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

Westinghouse

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

Original Version: April 2004

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SECTION A

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

September 15, 2005

Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
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SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT (TR) WCAP-16259-P, REVISION 0, "WESTINGHOUSE METHODOLOGY FOR APPLICATION OF 3-D TRANSIENT NEUTRONICS TO NON-LOCA ACCIDENT ANALYSIS" (TAC NO. MC3036)

Dear Mr. Gresham:

By letter dated April 29, 2004, and as supplemented by letters dated December 16, 2004, and March 22, 2005, Westinghouse Electric Company (Westinghouse) submitted topical report (TR) WCAP-16259-P, Revision 0, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA [loss-of-coolant accident] Accident Analysis," to the U.S. Nuclear Regulatory Commission (NRC) for review. By letter dated August 1, 2005, an NRC draft safety evaluation (SE) regarding our approval of WCAP-16259 was provided for your review and comments. By letter dated August 11, 2005, Westinghouse commented on the draft SE. The NRC staff accepted all of Westinghouse's comments, which were clarifications for readability and corrections to the references.

The NRC staff has found that WCAP-16259-P, Revision 0, is acceptable for referencing in licensing applications for Westinghouse and Combustion Engineering designed pressurized-water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that Westinghouse publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Westinghouse and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

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

Project No. 700

Enclosure: Final SE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16259-P, REVISION 0,

"WESTINGHOUSE METHODOLOGY FOR APPLICATION OF 3-D TRANSIENT

NEUTRONICS TO NON-LOCA ACCIDENT ANALYSIS"

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

1.0 INTRODUCTION AND BACKGROUND

By letter dated April 29, 2004, and as supplemented by letters dated December 16, 2004, and March 22, 2005, (References 1, 2, and 3) Westinghouse Electric Company (Westinghouse) submitted Topical Report WCAP-16259-P, "Westinghouse Methodology for Application of 3-D [three-dimensional] Transient Neutronics to Non-LOCA [loss-of-coolant accident] Accident Analysis," to the Nuclear Regulatory Commission (NRC) for review and approval. The objective of this report is to provide the information and data necessary to approve WCAP-16259-P, Revision 0, as a methodology for a complete nuclear design code system for core design, safety and operational calculations. This report presents the Westinghouse Electric Company developed methodology for the analysis of non-LOCA transients and accidents for pressurized-water reactors (PWRs) using a 3-D core kinetics model.

2.0 REGULATORY EVALUATION

Part 50.34 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Contents of applications; technical information," requires that safety analysis reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload design process, licensees (or vendors) perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, licensees confirm that key inputs to the safety analyses (such as neutronic and thermal hydraulic parameters) are and will remain conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, a reanalysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

The staff review was based on the evaluation of technical merit and compliance with all applicable staff guidance associated with reviews of topical reports, including NUREG-0800 (Reference 4).

3.0 TECHNICAL EVALUATION

The objective of this report is to present the Westinghouse methodology for the application of three-dimensional core neutron kinetics and thermal hydraulics to the analysis of non-LOCA final safety analysis report (FSAR) transient and accident events. This methodology uses the NRC-approved core neutron kinetics code SPNOVA (References 5 and 6) and the NRC-approved core thermal-hydraulics code VIPRE-01 (VIPRE) (References 7 and 8), in conjunction with the NRC-approved reactor coolant system (RCS) loop thermal/hydraulics code RETRAN-02 (RETRAN) (References 9 and 10).

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 control rod ejection accidents (Reference 11). This report demonstrates that with the additional linkage to the RETRAN computer code, the updated methodology allows for a more realistic, yet conservative non-LOCA analysis with respect to the current licensing acceptance criteria. The independent code limitations and uncertainties will continue to be applicable when the codes are linked using an external communication interface. 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 Combustion Engineering (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 an approved methodology is available for the base codes (i.e., SPNOVA, VIPRE, and RETRAN).

