. These key parameters are not expected to change significantly from cycle-to-cycle unless there is a significant change in the fuel loading pattern. In the reload design process, cycle-specific static calculations are performed to determine if the calculated values of the parameters remain bounded by the value used in the licensed safety analysis. The transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. This methodology is described in more detail in Reference 15.

For this event, a cycle-specific calculation will continue to be performed to confirm that the core acceptance criteria are met for the fuel reload.

Table 3.4-1
Steamline Break at Hot Full Power Analysis Results

Analysis Method	Minimum DNBR	Time of Min. DNBR (sec.)
Current Point-Kinetic Methodology	[ ] a,c	[ ] a,c
Updated 3-D Core Transient Method	[ ] a,c	[ ] a,c

Table 3.4-2
Steamline Break at Hot Full Power Sequence of Events
(Updated 3-D Core Transient Method)

Event	Time (seconds)
Transient Begins	[ ] a,c
Steamline Rupture Initiated	[ ] a,c
Minimum DNBR Occurs	[ ] a,c
Overpower Delta-T Reactor Trip Setpoint Reached in 2 Loops	[ ] a,c
Rods Begin to Drop	[ ] a,c

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Figure 3.4-1
Steamline Break at HFP
Nuclear Power and Core Average Heat Flux vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.4-2
Steamline Break at HFP
Reactor Vessel Inlet Temperature vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.4-3
Steamline Break at HFP
Pressurizer Pressure vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

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Figure 3.4-4
Steamline Break at HFP
Steam Pressure vs. Time
Current Method vs. Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.4-5
Steamline Break at HFP
$F_{\Delta H}$ and Axial Offset vs. Time
Updated 3-D Core Transient Method

a, c

[ ]

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Figure 3.4-6
Steamline Break at HFP
Minimum DNBR vs. Time
Updated 3-D Core Transient Method

a, c

[ ]

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3.5 Main Steamline Break at Hot Zero Power (HZP)

A main steamline break (large break) at hot zero power (HZP) accident analysis was performed for two cases, a case using the current analysis method and a case using the updated 3-D core transient methodology, for purposes of comparison. A description of the accident, discussion of the current and updated 3-D core methodology, and comparison of the analysis results, are presented below.

3.5.1 Accident Description

A main steamline break transient results in an uncontrolled increase in steam flow from the steam generators, with the flow decreasing as the steam pressure drops. This steam flow release increases the heat removal from the RCS, resulting in a decrease in the RCS temperature and pressure. Due to the presence of a negative moderator temperature coefficient (MTC), the RCS cooldown results in a positive reactivity insertion, and consequently a reduction of the core shutdown margin. If the most reactive RCCA is assumed to be stuck in its fully withdrawn position after reactor trip, the possibility is increased that the core will become critical and return to power. A return to power following a steamline break is of concern due to the high-power peaking factors that may exist when the most reactive RCCA is stuck in its fully withdrawn position. This could result in DNB in the high power region of the core, possibly leading to localized fuel rod damage. The response of the core to a steamline break event is therefore analyzed to ensure the core remains in place and intact. Following a steamline break, the core is ultimately shut down by the boric acid injected into the RCS by the emergency core cooling system (safety injection).

3.5.2 Reactor Protection

The reactor protection for a main steamline break involves the initiation of safety injection, isolation of main feedwater, steamline isolation, and initiation of auxiliary feedwater. If the reactor is not already tripped, the reactor trip would occur on the SI signal. The protective functions vary from plant-to-plant, and are addressed in more detail in Reference 26. The specific protection functions assumed for this analysis were:

Safety Injection System Actuation: The SI system was assumed to be actuated by a low steamline pressure signal in two-out-of-three steamlines. SI actuation can also be provided on two-out-of-three low pressurizer pressure signals. SI actuation would also cause a reactor trip.

Main Feedwater Isolation: Feedwater isolation is actuated by the safety injection signal. This signal also initiates auxiliary feedwater.

Main Steamline Isolation: Steamline isolation was assumed to be actuated by the low steamline pressure signal in two-out-of-three steamlines. Due to the provision of redundant isolation valves, only one steam generator can blow down completely, even if one of the isolation valves fails to close.

3.5.3 Accident Limits

The main steamline break event is classified as an ANS Condition IV "Limiting Fault" as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants." Limiting faults are not expected to occur, but are postulated because their consequences would include the potential release of significant amounts of radioactive material. The event is conservatively analyzed to demonstrate that there is no consequential damage to the primary system and that the core remains in place and intact.

Although DNB and fuel cladding damage are not necessarily unacceptable consequences of a steamline break transient, the analysis demonstrates that there is no consequential damage to the primary system, and that the core remains in place and intact, by showing that the DNB design basis is satisfied following a steamline break.

3.5.4 Current Analysis Method

In the current analysis method, the RETRAN computer code (Reference 26) is used to determine the reactor conditions resulting from a main steamline break. The code models the core neutron kinetics, RCS loops, pressurizer, steam generators, safety injection system and the main and auxiliary feedwater system. The break flow is modeled using the Moody critical flow correlation assuming dry saturated steam. Perfect moisture separation is assumed unless the mixture level reaches the top of the steam generator. The code computes the pertinent variables, including the core nuclear power and heat flux transients and the RCS temperature and pressure vs. time. The code calculates the time the safety injection actuation signal is reached, and initiates safety injection, main feedwater isolation, and auxiliary feedwater start. The resulting reactor conditions are then used as input to the detailed thermal-hydraulic digital computer code, VIPRE, (Reference 5) to determine if DNB occurs. The radial and axial power distributions needed by VIPRE are provided from a detailed 3-dimensional static core model using the ANC code (Reference 10). Because of the low pressure condition at the limiting time steps, DNBR is predicted using the W-3 DNB correlation (Reference 14).

The transient calculation is performed in RETRAN at end-of-cycle (EOC) starting from a critical condition at hot zero power (HZP). The reactor is assumed to trip at the start of the event, using a trip worth equal to the minimum required shutdown margin at EOC hot zero power (no-load) conditions with the most reactive RCCA assumed to be stuck in its fully withdrawn position. The calculation uses a highly negative moderator temperature coefficient, and a Doppler power feedback model corresponding to an EOC rodded core with the most reactive RCCA removed (N-1 condition). The stuck RCCA is assumed to be conservatively located in the core sector near the loop with the faulted steam generator. All reactivity feedback parameters (MTC, Doppler power coefficient) are weighted toward the core sector exposed to the greatest cooldown from the faulted loop.

Additional assumptions regarding the safety injection flow, boron concentration, feed and steamline isolation, and auxiliary feedwater flow can be found in a typical FSAR write-up for this event.

3.5.5 Updated 3-D Transient Neutronics Method and Sample Calculation

a) Computer Codes

The analysis was performed using the NRC-approved SPNOVA, VIPRE and RETRAN computer codes and models, linked by an external communication interface. The computer codes are described in Section 2.1.

The VIPRE code is also used in a separate calculation to determine the hot rod minimum DNBR vs. time. The minimum DNBR is calculated using the subchannel model described in Section 2.4.1.

b) Assumptions Used in the Reactor Core Calculation

The following assumptions are applicable to the reactor core calculations performed for the main steamline break at hot zero power event using the SPNOVA/VIPRE computer codes:

Initial Core Conditions: The main steamline break transient calculation was performed at end-of-cycle (EOC) starting from a critical condition at hot zero power (HZP). [

]^{a,c}.

Reactivity Feedback: The analysis used maximum moderator temperature feedback and minimum Doppler feedback, consistent with the current analysis method. [

]^{a,c}.

Delayed Neutron Fraction: The analysis assumed a minimum (EOC) value of the delayed neutron fraction of 0.0045, which is the same as was used in the current analysis method.

Shutdown Margin: A calculation was performed using an end-of-cycle (EOC) shutdown margin of 1.77% Δk/k at hot zero power (no-load) conditions, assuming the most reactive RCCA is stuck in its fully withdrawn position. This is a typical shutdown margin value for this type of plant. [

]^{a,c}.

Calculations were also performed using a minimum shutdown margin of only 1.0% Δk/k. [

]^{a,c}.

c) **Assumptions Used in the Reactor Coolant System Calculation**

The following assumptions are applicable to the reactor coolant system calculations performed for the main steamline break event at hot zero power event using the RETRAN computer code:

Initial RCS Conditions: Since the reactor is at hot zero power, the analysis was performed using the HZP (no-load) reactor vessel inlet temperature and nominal pressurizer pressure. The RCS flow rate was set to the thermal hydraulic design value assuming all RCPs in operation. (These assumptions are the same as used in the current analysis method.)

Accident Initiation: The accident was initiated by assuming the complete severance of a steam pipe, resulting in a break size limited only by the size of the integral flow restrictors, assumed to be 1.4 ft^2. (This is identical to the current analysis method case.)

Reactor Protection: The accident resulted in a SI signal on low steamline pressure. This initiates steamline isolation, feed line isolation, safety injection start, and auxiliary feedwater. (The low steamline pressure SI setpoint, safeguards features actuation times, delay times and uncertainties assumed are identical to the current analysis method.)

d) **DNB Evaluation**

The VIPRE code is used in a separate time-dependent calculation to determine the minimum DNBR, based on the core average power, power distribution, inlet temperature, core inlet flow, and core exit pressure vs. time. The core average power and power distribution are obtained from SPNOVA, including the time-dependent changes in radial enthalpy rise hot channel factor ($F_{\Delta H}$) and the axial power distribution. The current methodology pin-by-pin design power distribution (Reference 5) is used as the initial value for the DNBR calculations. The same standard uncertainty allowances are applied to the calculated $F_{\Delta H}$ as for current licensing applications (see Section 2.6). The reactor coolant conditions (inlet temperature, core inlet flow and core exit pressure vs. time) are obtained from RETRAN. The results are presented in Section 3.5.6 below.

3.5.6 Results and Comparison with Current Method

The main steamline break event was analyzed using both the current analysis method and the updated 3-D core transient analysis method assuming a 1.77% Δk/k SDM. [

]^{a,c}. The minimum DNBR obtained with both methods is shown in Table 3.5-1. Table 3.5-1 also shows the results of the same calculation performed with the updated 3-D core transient method, but assuming a minimum required shutdown margin of 1.0% Δk/k. Table 3.5-2 shows the sequence of events for the 1.77% Δk/k SDM case. The transient results of the updated 3-D analysis method for the 1.77% Δk/k SDM case are compared to the current analysis method in Figures 3.5-1 to 3.5-5. The results for the 1.0% Δk/k SDM case are presented in Figures 3.5-8 and 3.5-9. A case was also performed assuming a loss of offsite power (LOOP) at the time of initiation of the event, resulting in a coastdown of the reactor coolant pumps. These results are shown in Figure 3.5-10.

<u>Results of the 1.77% Δk/k SDM Case</u>:

[

]^{a,c}.

[

]^{a,c}.

[

]a,c.

[

]a,c.

[

]a,c.

[

]a,c.

Figure 3.5-7 shows the DNBR vs. time for the 3-D case. The minimum DNBR value reached is shown in Table 3.5-1.

Results of the 1.0% Δk/k SDM Case:

Figures 3.5-8 and 3.5-9 show the heat flux vs. time and DNBR vs. time transients for the same steam break event, with the only change being a reduction in shutdown margin from 1.77% Δk/k to 1.0% Δk/k. [

]a,c.

Results of the Loss of Offsite Power Case:

A sensitivity case was performed assuming a loss of offsite power 3 seconds after the time of the reactor trip and break initiation. The loss of offsite power causes a loss of power to the RCPs, resulting in a flow coastdown. The RCS flow during the event would then depend on natural circulation to remove the core heat. The reduced flow, however, greatly reduces the ability of the secondary side to extract heat from the RCS, resulting in a much slower RCS cooldown and a reduced core return to power level. [

]a,c.

[

]a,c.

3.5.7 Summary

The updated 3-D core transient methodology was used to analyze the main steamline break event from hot zero power conditions with a SDM of 1.77% Δk/k. [

]a,c.

[

]a,c.

Finally, the results of a "low flow" steamline break case show a very large margin to DNB, demonstrating that the "full flow" cases presented above are more limiting. It should be noted that the low-flow case has also been explicitly modeled for a CE-designed analog protection system plant, and the results were consistent with the Westinghouse-designed plants in that the case with offsite power available (the case with full reactor coolant flow) was found to be the limiting case.

3.5.8 Conclusions

A sensitivity study was performed for the updated 3-D core transient method, which addresses the effect of variations in the initial conditions and assumptions used in the analysis. The study showed that the most limiting nuclear and thermal power transient is associated with [

]a,c.

3.5.9 Reload Safety Evaluation

For a reload core using a safety evaluation performed with the updated 3-D core transient methodology, the continued applicability of the safety evaluation will be confirmed as part of the Reload Safety Evaluation (RSE) process described in Section 2.8. For the Main Steamline Break at Hot Zero Power event, the core neutronics parameters assumed in the analysis that may vary from cycle-to-cycle as a result of a reload are:

- Moderator feedback coefficient*
- Doppler feedback coefficient*
- Delayed neutron fraction
- Radial power peaking factor ($F_{\Delta H}$) with N-1 (tripped)*
- Shutdown margin*

* **Key parameters – see below.**

Based on the sensitivity study presented in Section C.5 of Appendix C, the transient is not sensitive to significant variations in the [

]^{a,c}.

These key parameters are not expected to change significantly from cycle-to-cycle unless there is a significant change in the fuel loading pattern. In the reload design process, cycle-specific static calculations are performed to determine if the calculated values of the parameters remain bounded by the value used in the licensed safety analysis. The transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. This methodology is described in more detail in Reference 15.

For this event, a cycle-specific calculation will continue to be performed to confirm that the core acceptance criteria are met for the fuel reload.

Table 3.5-1
Main Steamline Break at Hot Zero Power Analysis Results

Analysis Method	Minimum DNBR	Time of Min. DNBR (sec.)*
Current Point-Kinetic Methodology (1.77% Δk/k SDM)	[ ] a,c	[ ] a,c
Updated 3-D Core Transient Method (1.77% Δk/k SDM)	[ ] a,c	[ ] a,c
Updated 3-D Core Transient Method (1.0% Δk/k SDM)	[ ] a,c	[ ] a,c

* From initiation of steamline break (t = 0.)

Table 3.5-2
Main Steamline Break at Hot Zero Power Sequence of Events
(Updated 3-D Core Transient Method with 1.77% Δk SDM)

Event	Time (seconds)
Steamline Ruptures	[ ] a,c
Low Steamline Pressure Setpoint Reached in Two Loops	[ ] a,c
Feed Line Isolation Occurs	[ ] a,c
Steamline Isolation Occurs	[ ] a,c
SI Injection Begins	[ ] a,c
Borated Water from RWST Reaches the Core	[ ] a,c
Accumulators Actuate	[ ] a,c
Minimum DNBR Occurs	[ ] a,c

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Figure 3.5-1
Main Steamline Break at HZP
Nuclear Power vs. Time
Current Method vs. Updated 3-D Core Transient Method
1.77% SDM

a, c

[ ]

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Figure 3.5-2
Main Steamline Break at HZP
Core Average heat Flux vs. Time
Current Method vs. Updated 3-D Core Transient Method
1.77% SDM

a, c

[ ]

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Figure 3.5-3
Main Steamline Break at HZP
Cold Leg Temperature vs. Time
Current Method vs. Updated 3-D Core Transient Method
1.77% SDM

a, c

[ ]

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Figure 3.5-4
Main Steamline Break at HZP
Pressurizer Pressure vs. Time
Current Method vs. Updated 3-D Core Transient Method
1.77% SDM

a, c

[ ]

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Figure 3.5-5
Main Steamline Break at HZP
Core Boron Concentration vs. Time
Current Method vs. Updated 3-D Core Transient Method
1.77% SDM

a, c

[ ]

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Figure 3.5-6
Main Steamline Break at HZP
$F_{\Delta H}$ and Axial Offset vs. Time
Updated 3-D Core Transient Method Case
1.77% SDM

a, c

[ ]

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Figure 3.5-7
Main Steamline Break at HZP
Minimum DNBR vs. Time
Updated 3-D Core Transient Method Case
1.77% SDM

a, c

[ ]

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Figure 3.5-8
Main Steamline Break at HZP
Core Average Heat Flux vs. Time
Current Method vs. Updated 3-D Core Transient Method Case
1.0% SDM

a, c

[ ]

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Figure 3.5-9
Main Steamline Break at HZP
Minimum DNBR vs. Time
Updated 3-D Core Transient Method Case
1.0% SDM

a, c

[ ]

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Figure 3.5-10
Main Steamline Break at HZP
Core Average Heat Flux vs. Time
Updated 3-D Core Transient Method-LOOP Case
1.77% SDM

a, c

[ ]

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3.6 Application to Other Non-LOCA Accidents

The sample calculations presented in this report were performed for a typical Westinghouse-designed 3-loop plant. The application of the computer codes and the basic methodology for the updated 3-D core transient model is applicable to any plant design for which the backbone codes (SPNOVA, VIPRE and RETRAN) are appropriate. This topical report demonstrates that the codes can be applied in a very similar fashion as the current point-neutron kinetics method. The methods and models used are consistently applied, independent of plant design. The methodology uses only NRC-approved computer codes, and the report shows that the use of an external communication interface does not disrupt the function of the individual codes.

The application of the RETRAN code to Westinghouse reactors, including the nodalization for the various system models, was presented to the NRC in the Westinghouse report WCAP-14882-P-A (Reference 26). This report was reviewed by the NRC, and approved for application to all 2-, 3- and 4-loop Westinghouse plants. The application of RETRAN to the CE analog plants is currently under review as part of the St. Lucie Unit 2 30% steam generator tube plugging (SGTP) program. The Westinghouse RETRAN model includes approval for the use of only the point neutron kinetics model in the core neutronics calculations. The updated 3-dimensional core analysis methodology addressed in this report uses the same models as approved in Reference 26, except that the point kinetics and fuel rod heat transfer models are not used. Instead, the core neutron kinetic and thermal kinetic behavior is calculated externally using the SPNOVA and VIPRE codes (see Section 2.2), and the calculated heat flux is automatically transferred to the RETRAN core model using the existing RETRAN non-conducting heat exchanger model. No new models were developed for the RETRAN calculation. The RCS primary and secondary nodalization is unchanged, except for the addition of [

]^{a,c}.

Therefore, so long as the individual computer codes and models are appropriate for use at a given plant, the use of those same computer codes using an external communication interface to transfer data is also appropriate using the same models. This includes both Westinghouse and non-Westinghouse designed plants.

The list of accident events which are analyzed as part of a typical plant licensing basis is shown in Table 3.6-1. Also shown in this Table (designated by check marks) are the events which are analyzed using the RETRAN computer code, for which specific approval was obtained in the NRC review of the code application topical, WCAP-14882-P-A (Reference 26). The topical report provided here for the updated 3-D core transient methodology shows sample calculations for a representative subset of the above non-LOCA events: loss of forced reactor flow, locked rotor (DNB), locked rotor (peak pressure), hot zero power steamline break (HZP SLB) and hot full power steamline break (HFP SLB). The use of an external communication interface to link the 3-D core calculations with the RCS loop model was mainly implemented to develop a better understanding of the existing margins in the DNB limiting events. Therefore, the representative events presented in this topical report were selected based on their severity with respect to the DNB licensing basis. However, the methodology presented in this topical

report would be applicable to all of the events currently analyzed with RETRAN as listed in Table 3.6-1. The generic applicability of this methodology to the various events is discussed in the following paragraphs. The five demonstration transients presented herein utilize all of the functionalities required for the remainder of the non-LOCA events.

Category 1: Increase in Heat Removal by Secondary System

With respect to the applicability of this methodology to the Increase in Heat Removal by Secondary System events, both the hot full power and hot zero power steamline break analyses are explicitly demonstrated in this topical report. The steamline break events are the most severe events with respect to an increase in heat removal by the secondary system. The HFP SLB analysis was performed both with and without rod motion to confirm that rod motion was accurately implemented in the data transfer. The remaining events in this category (feedwater system malfunctions, excessive increase in steam flow and inadvertent opening of a steam generator relief or safety valve) are similar, but less severe, than the steamline break events. The Westinghouse RETRAN model was approved for applicability to all of these events. Therefore, the methodology set forth herein would also be applicable to the remaining events in this category, if better understanding of the margins for those events was required for a given plant.

Category 2: Decrease in Heat Removal by Secondary System

With respect to the applicability of this methodology to the Decrease in Heat Removal by the Secondary System events, the Loss of Load/Turbine Trip, Loss of Offsite Power, Loss of Normal Feedwater and Feedline Rupture events are all included in the list of transients officially approved for analysis with the Westinghouse RETRAN model. The Loss of Offsite Power, Loss of Normal Feedwater and Feedline Rupture events are analyzed with respect to long-term core coolability, which is primarily influenced by the decay heat and not by the 3-D core kinetics implemented in this methodology. The DNB behavior of the Loss of Offsite Power event is already covered by the Complete Loss of Forced Reactor Flow event. The loss of external load/turbine trip event is analyzed for both DNB and peak pressure behaviors. [

] a,c.

Category 3: Decrease in Reactor Coolant Flow Rate

With respect to the applicability of this methodology to the Decrease in Reactor Coolant Flow Rate events, the loss of flow and locked rotor events are explicitly demonstrated in this topical report. The locked rotor sensitivities consider both the DNB and peak pressure aspects of the event. The Westinghouse RETRAN model was approved for the analysis of all of these events.

Category 4: Reactivity and Power Distribution Anomalies

With respect to the reactivity and power distribution anomalies events, the use of the SPNOVA/VIPRE 3-D core model for the analysis of the Spectrum of RCCA Ejection Accidents has already been reviewed and approved by the NRC (Reference 7). The Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition is a prompt-critical event, very similar to (but less limiting than) the zero power RCCA ejection event for which approval was received in Reference 7. In both of these events, the core transient is not sensitive to the RCS loop model since the inlet conditions do not change over the time of interest in these very fast events. The linking of the RETRAN code with SPNOVA/VIPRE does not affect the ability of SPNOVA/VIPRE to model these events. Therefore, although the point-kinetics model in RETRAN was not approved for use in these events, the updated 3-D core transient methodology, in which the point kinetics model is replaced by the 3-D core model, is applicable to these transients.