3.1 Overview of Computer Codes

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 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 separate stand-alone calculations for the hot rod departure from nucleate boiling ratio (DNBR) and for 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 within the interface program were limited to those needed to facilitate the data transfer and interface; no changes or additions have been made to the NRC-approved models within 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 were reviewed and approved by the NRC for a severe rod ejection transient event in WCAP-15806-P-A, (Reference 11). The methodology for using VIPRE with RETRAN to provide input to SPNOVA for core reactivity and power calculations for other non-LOCA transient and accident analyses has not been previously reviewed by the staff. In addition, the

use of VIPRE for peak fuel/clad post DNBR temperature transient evaluation has not been previously reviewed by the staff. The staff's evaluations for these two new uses of VIPRE are discussed in this safety evaluation report.

In performing the required analyses for reload cores, Westinghouse will use already approved methodology (Reference 12), which provides for use of conservative code input so as to bound the expected conditions for subsequent reloads. For each reload, the "bounding" safety analysis input parameters are compared to the reload cycle's actual design values to ensure that they remain bounding. If a reload parameter is not bounded by the value used in the safety analysis, the impacted analyses are either re-analyzed or evaluated to ensure that the required margin of safety is maintained for the analyses in question.

The RAVE methodology will be implemented in accordance with the Westinghouse Quality Management System (QMS), which has been reviewed and approved by the NRC staff. The QMS provides the basis for implementation of programs such as RAVE. Work instructions are provided with detailed steps of the specific work activities. Westinghouse will maintain training guidelines that assure only qualified analysts perform and verify the analyses being performed.

3.1.1 Use of SPNOVA in Westinghouse RAVE Methodology

The current Westinghouse standard core design methodology uses a 3-D nodal expansion method for the static analysis of the cores. This methodology is approved and has been incorporated into the NRC-approved SPNOVA computer program. The static neutronics solution in SPNOVA is also consistent with the NRC-approved ANC computer program (References 13, 14, 15, and 16).

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.

Due to the previous cycle length, a potential cycle history factor is utilized to account for the impact of last cycle depletion design on the upcoming beginning-of-cycle (BOC) fuel management. 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.

The Westinghouse methodology, as presented in this submittal, will continue to use the reload safety evaluation process. Through this process, the impact of the reload cycle can be determined from static nuclear design calculations. The transient safety analysis calculations are re-performed only if the evaluated results are outside of the space defined by previously utilized key parameter limits and criteria. Key parameters for each accident are defined in Chapter 3 of the April 29, 2004, (Reference 1) submittal. All key parameters were found to be consistent with the key parameters identified in the current Westinghouse reload cycle methodology presented in Reference 12. Calculational methods of typical current static kinetic parameters, such as Doppler, moderator feedback, delayed neutron fraction and trip reactivity worth, that may affect the transient accident analysis, were also provided in the April 29, 2004, submittal.

The SPNOVA computer code also 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 are directly utilized for the transient solution. The applicable limitations and compliance associated with the use of SPNOVA for static and transient analyses are contained in the conclusion of the NRC staff's Safety Evaluation (SE), (Reference 5).

Results of the staff's review of SPNOVA show that the kinetics benchmarking provided in Reference 5 demonstrates that SPNOVA provides an accurate method for determining both the core-wide and local power and flux response during core reactivity transients. In licensing applications of SPNOVA, these conditions and limitations are required to ensure an acceptable margin to the fuel safety limits and must be provided in plant-specific submittals. These conditions and limitations associated with the application of SPNOVA within the context of the methodology proposed in the April 29, 2004, submittal of topical report WCAP-16259-P and in the supplements provided in response to NRC staff questions.

3.1.2 Use of RETRAN-02 in Westinghouse RAVE Methodology

RETRAN-02 is a flexible, general purpose, thermal/hydraulic computer code that is used to evaluate the effect of various upset reactor conditions on the RCS. The code models the reactor coolant as a single phase or as two equilibrium phases with the exception that a non-equilibrium pressurizer component can be included. Conductive heat structures can be described, including the fuel elements in the reactor core. Changes in reactor power from neutron kinetics and decay heat considerations can be calculated to occur with time.