The application of the Westinghouse RETRAN model to the Uncontrolled RCCA Bank Withdrawal at Power and RCCA Misoperation events was approved in Reference 26. For the updated 3-D core transient methodology, the rod control system model used in the approved Westinghouse RETRAN model was not modified; however, the rod speed (and direction) signals are transferred to the SPNOVA code to move the control rods instead of to the RETRAN point kinetics model. The accurate implementation of rod motion in the data transfer was demonstrated in the at-power steamline break sensitivities performed explicitly for this report (see Sections 3.4.6 and C.4.3). Therefore, implementation of the 3-D core methodology is appropriate for both the Uncontrolled RCCA Bank Withdrawal at Power and RCCA Misoperation events.

Category 5: Increase in Reactor Coolant Inventory

With respect to the applicability of this methodology to the Increase in Reactor Coolant Inventory events, both the Inadvertent ECCS Actuation at Power and CVCS Malfunction Causing an Increase in Reactor Coolant Inventory events are included in the list of transients officially approved for analysis with the Westinghouse RETRAN model. [

]^{a,c}.

Category 6: Decrease in Reactor Coolant Inventory

With respect to the applicability of this methodology to the Decrease in Reactor Coolant Inventory events, both the Inadvertent Opening of a Pressurizer Safety or Relief Valve and Steam Generator Tube Failure events are included in the list of transients officially approved for analysis with the Westinghouse RETRAN model. The Inadvertent Opening of a Pressurizer Safety or Relief Valve event is typically a non-limiting event. [

]a,c.

Table 3.6-1
US NRC Reg. Guide-1.70 Classification of Events
✓ = Non-LOCA Events Approved for Analysis Using RETRAN Model
(WCAP-14882-P-A, Reference 26)

1.	**Increase in Heat Removal by Secondary System**
a.	✓ Feedwater Malfunctions Causing a Decrease in Feedwater Temperature
b.	✓ Feedwater Malfunction Causing an Increase in Feedwater Flow
c.	✓ Excessive Increase in Secondary Steam Flow
d.	✓ Inadvertent Opening of a SG Safety or Relief Valve
e.	✓ **Steam System Piping Failure**
2.	**Decrease in Heat Removal by Secondary System**
a.	✓ Loss of Electrical Load and/or Turbine Trip
b.	✓ Loss of Non-Emergency AC Power
c.	✓ Loss of Normal Feedwater
d.	✓ Feedwater System Pipe Break
3.	**Decrease in Reactor Coolant Flow Rate**
a.	✓ Partial Loss of Forced Reactor Coolant Flow
b.	✓ **Complete Loss of Forced Reactor Coolant Flow**
c.	✓ **RCP Shaft Seizure (with & w/o Loss of AC Power)**
d.	✓ **RCP Shaft Break**
4.	**Reactivity and Power Distribution Anomalies**
a.	Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
b.	✓ Uncontrolled RCCA Bank Withdrawal at Power
c.	✓ RCCA Misoperation (Rod Drop Accident)
d.	Startup of an Inactive Reactor Coolant Loop
e.	Uncontrolled Boron Dilution
f.	Inadvertent Loading of a Fuel Assembly in an Improper Location
g.	**Spectrum of RCCA Ejection Accidents ***
	***Approved for analysis using SPNOVA/VIPRE**
5.	**Increase in Reactor Coolant Inventory**
a.	✓ Inadvertent ECCS Actuation at Power
b.	✓ CVCS Malfunction Causing an Increase in Reactor Coolant Inventory
6.	**Decrease in Reactor Coolant Inventory**
a.	✓ Inadvertent Opening of a Pressurizer Safety or Relief Valve
b.	✓ Steam Generator Tube Failure
c.	Loss of Coolant Accident

4.0 SUMMARY AND CONCLUSIONS

This report describes the Westinghouse updated methodology for the analysis of non-LOCA reactor system transients in pressurized water reactor cores using 3-D neutron kinetics in a more realistic and consistent, but still conservative, manner. The methodology utilizes the NRC-approved codes SPNOVA (References 3 & 4), VIPRE-01 (References 5 & 6) and RETRAN-02 (Reference 26), which have been linked through an external communication interface to pass the necessary data for the nuclear, core fluid and fuel temperature, and reactor coolant system calculations. The solution methods are the same as those previously approved for each code. No new calculational models were developed for these codes. The external communication interface between the SPNOVA and VIPRE codes for use in the Westinghouse 3-D control rod ejection accident analysis methodology has already received NRC approval (Reference 7).

The hot rod analysis is performed separately in VIPRE using the core transient power as the forcing function with the actual 3-D peaking factors, including uncertainty allowances, as a function of time. The thermal models for the hot rod analyses are the same as the current licensed models, and provide a conservative analysis for the parameter of interest (maximum fuel temperature or minimum DNBR).

To demonstrate the application of the methodology, several accidents were selected from the list of event categories in NRC Regulatory Guide 1.70 (Reference 1) and listed in Table 1.0-1. The accidents selected represent limiting events with respect to DNB or overpressure in each category. In the analysis of each of the accident events presented, conservative preconditions were chosen, including time in cycle, the effect of xenon distributions on axial power shape, and allowable control rod positions. The difference from the current methodology is that the linked core neutronics/thermal hydraulics calculation allows using parameters consistent with the time in cycle for which the accident is investigated, and allows taking into account the variations in these parameters as the transient progresses, instead of using unrealistic constant bounding values or a mixture of values from different times in the cycle. Uncertainty allowances, as discussed in the report, are applied on the key parameters to ensure a conservative analysis result. In general, these are the same uncertainties applied to the same parameters as for the current methodology.

A sensitivity study (Appendix C) was performed for each event to ensure that the analysis parameters and uncertainties chosen for the analysis are conservative. As a result of the sensitivity study, a reference bounding analysis case (or cases) was defined for each event. This defines the methodology proposed for each plant for which the method is applied.

For each event, key parameters that were used in the analysis and may vary as a result of a reload cycle design were identified. These key parameters are not expected to change significantly from cycle-to-cycle unless there is a significant change in the fuel loading pattern. In general, these are the same parameters which are evaluated as part of the reload design process in the current methodology. For a reload core using a safety evaluation performed with the updated 3-D transient neutronics methodology, the continued applicability of the safety evaluation will be confirmed as part of the Reload Safety Evaluation (RSE) process described in Section 2.8. As part of this process, cycle-specific static calculations are performed to determine if the calculated values of the parameters remain bounded by the

value used in the licensed safety analysis. The transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameters. This methodology is described in more detail in Reference 15.

The 3-D kinetics methodology described herein is not limited to a specific plant type. The same computer codes employed here have been used in licensing applications for many Westinghouse-designed 2-, 3- and 4-loop plants with various fuel designs, and by Westinghouse for a CE-designed analog protection system plant. The computer codes and method of data transfer between the codes (the external communication interface) are applicable to any PWR for which a licensed model is available for the base codes (ANC/SPNOVA, VIPRE and RETRAN).

The functionality of base codes have not been affected by the replacement of the point-kinetics reactor core model in RETRAN with the SPNOVA/VIPRE 3-D core kinetics model. This is evidenced by the comparisons with the current point-kinetics method, which exhibited only the expected differences due to the application of 3-D kinetics methods. The continued functionality of the codes, and the validity of the data transfer between the codes, is further evidenced by the excellent agreement with the results of the OECD Main Steamline Break benchmark problem shown in Appendix B.

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APPENDIX A
OVERVIEW OF COMPUTER CODES

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APPENDIX A OVERVIEW OF COMPUTER CODES

A.1 Introduction

The analysis of reactor system transients using a 3-D representation of the reactor core requires that the nuclear calculations, the core thermal-hydraulic and fuel temperature calculations, and the RCS calculations to be performed in a linked manner in both the steady state mode (for initialization) and the transient mode. The 3-D methodology utilizes computer programs previously reviewed and approved by the NRC. The codes are: the SPNOVA computer program for the neutron kinetics, the VIPRE computer program for the core thermal hydraulics and fuel temperature calculation, and the RETRAN code for the reactor coolant system response calculation. In addition, the VIPRE code is used in a separate calculation for the hot rod DNBR and peak fuel/clad temperature transient evaluation. These codes are described in more detail below. The data transfer between the codes has been automated to prevent errors that could occur with hand manipulation of data. All programming changes have been limited to those needed to facilitate the data transfer and interface; no changes or additions have been made to the NRC-approved models in the codes as a result of the updated 3-D core transient methodology.

The use of the 3-D SPNOVA and VIPRE codes, and the method of data transfer, was reviewed and approved by the NRC for a severe rod ejection transient event in WCAP-15806-P-A. The additional data transfer between SPNOVA/VIPRE and RETRAN is described in Section A.5.

A.2 SPNOVA

A.2.1 Nodal Solution

The Westinghouse standard core design methodology uses a 3-D nodal expansion method for the static analysis of the cores. This methodology is licensed and has been incorporated into the NRC-approved SPNOVA computer program (References 3 & 4). The static neutronics solution in SPNOVA is also consistent with the NRC-approved ANC computer program (References 9, 10, 11, 12).

A.2.2 Neutron Kinetics

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

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

WCAP-12983 SER Limitation:

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

Compliance for Reactor Coolant System Transient Analysis:

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

A.3 VIPRE-01

VIPRE-01 is a subchannel code developed from several versions of the COBRA code by the Battelle Northwest National Laboratories under the sponsorship of Electric Power Research Institute (EPRI). The subchannel analysis concept used in VIPRE is the same as in COBRA-IIIC. Conservation equations of mass, axial and lateral momentum and energy are solved for the fluid enthalpy, axial flow rate, lateral flow and momentum pressure drop. A detailed description of the VIPRE code can be found in Reference 6.

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

The Westinghouse version of VIPRE-01 (Reference 5) contains additional features as compared to the original VIPRE-01, including Westinghouse DNB correlations and heat transfer correlations consistent with the FACTRAN code (Reference 21). For the hot fuel rod transient calculations, the following FACTRAN features have been incorporated into VIPRE-01: a) the Bishop-Sandberg-Tong heat transfer correlation for film boiling (Reference 19), b) Baker-Just model for calculating heat generation in the cladding due to zirconium-water reaction (Reference 20), and c) fuel enthalpy and melting predictions. However, the code additions do not alter the fundamental VIPRE-01 computational methods and functional capabilities. The modified version of VIPRE-01 is maintained in accordance with Westinghouse Quality Assurance (QA) procedures for software control.

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

WCAP-14565 SER Condition 1:

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

Compliance for Reactor Coolant System Transient Analysis:

DNBR calculations for radiological consequence evaluation are performed with the NRC-approved VIPRE modeling assumptions described in Reference 5. Selection of a DNB correlation, DNBR limit and hot channel factors will be justified on a plant specific basis depending on fuel type. For fuel temperature evaluations, as described in Chapter 2 of this report, the hot fuel rod transient calculation is consistent with that for the post-CHF locked rotor analysis in Reference 5 and with the FACTRAN model described in Reference 21.

WCAP-14565 SER Condition 2:

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

Compliance for Reactor Coolant System Transient Analysis:

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

WCAP-14565 SER Condition 3:

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

Compliance for Reactor Coolant System Transient Analysis:

Only NRC-approved DNB correlations will be used for the RCS transient analysis DNBR calculations.

WCAP-14565 SER Condition 4:

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

Compliance for Reactor Coolant System Transient Analysis:

The VIPRE hot rod modeling retains the same conservatism as the current design methods using FACTRAN. Specifically, the following conservative assumptions are made in the VIPRE calculation, in order to maximize the increase in fuel and clad temperature:

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

A.4 RETRAN-02

RETRAN-02 is a flexible transient thermal-hydraulic code which is used to compute the thermal-hydraulic behavior of a light-water reactor system to normal operational transients and accident conditions. The RETRAN computer code includes a point-kinetics model to model the neutronics behavior of the core, and a core thermal-hydraulics and fuel rod model to calculate the local fluid conditions and fuel temperature for the moderator and Doppler feedback. The RETRAN code models the RCS components including the reactor vessel, a two-region non-equilibrium pressurizer, steam generator models, reactor coolant pumps, and the action of relief and safety valves. In addition, the RETRAN code models the reactor control and protection system, the Engineered Safety Feature Actuation System (ESFAS), safety injection (SI), charging and letdown flow, and secondary side models including main/auxiliary feed flow and steamline and feed line valve actuations. The licensed core inlet temperature mixing models include zero (no) mixing, perfect (uniform) mixing or a "design" partial mixing model for a 2-, 3- or 4-loop plant.

In the application described here, the point neutron kinetics and fuel heat transfer models are not used; instead, the RETRAN non-conducting heat exchanger model is used in the core with the core heat flux for each spatial node supplied from VIPRE. The only change from the RCS nodalization model described in the NRC-approved WCAP-14882-P-A (Reference 26) is the use of six axial core volume nodes per core sector instead of three nodes per core sector. (There is one core sector per RCS cold leg loop.) This was done to facilitate the data transfer from the VIPRE model to the RETRAN core model. No other changes or additions were made to the reviewed and approved Westinghouse plant nodalization model. All programming changes have been limited to those needed to facilitate the data transfer; no programming changes or additions have been made to the NRC-approved calculational models in the RETRAN code.

The NRC SER on WCAP-14882-P-A (included in Reference 26) concludes that the Westinghouse RETRAN code applications are acceptable and that RETRAN can be used to replace the LOFTRAN code in non-LOCA safety analysis. The staff generic SER for the RETRAN-02 code lists certain limitations on the use of the code and items for additional justification (included as Reference 3 in Appendix A of WCAP-14882-P-A). These were addressed in a letter to the NRC (NSD-NRC-98-5809 dated November 12, 1998, copy included in the approved version of WCAP-14882-P). These limitations and justifications were revisited considering the use of the Westinghouse RETRAN model for accident events

linked to an external 3-D core neutronics model. No exceptions to the Westinghouse responses to the non-core kinetics model limitations or justifications discussed in the reference were found. Any changes with respect to the core kinetics model are addressed in this report.

A.5 Automated Data Transfer Method

The effective 3-D analysis of reactor coolant system transient events requires nuclear calculations, thermal-hydraulic and fuel temperature calculations, and reactor coolant systems calculations to be performed in a linked manner for the entire core in both a steady-state condition and the transient mode. The methodology uses the NRC-approved programs with a distributed architecture. The architecture uses a standard external communication interface protocol for communication between running programs on the same or different computers to transfer data. Currently the programs utilize the Parallel Virtual Machine (Reference 25) software for the data transfer, but this interface could be replaced with another product with no change in computational results. Thus, the actual mechanism used for the data transfer is not an inherent part of the methodology.

The only modification needed by the programs was the ability to transfer selected data into and out of the executing program. To further simplify, the data communication between the major programs is not direct; an intermediate auxiliary program (ANCKVIPRE) is utilized to coordinate the data transfer between SPNOVA and VIPRE, and another auxiliary program (RAVE) is utilized to coordinate data transfer between ANCKVIPRE and RETRAN. A schematic of the data flow is presented in Figures A.5-1 and A.5-2. In addition to the data transfer, the auxiliary program also saves the hot rod information for later processing. This information is used to generate the forcing functions for the hot rod analysis.

One subject that had to be addressed in the automated data transfer process is the translation of the two-, three- or four-channel conditions of the RETRAN model (depending upon the number of cold loops in the model) to the flow conditions applied to the inlet for each of the SPNOVA/VIPRE model core channels. Four different methods of translation are provided by the data transfer interface:

1) Average model: This simulates perfect mixing in the reactor vessel lower plenum and therefore provides uniform conditions across the core.

2) Core sector model: This combines the mixing characteristics implemented in the RETRAN model (See Reference 26), which can be varied from a near-perfect mixing to a near-zero mixing behavior, with a user-defined map to define the contribution of each RETRAN flow channel to each VIPRE model channel.

3) Currently licensed model: This implements a predefined distribution based upon scaled model tests performed for a Westinghouse reactor vessel. This distribution model is applied for HZP SLB analyses documented in WCAP-9226-P-A (Reference 14). This is essentially a specific case of the "core sector" model.

4) Fine mesh model: This provides a mathematically distributed temperature across the inlet of the core based upon the number of cold legs and user-defined characteristics which control the level of mixing.

It should be noted that the linkage of the SPNOVA and VIPRE codes by means of the external communication interface described here, was reviewed and approved by the NRC for application to the Westinghouse 3-D Control Rod Ejection methodology in WCAP-15806-P-A (Reference 7).

Figure A.5-1
Computer Program Data Transfer Schematic Diagram
Core Nuclear/Thermal-Hydraulic Data Transfer

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Figure A.5-2
Computer Program Data Transfer Schematic Diagram
Core and RCS Loop Data Transfer

APPENDIX B
OECD MAIN STEAMLINE BREAK (MSLB) BENCHMARK

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APPENDIX B OECD MAIN STEAMLINE BREAK (MSLB) BENCHMARK

B.1 The PWR Main Steamline Break Benchmark Problem

The Nuclear Science Committee (NSC) of the Nuclear Energy Agency (NEA)/Organization for Economic Co-Operation and Development (OECD) has released a set of computational benchmark problems for the assessment of computer codes used in nuclear plants safety analysis. Recently, in a cooperative program sponsored by the OECD, the United States Nuclear Regulatory Commission (US NRC), and the Pennsylvania State University (PSU), a PWR Main Steamline Break (MSLB) benchmark problem has been defined in order to simulate the core response and the reactor coolant system response to a relatively severe steamline break accident condition. A PWR Main Steamline Break (MSLB), which may occur as a consequence of the rupture of one steamline upstream of the main steam isolation valves, is characterized by significant space-time effects in the core caused by asymmetric cooling and an assumed stuck-out control rod during reactor trip. Simulation of the transient requires evaluation of the core response from a multi-dimensional 3-D neutronics/core thermal-hydraulics perspective supplemented by a 1-D simulation of the remainder of the reactor coolant system. This problem was therefore considered appropriate to test the incorporation of a full three-dimensional (3-D) modeling of the reactor core into a system transient code to allow simulations of interactions between reactor core behavior and plant dynamics.

This benchmark was structured into three separate phases, and the specifications required to perform the three exercises were provided in References B-1 through B-4.

- ***Phase I:*** ***Plant transient simulation with point kinetics***

  The purpose of this exercise was to test the primary and secondary system model responses. Point kinetics model inputs were provided to simulate the axial and radial power distribution and tripped rod reactivity from Exercise 3. The results of this phase of the benchmark were documented in Reference B-2.

- ***Phase II:*** ***Transient simulation with 3-D neutronics/core thermal-hydraulics***

  The purpose of this exercise was to model the core region only. The core transient boundary conditions were provided. The results of this phase of the benchmark were documented in Reference B-3.

- ***Phase III:*** ***Plant transient simulation with 3-D core neutronics***

  This exercise combined elements of the first two exercises in this benchmark and provided an analysis of the transient in its entirety. The results of this phase of the benchmark were documented in Reference B-4. Note that two different scenarios were part of this exercise, differing only for the shutdown margin available and thus for the magnitude of the predicted return to power during the transient. Only the second scenario, with a lower shutdown margin so to enhance the amount of return to power in the core during the transient, has been performed for this phase.

The three benchmark exercises were performed to provide additional validation of the external communication interface.

B.2 Phase I: Plant Transient Simulation with Point Kinetics

The RETRAN-02 code was used to perform this exercise. The data required for the preparation of the input deck for this phase were obtained from the final benchmark specifications provided in Reference B-1. The plant model was set up following the benchmark specifications as closely as possible, and this exercise was performed mostly to verify the implementation of the plant model, and to support the sensitivity evaluations provided during the Phase III investigation. Excellent agreement between the RETRAN results was verified, from both a qualitative and quantitative point of view. Results for this phase are not discussed in detail in this appendix, since most of the conclusions are common with the Phase III results.

B.3 Phase II: Transient Simulation with 3-D Neutronics/Core Thermal-Hydraulics

The core thermal-hydraulic (T&H) and neutronics models of SPNOVA/VIPRE in Chapter 2 of this report were modified and adjusted according to the benchmark specifications. The core layout is shown in Figure B.3-1.

One channel per fuel assembly was used in the VIPRE model of the core giving a total of 177 coolant channels. For the active fuel length, an axial mesh with twenty four (24) nodes with the node lengths specified by the Exercise 2 of the benchmark (i.e., one-to-one mapping between thermal-hydraulic and neutronics meshes) was used. In addition, two unheated nodes were used, one at the bottom and one at the top of the fuel assembly to model the axial reflectors. Two T&H models were considered: Open (crossflow between channels) and Closed (no crossflow between channels) fuel channels. The results are presented for both fuel channel models. The VIPRE model considers two different fuel rod types depending on the burnup value: a) Region 1: burnup from 32,000 to 58,000 MWD/MTU and b) Region 2: burnup from 23,000 to 31,000 MWD/MTU. The radial reflector assemblies were not considered in the thermal-hydraulic model as no flow is allowed in that region. Finally, the direct energy deposition in the coolant was assumed equal to 2.6%.

The SPNOVA neutronics model used one radial node per assembly. The radial and axial reflector assemblies were explicitly modeled. Coolant density was set to the inlet density at the bottom and the radial reflectors, and to the outlet density at the upper reflector. Cross-sections for the fuel and the coolant were calculated by interpolation of the supplied cross-section libraries with no extrapolation beyond the defined boundaries. Spatial 3-D decay heat distribution was used.

Steady state and transient simulations were performed for Phase II of the PWR MSLB benchmark. The steady state simulations were intended to determine the control rod and stuck rod worths. Good agreement was observed between the SPNOVA/VIPRE predictions and the benchmark average results, with all relevant parameters predicted within a single standard deviation of the average of the benchmark participants.

Figure B.3-1
OECD MSLB Benchmark Problem Core Description

		1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*	*13*	*14*	*15*	
1							14	14	14	14	14						
2					15	14	14	14	14	14	14	14	13				
3				15	15	9	8	8	8	8	8	7	13	13			
4			15	15	15	9	8	8	8	8	8	7	13	13	13		
5			15	9	9	9	3	2	2	2	1	7	7	7	13		
6		15	15	9	9	3	3	2	2	2	1	1	7	7	13	13	
7		15	15	9	9	3	3	3	2	1	1	1	7	7	13	13	
8		16	15	9	10	4	3	4	2	6	1	6	12	7	13	18	
9		16	16	10	10	4	4	4	5	6	6	6	12	12	18	18	
10		16	16	10	10	4	4	5	5	5	6	6	12	12	18	18	
11			16	10	10	10	4	5	5	5	6	12	12	12	18		
12			16	16	16	10	11	11	11	11	11	12	18	18	18		
13				16	16	10	11	11	11	11	11	12	18	18			
14					16	17	17	17	17	17	17	17	18				
15							17	17	17	17	17						

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Two different scenarios were considered for the transient simulations, differing only in the total rod worth. The first scenario was defined using a realistic rod worth and it was expected that no return to power during the cooldown part of the transient would be observed. In the second scenario a lower rod worth was used, and a return to power during the cooldown was expected. The discussions provided herein are limited to the results of the more challenging "return to power" scenario. The simulations were performed using the inlet temperatures and flow rates for eighteen different core regions (depicted on Figure B.3-1) and an average core exit pressure. These boundary conditions are provided in the MSLB benchmark specifications.

The results obtained for the Return to Power Scenario at the highest power before and after the scram are summarized in Table B.3-1. As can be see in these tables, both SPNOVA/VIPRE open and closed channel models are in good agreement with the benchmark results, but the closed channel model gives better agreement since most of the participants in the benchmark used closed channels in the T&H core model. The total power after scram is slightly higher than the mean value of the benchmark (see also Figure B.3-2). This difference is mainly caused by differences in the axial nodalization. In particular, for the active fuel length, the VIPRE model used 24 nodes having different lengths (one-to-one mapping between thermal-hydraulic and neutronics meshes) while most of the benchmark participants employed 24 nodes of equal length or more refined meshes.

Table B.3-1
Total Core Power for OECD MSLB Benchmark - Phase II

	Calculated Value (and deviation from mean)		Average of Benchmark Participants (and Standard Deviation)
	Open Channel	Closed Channel	
Maximum Core Power Before Reactor Trip			
Total Core Power (MWt)	3234.35 (Δ = -37.53)	3244.85 (Δ = -27.03)	3271.88 (σ = 36.50)
Time (seconds)	6.80 (Δ = -0.15)	7.02 (Δ = +0.07)	6.95 (σ = 0.17)
Maximum Core Power After Reactor Trip			
Total Core Power (MWt)	951.59 (Δ = -84.68)	924.16 (Δ = -57.25)	866.91 (σ = 54.13)
Time (seconds)	57.40 (Δ = -0.98)	57.50 (Δ = -0.88)	58.38 (σ = 1.81)

The comparisons of the time histories for the total power, coolant density and the core-average and maximum Doppler temperatures for the open and closed channel SPNOVA/VIPRE models and the MSLB Benchmark results are presented in Figures B.3-2 to B.3-5.

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Figure B.3-2
MSLB Benchmark Phase II Scenario 2: Core-average Total Power vs. Time

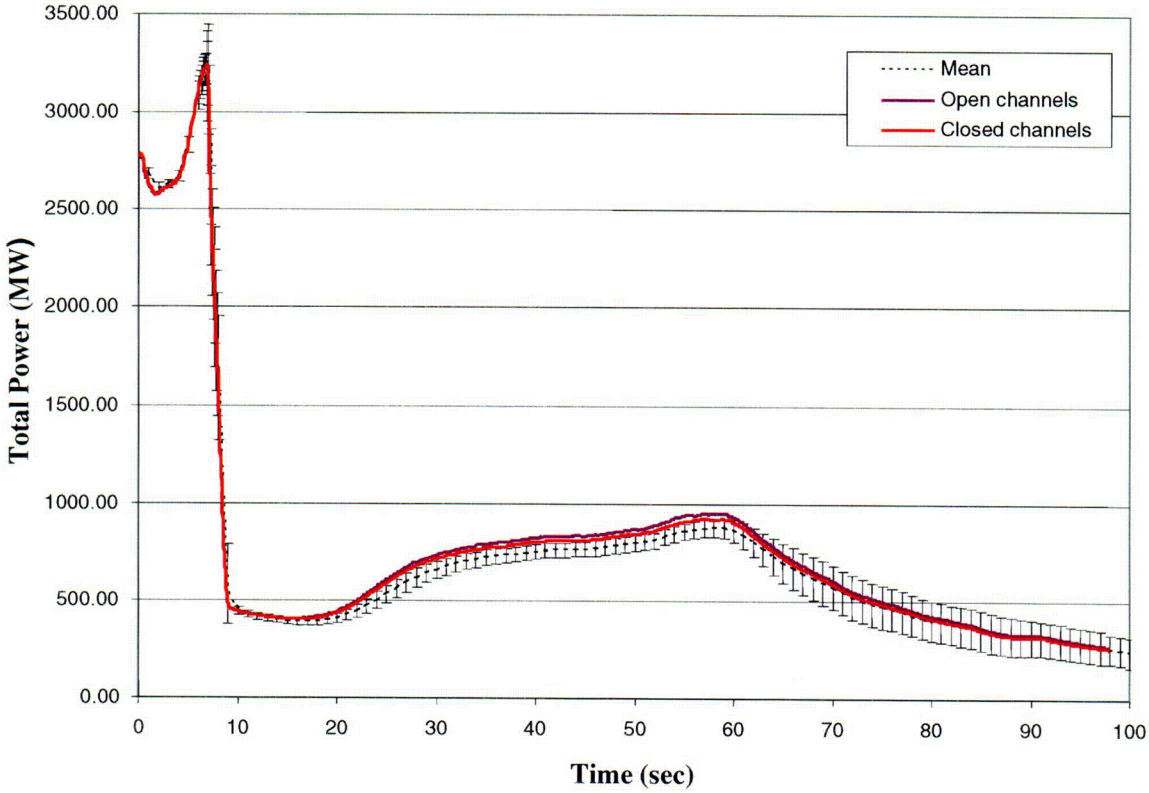

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Figure B.3-3
MSLB Benchmark Phase II Scenario 2: Core-average Coolant Density vs. Time

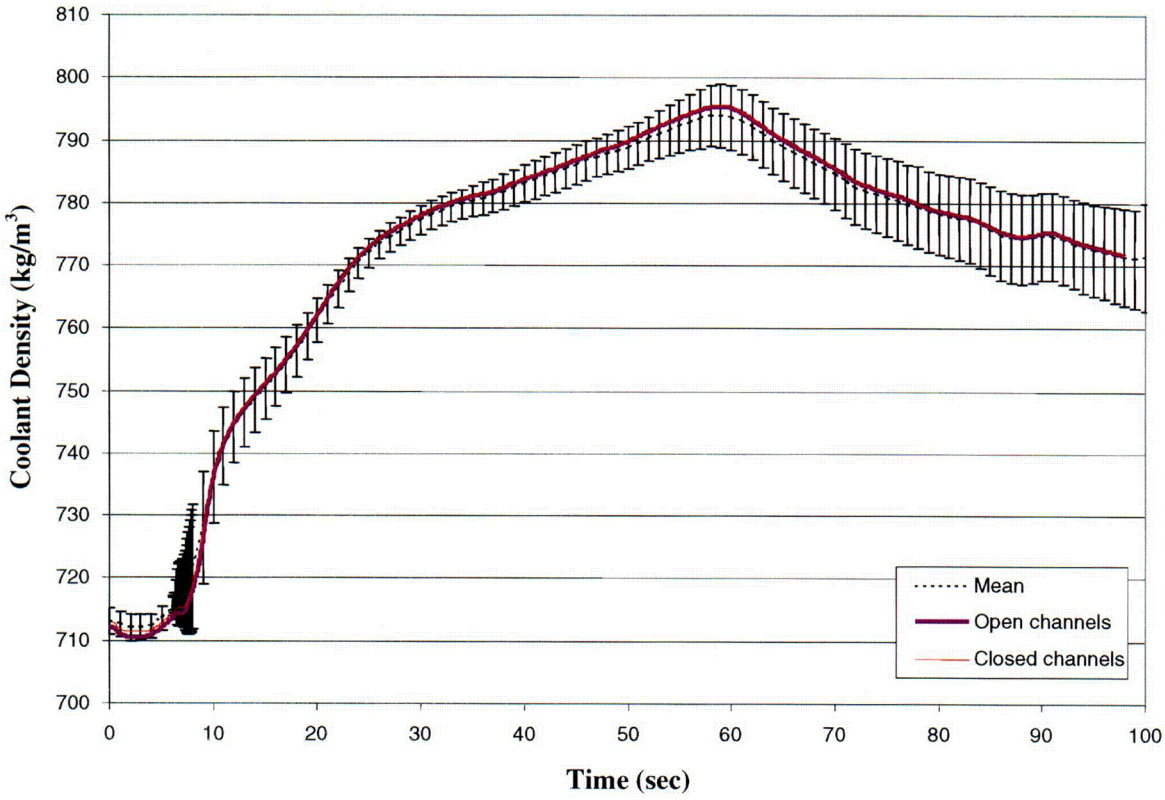

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Figure B.3-4
MSLB Benchmark Phase II Scenario 2: Core-averaged Doppler Temperature vs. Time

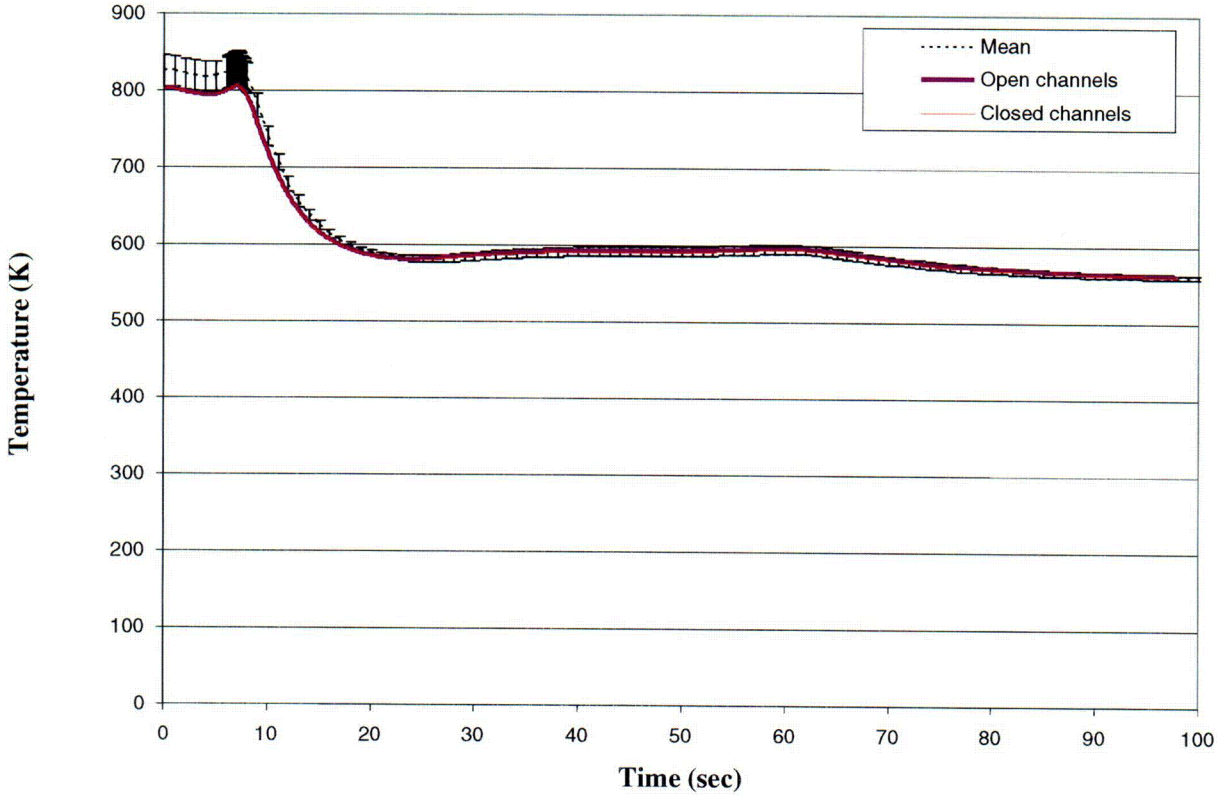

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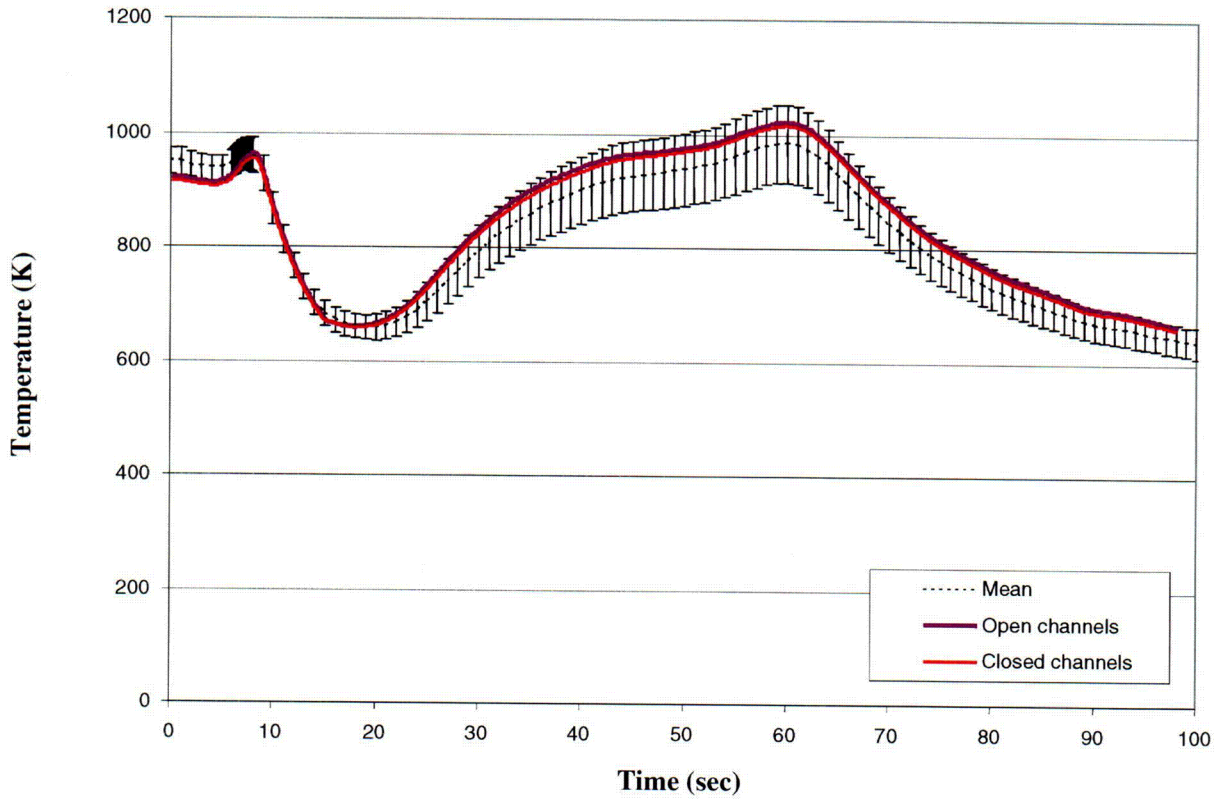
Figure B.3-5
MSLB Benchmark Phase II Scenario 2: Maximum Core Doppler Temperature vs. Time



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B.4 Phase III: Plant Transient Simulation with 3-D Core Neutronics

The externally linked RETRAN/SPNOVA/VIPRE codes were utilized to perform the coupled core-plant transient. Phase III uses the input decks developed in Phase I and Phase II, with minor modifications that were necessary to set up the interface between the base codes. Additionally, the effects of selected input assumptions on the results were investigated to better characterize the predicted transient evolution.

The results were confirmed to be in excellent agreement with the results reported for the average benchmark solution given in Reference B-4. Table B.4-1 compares the predicted total core power at key moments during the transient to the average results of the other benchmark participants.

Table B.4-1
Total Core Power for OECD MSLB Benchmark - Phase III

	Calculated Value (and deviation from mean)	Average of Benchmark Participants (and Standard Deviation)
State 5 – Maximum Core Power During the Transient		
Total Core Power (MWt)	3254.5 (Δ = -18.87)	3273.37 (σ = 40.90)
Time of State 5 (seconds)	6.32 (Δ = -0.18)	6.50 (σ = 0.71)
State 6 – Maximum Core Power After Reactor Trip		
Total Core Power (MWt)	891.14 (Δ = -70.67)	961.81 (σ = 135.71)
Time of State 6 (seconds)	66.1 (Δ = 0.59)	65.51 (σ = 4.86)
State 8 – End of Transient (100.0 seconds after break)		
Total Core Power (MWt)	247.85 (Δ = -74.5)	322.35 (σ = 127.91)

The results of all key parameters presented in Reference B-4 were confirmed to be within a single standard deviation from the average benchmark solution. Additionally, the transient behavior for various key parameters were reviewed and confirmed to be in agreement with the consensus benchmark solution. In particular, the following observations on the predicted transient behavior were considered the most relevant in the interpretation of the transient results.

Figure B.4-1 shows the total predicted break flow rate. By comparing this result with the benchmark results, it was observed that a higher than average break flow is predicted by RETRAN during the central part of the transient, between about 20 and 60 seconds. To determine the cause of this difference, the liquid and vapor flow rates at the break were reviewed and it was confirmed that the reason for this higher flow rate is the larger amount of water entrainment in the steam flow from the steam generators. This is consistent with the Homogeneous Equilibrium Model (HEM) model used by RETRAN, which assumes a uniform mixture of steam and water with no slip between the phases. This leads to an increased liquid flow rate at the break. As discussed in the benchmark conclusions (Reference B-4), the slip, steamline modeling, code correlations and various other modeling assumptions caused a number of local deviations throughout the transient.

An additional difference between the RETRAN results and the benchmark solution is observed in the split of flow between the two breaks: the RETRAN model tends to predict a larger break flow at the 8-inch break and a smaller one at the 24-inch break. This is due to differences in the steamline models used by benchmark participants. Some participants forced the two steamlines to be at the same pressure or used a single steamline with the two breaks connected. These differences were expected and were also consistent with the benchmark conclusions that provide an analogous explanation of break flow rates differences between the participants.

Figures B.4-2 and B.4-3 show the cold leg temperatures for the broken and intact loop. The results are in very good agreement with the benchmark results. The RETRAN prediction was confirmed to be sensitive to the vessel inlet and outlet mixing model. A higher amount of core inlet mixing leads to more uniform temperatures at the end of transient, but on the other hand it leads to a lower return to power during the transient. The core inlet mixing model used in RETRAN was calibrated on the specifications provided in the benchmark, but it was observed that a large range of mixing assumptions were used by the benchmark participants, which explains some of the differences in the prediction of the transient evolution. The RETRAN core inlet mixing model was observed to predict a slightly larger amount of core inlet mixing than the average of the participants.

The Total Core Power provided in Figure B.4-4 is perhaps the primary parameter of interest for this transient, as it combines the effects of both the plant parameters (inlet core temperatures, mixing models, cooldown rates) and the SPNOVA/VIPRE core model in a single parameter that can be used to evaluate the overall response of each of the different codes. Phase III results are in excellent agreement with the average benchmark solution, well within a single standard deviation. The results showed a slight under prediction of the average solution. Based on similarity between the transient parameters discussed above, this is mostly due to some input differences (model of the steamlines, core inlet mixing model).

Figure B.4-1
MSLB Benchmark Phase III Scenario 2:
Total Break Flow Rate vs. Time

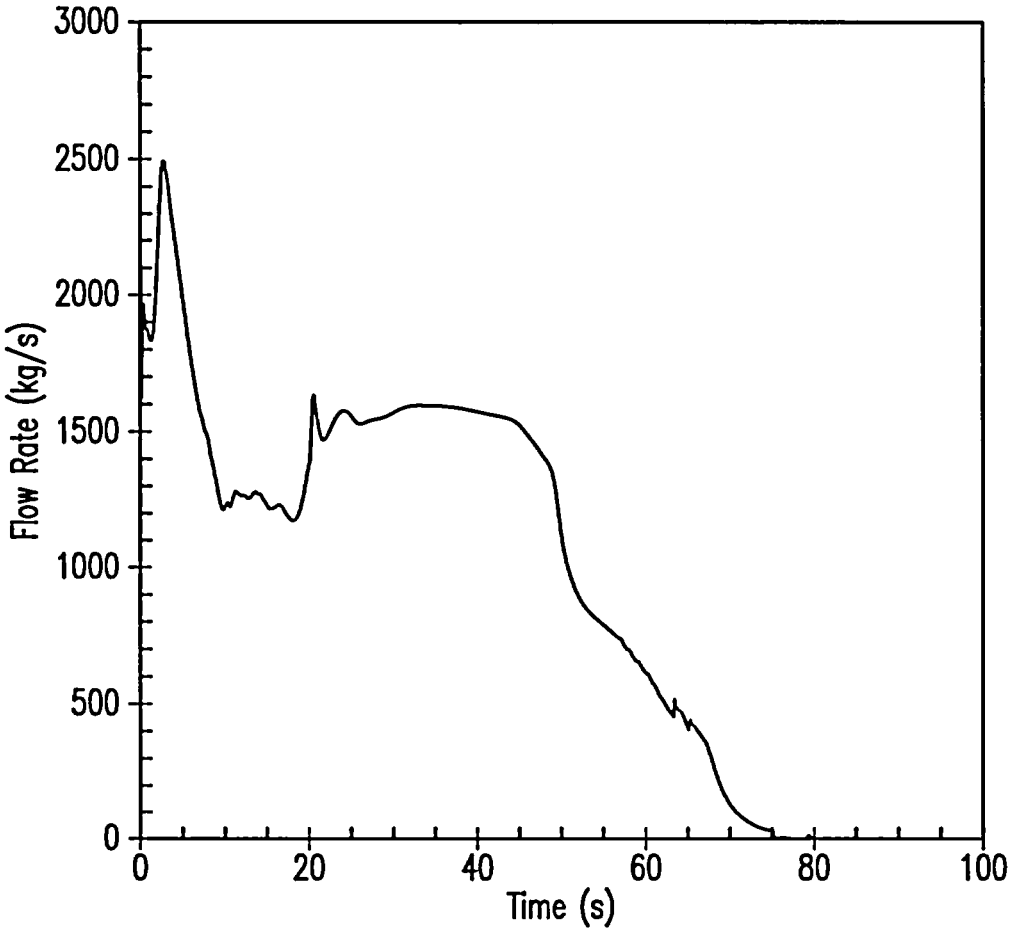

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Figure B.4-2
MSLB Benchmark Phase III Scenario 2:
Broken Loop Cold Leg Temperature vs. Time

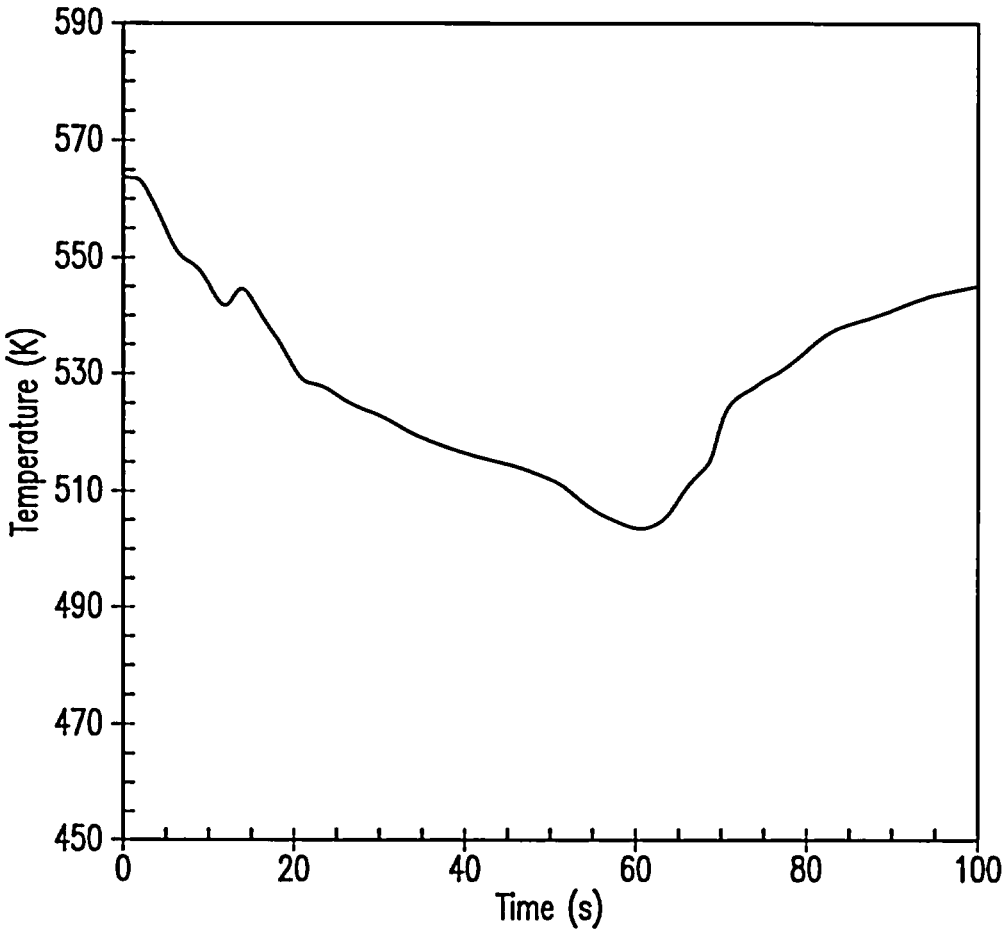

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Figure B.4-3
MSLB Benchmark Phase III Scenario 2:
Intact Loop Cold Leg Temperature vs. Time

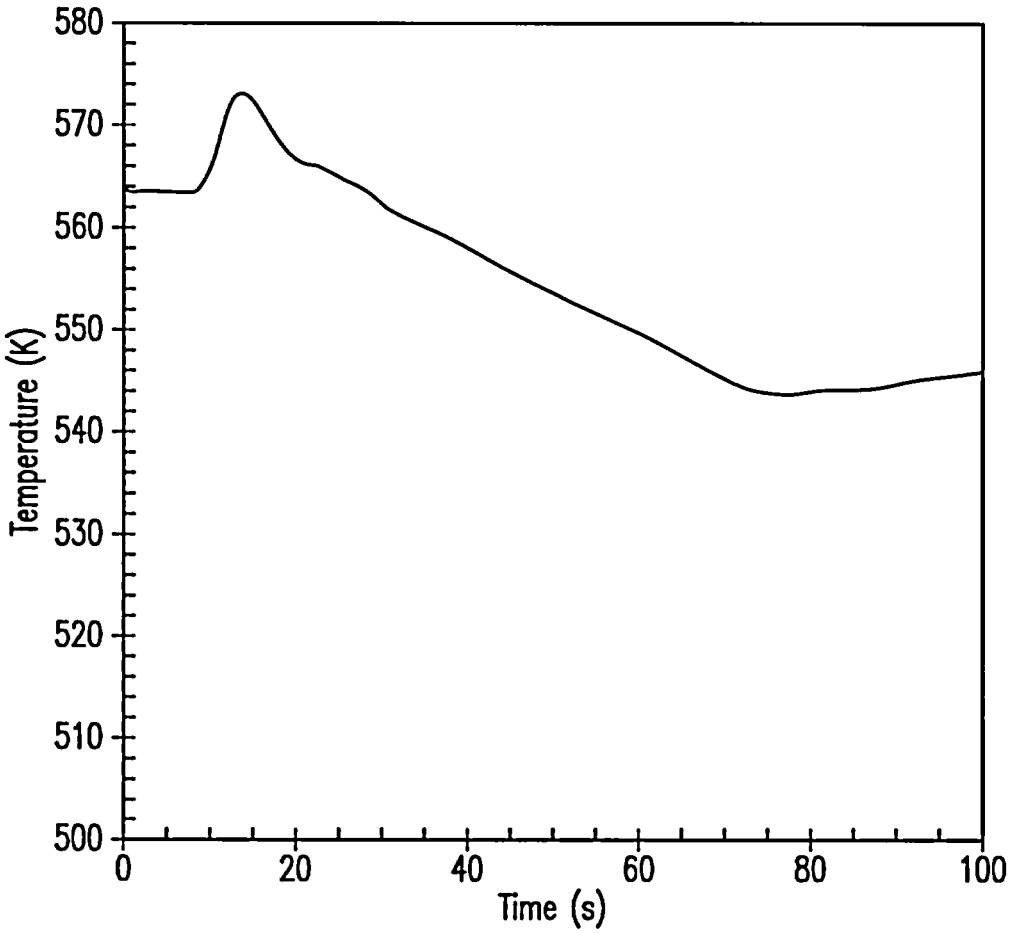

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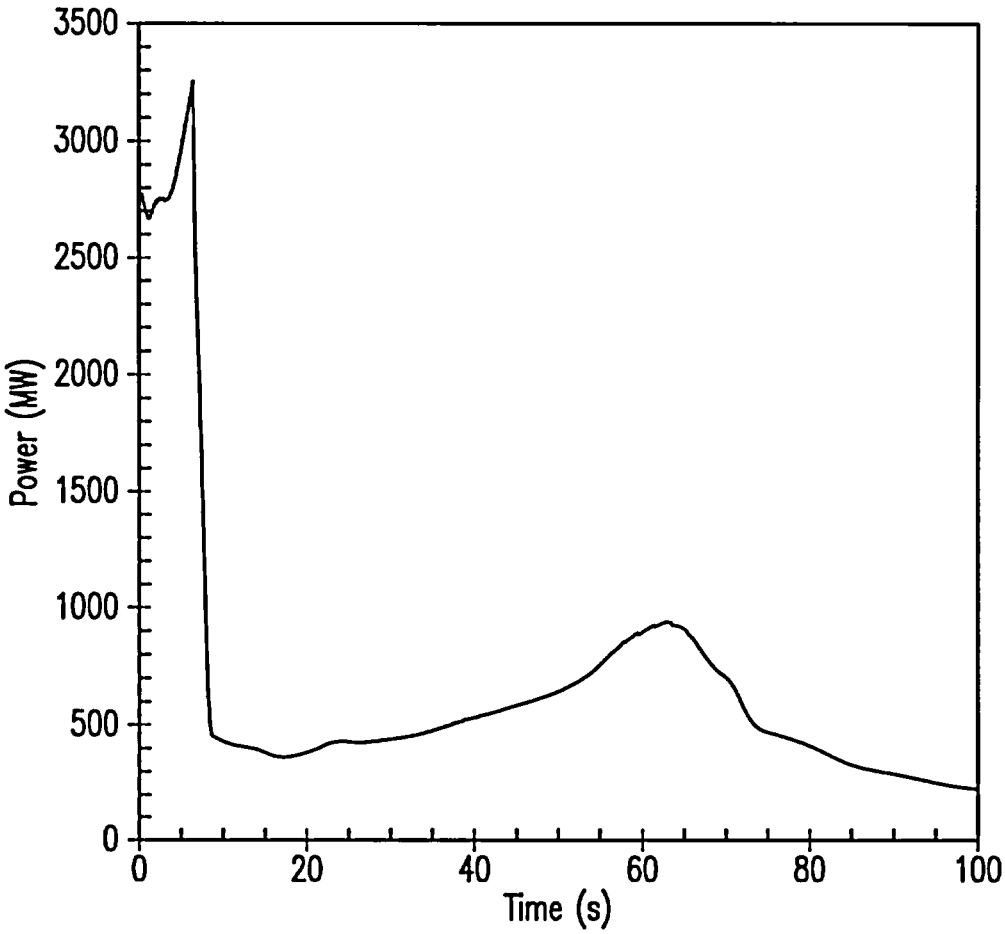
Figure B.4-4
MSLB Benchmark Phase III Scenario 2:
Total Core Power vs. Time



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B.5 Summary and Conclusions

A review of the results of the OECD MSLB Benchmark problem confirms the adequacy of the RETRAN/SPNOVA/VIPRE codes in the externally linked mode in analyzing a severe coupled core-plant transient. The stand-alone RETRAN and SPNOVA/VIPRE models were first assessed against the other benchmark participants using the results provided for the first two phases of the benchmark program. The Phase I and Phase II results were confirmed to be in very good agreement with the benchmark participants. The minor differences observed between the Phase I and Phase II results and the consensus benchmark solutions were addressed.

The Phase III of the OECD MSLB benchmark problem was performed using the externally linked RETRAN/SPNOVA/VIPRE codes. Excellent agreement against the benchmark solution was observed, with differences determined to be caused by the input and modeling assumptions observed during the first two exercises.

B.6 References

B-1. NEA/NSC/DOC(99)8, "PWR Main Steamline Break (MSLB) Benchmark, Volume I: Final Specifications," April 1999.

B-2. NEA/NSC/DOC(2000)21, "PWR Main Steamline Break (MSLB) Benchmark, Volume II: Summary Results of Phase I (Point Kinetics)," December 2000.

B-3. NEA/NSC/DOC(2002)12, "PWR Main Steamline Break (MSLB) Benchmark, Volume III: Results of Phase 2 on 3-D Core Boundary Conditions Model," 2002.

B-4. NEA/NSC/DOC(2003)21 / ISBN 92-64-02152-3, "PWR Main Steamline Break (MSLB) Benchmark, Volume IV: Results of Phase III on Coupled Core-plant Transient Modeling," 2003.

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APPENDIX C
SENSITIVITY STUDIES

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APPENDIX C SENSITIVITY STUDIES

C.1 Sensitivity Study for the Complete Loss of Flow Event

C.1.1 Description

A sensitivity study was performed to determine if the parameters selected for the base case yield the most limiting results, and to document the sensitivity of the results to variations in the parameters in order to establish the appropriateness of the application of the uncertainties. The parameters selected for the sensitivity study are essentially all 3-D neutronics parameters, since the only change to the current analysis method is the replacement of the RETRAN point kinetics model with an external 3-D core kinetics calculation. The reactor coolant system models, control and protection functions, and application of uncertainties for the systems parameters remain unchanged from the current analysis method.

C.1.2 Sensitivity Cases

The cases performed for the sensitivity study and the DNBR results are shown in Table C.1-1. Table C.1-1 also shows the minimum WRB-2 DNBR results for the base case, which was performed using the input assumptions described in Section 3.1.5. The base case input assumptions which will be varied in the sensitivity study are listed in the table under the column titled "Base Case Parameter Value". For the sensitivity studies, all parameters were assumed to be at the Base Case value, with the exception of the parameter value chosen for the specific sensitivity case. The sensitivity study value for each parameter is listed in the table under the column titled "Sensitivity Case Parameter Value". The results and conclusions are discussed below.

C.1.3 Results

Comparing the results of the sensitivity cases to the base case (Case 1) in Table C.1-1 for the complete loss of flow event shows the following:

1. [ ]a,c.

2. [ ]a,c.

3. [

]a,c.

4. [

]a,c.

5. [

]a,c.

6. [

]a,c.

7. [

]a,c.

8. [

]a,c.

9. [

]a,c.

10. [

]a,c.

11. [

]a,c.

12. [

]a,c.

C.1.4 Conclusions and Selection of a Reference Bounding Analysis Case

[

]a,c.

[

]a,c.

[

]a,c.

This combination of analysis assumptions constitutes the updated 3-D core transient methodology Reference Bounding Analysis Case for this transient.

Table C.1-1
Results of Sensitivity Study for Complete Loss of Flow Event

a, c

C.2 Sensitivity Study for the Locked Rotor – Rods in DNB Event

C.2.1 Description

A sensitivity study was performed to determine if the parameters selected for the base case yielded the most limiting results, and to document the sensitivity of the results to variations in the parameters in order to establish the appropriateness of the application of the uncertainties. The parameters selected for the sensitivity study are essentially all 3-D neutronics parameters, since the only change to the current analysis method is the replacement of the RETRAN point kinetics model with an external 3-D core kinetics calculation. The reactor coolant system models, control and protection functions, and application of uncertainties for the systems parameters remain unchanged from the current analysis method.

C.2.2 Sensitivity Cases

The cases performed for the sensitivity study and the DNBR results are shown in Table C.2-1. Table C.2-1 also shows the minimum WRB-2 DNBR results for the base case, which was performed using the input assumptions described in Section 3.2.5. The base case input assumptions which will be varied in the sensitivity study are listed in the table under the column titled "Base Case Parameter Value". For the sensitivity studies, all parameters were assumed to be at the Base Case value, with the exception of the parameter value chosen for the specific sensitivity case. The sensitivity study value for each parameter is listed in the table under the column titled "Sensitivity Case Parameter Value". The results and conclusions are discussed below.

C.2.3 Results

Comparing the results of the sensitivity cases to the base case (Case 1) in Table C.2-1 for the locked rotor rods in DNB event shows the following:

1. [ ]a,c.

2. [ ]a,c.

3. [ ]a,c.

4. [

] a,c.

5. [

] a,c.

6. [

] a,c.

7. [

] a,c.

8. [

] a,c.

9. [

] a,c.

10. [

] a,c.

11. [

] a,c.

12. [

] a,c.

It should be noted that the 3-D methodology approach for this sample plant resulted in no rods predicted to be in DNB. This will not necessarily be the case for other plants. However, since the same analysis methodology will be used for all plants, this does not affect the conclusions obtained from the sensitivity study.

C.2.4 Conclusions and Selection of a Reference Bounding Analysis Case

[

] a,c.

[

] a,c.

[

] a,c.

This combination of analysis assumptions constitutes the updated 3-D core transient methodology Reference Bounding Analysis Case for the Locked Rotor-Rods in DNB event.

Table C.2-1
Results of Sensitivity Study for Locked Rotor Rods in DNB Event

a, c

[ ]

C.3 Sensitivity Study for the Locked Rotor – Peak RCS Pressure Event

C.3.1 Description

A sensitivity study was performed to determine if the parameters selected for the base case yield the most limiting results, and to document the sensitivity of the results to variations in the parameters in order to establish the appropriateness of the application of the uncertainties. The parameters selected for the sensitivity study are essentially all 3-D neutronics parameters, since the only change to the current analysis method is the replacement of the RETRAN point kinetics model with an external 3-D core kinetics calculation. The reactor coolant system models, control and protection functions, and application of uncertainties for the systems parameters remain unchanged from the current analysis method.

C.3.2 Sensitivity Cases

The cases performed for the sensitivity study and the peak RCS pressure results are shown in Table C.3-1. Table C.3-1 also shows the peak RCS pressure reached in the base case, which was performed using the input assumptions described in Section 3.3.5. The base case input assumptions which will be varied in the sensitivity study are listed in the table under the column titled "Base Case Parameter Value". For the sensitivity studies, all parameters were assumed to be at the Base Case value, with the exception of the parameter value chosen for the specific sensitivity case. The sensitivity study value for each parameter is listed in the table under the column titled "Sensitivity Case Parameter Value". The results and conclusions are discussed below.

C.3.3 Results

Comparing the results of the sensitivity cases to the base case (Case 1) in Table C.3-1 for the locked rotor peak RCS pressure event shows the following:

1. [ ]a,c.

2. [ ]a,c.

3. [

]a,c.

4. [

]a,c.

5. [

]a,c.

6. [

]a,c.

7. [

]a,c.

8. [

]a,c.

C.3.4 Conclusions and Selection of a Reference Bounding Analysis Case

[

]a,c.

[

]a,c.

This combination of analysis assumptions constitutes the updated 3-D core transient methodology Reference Bounding Analysis Case for the Locked Rotor-Peak RCS Pressure event.

Table C.3-1
Results of Sensitivity Study for Locked Rotor Peak RCS Pressure Event

a, c

[ ]

C.4 Sensitivity Study for the Steamline Break from Hot Full Power Event

C.4.1 Description

A sensitivity study was performed to determine if the parameters selected for the base case yield the most limiting results, and to document the sensitivity of the results to variations in the parameters in order to establish the appropriateness of the application of the uncertainties. The parameters selected for the sensitivity study are essentially all 3-D neutronics parameters, since the only change to the current analysis method is the replacement of the RETRAN point kinetics model with an external 3-D core kinetics calculation. The reactor coolant system models, control and protection functions, and application of uncertainties for the systems parameters remain unchanged from the current analysis method.

C.4.2 Sensitivity Cases

The cases performed for the sensitivity study and the DNBR results are shown in Table C.4-1. Table C.4-1 also shows the minimum WRB-2 DNBR results for the base case, which was performed using the input assumptions described in Section 3.4.5. The base case input assumptions which will be varied in the sensitivity study are listed in the table under the column titled "Base Case Parameter Value". For the sensitivity studies, all parameters were assumed to be at the Base Case value, with the exception of the parameter value chosen for the specific sensitivity case. The sensitivity study value for each parameter is listed in the table under the column titled "Sensitivity Case Parameter Value". The results and conclusions are discussed below.

C.4.3 Results

Comparing the results of the sensitivity cases to the base case (Case 1) in Table C.4-1 for the steamline break from hot full power event shows the following:

1. [ ]a,c.

2. [ ]a,c.

3. [

]a,c.

4. [

]a,c.

5. [

]a,c.

6. [

]a,c.

7. [

]a,c.

8. [

]a,c.

9. [

]a,c.

10. [

]a,c.

C.4.4 Conclusions and Selection of a Reference Bounding Analysis Case

[

]a,c.

[

]a,c.

This combination of analysis assumptions constitutes the updated 3-D core transient methodology Reference Bounding Analysis Case for the Steamline Break from Hot Full Power event.

Table C.4-1
Results of Sensitivity Study for Steamline Break From Hot Full Power Event

a, c

[ ]

C.5 Sensitivity Study for the Main Steamline Break from Hot Zero Power Event

C.5.1 Description

A sensitivity study was performed to demonstrate that the parameters selected for the base case yielded the most limiting results, and to document the sensitivity of the results to variations in the parameters in order to establish the appropriateness of the application of the uncertainties. The parameters selected for the sensitivity study are essentially all 3-D neutronics parameters, since the only change to the current analysis method is the replacement of the RETRAN point kinetics model with an external 3-D core kinetics calculation. The reactor coolant system models, control and protection functions, and application of uncertainties for the systems parameters remain unchanged from the current analysis method.

C.5.2 Sensitivity Cases

The cases performed for the sensitivity study and the DNBR results are shown in Table C.5-1. Table C.5-1 also shows the minimum W-3 DNBR results for the base case assuming 1.0% Δk/k shutdown margin (Case 1), which was performed using the input assumptions described in Section 3.5.5. (The assumption of 1.0% Δk/k shutdown margin was used since this provides the most limiting results. The base case with 1.77% Δk/k SDM is presented as Case 6.) The base case input assumptions which will be varied in the sensitivity study are listed in the Table under the column titled "Base Case Parameter Value". For the sensitivity studies, all parameters were assumed to be at the Base Case value, with the exception of the parameter value chosen for the specific sensitivity case. The sensitivity study value for each parameter is listed in the Table under the column titled "Sensitivity Case Parameter Value". The results and conclusions are discussed below.

C.5.3 Results

Comparing the results of the sensitivity cases to the base case (Case 1) in Table C.5-1 for the main steamline break from hot zero power event shows the following:

1. [

   ]^{a,c}.

2. [

   ]^{a,c}.

3. [

]a,c.

4. [

]a,c.

5. [

]a,c.

6. [

]a,c.

7. [

]a,c.

8. [

]a,c.

9. [

]a,c.

C.5.4 Conclusions and Selection of a Reference Bounding Analysis Case

The sensitivity study shows that the analysis parameters chosen for the base case for this event yield the most limiting minimum DNBR. [

]a,c.

This combination of analysis assumptions constitutes the updated 3-D core transient methodology Reference Bounding Analysis Case for the Main Steamline Break from Hot Zero Power event.

Table C.5-1
Results of Sensitivity Study for Main Steamline Break from Hot Zero Power Event

a, c

[ ]