RETRAN-02 was developed by Energy, Incorporated, for the Electric Power Research Institute (EPRI) and is similar to the RELAP4 thermal/hydraulic computer code developed by the NRC. The first version, RETRAN-01, was released by EPRI in December 1978. The code was subsequently improved to account for the slip between the phases, two-phase natural convection heat transfer, improved numerics, and other improvements. The revised code as described in reference 9, was submitted to the NRC for review as RETRAN-02. The NRC staff completed review of RETRAN-01 Mod003 and RETRAN-02 Mod002 as described in Reference 17. The countercurrent flow logic and the slip flow modeling were modified and a new heat slab model was added to the non-equilibrium pressurizer in Mod003. A new control rod model was added as an option to produce Mod004. These modifications were also approved by the NRC staff (Reference 18). The 1979 ANS 5.1 decay heat model was added to the code as Mod005. This version was also approved by the NRC staff (Reference 19). The staff's generic approval of RETRAN-02 is subject to limitations defined in the SERs for the various RETRAN-02 versions and in the technical evaluation reports (TERs) prepared by the NRC staff's contractors. In addition, because of the large flexibility in user-supplied input selection and choice of nodalization schemes, the NRC staff required that proposed applications of RETRAN-02 be accompanied by a detailed review of the suitability of the code for each specific application. These concerns were addressed by Westinghouse in WCAP-14882-P-A (Reference 10) which contains the staff SER approving use of RETRAN-02 Mod005 by Westinghouse for analysis of non-LOCA transients and accidents in 2-, 3-, and

4-loop operating plants designed by Westinghouse. This is the version of RETRAN that Westinghouse will use in the RAVE methodology. The transients and accidents for which Westinghouse received NRC staff approval for use of the RETRAN methodology in WCAP-14882-P-A are listed in Table 1.

Table 1
Non-LOCA Transients To Be Analyzed Using RETRAN

- Feedwater system malfunctions
- Excessive increase in steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steamline break
- Loss of external load/turbine trip
- Loss of offsite power
- Loss of normal feedwater flow
- Feedwater line rupture
- Loss of forced reactor coolant flow
- Locked reactor coolant pump rotor/sheared shaft
- Control rod cluster withdrawal at power
- Dropped control rod cluster/dropped control bank
- Inadvertent increase in coolant inventory
- Inadvertent opening of a pressurizer relief or safety valve
- Steam generator tube rupture

In addition to the events listed above, the Westinghouse methods and codes have been successfully applied to the analysis of asymmetric steam generator transients for CE-designed plants. Westinghouse may utilize the RAVE coupled code methodology for the analyses of CE-designed plants including asymmetric steam generator transients provided that all applications of the codes within the RAVE methodology have been reviewed and approved by the NRC staff. The conditions which the staff finds acceptable for application of the RAVE methodology including CE designs are discussed in Section 4.0, "Conclusions," to this SER.

The calculational assumptions that Westinghouse will use with the RETRAN code to describe currently operating nuclear plants were derived from the input models previously approved for use with the LOFTRAN code (Reference 20). Models for currently operating Westinghouse 2-, 3-, and 4-loop plants are described (Reference 10). Westinghouse has developed a set of "RETRAN Safety Analysis Standards" to govern the development of the input models and to define the options to be used in application to specific plant transients. Westinghouse will continue to utilize the approved RETRAN input with the RAVE methodology with the exception of the core noding and power calculation.

The principal difference between utilization of the RETRAN code as previously approved by the NRC staff and it's utilization in the RAVE methodology is in the calculation of core power and the transfer of heat from the nuclear fuel to the coolant. When RETRAN was run separately, core power was calculated using the RETRAN point kinetics model and heat generated in the fuel was transferred through the cladding and into the coolant. In the RAVE methodology, core power will be calculated by SPNOVA and heat transfer from the fuel through the cladding to the

coolant will be calculated by VIPRE. Heat flow from the fuel elements to the coolant as calculated by VIPRE is a dynamic input to the RETRAN core fluid model. The RETRAN core noding is increased from that of Reference 10 to facilitate this transfer. The NRC staff determined that the core noding in the stand-alone RETRAN model was of sufficient detail. The finer noding in the RETRAN model for RAVE is therefore also acceptable.

For analysis of main steamline breaks using the previous methodology, the iteration between the point kinetics model in RETRAN and more sophisticated multidimensional neutron kinetics computer codes was required. This is because, if the most reactive control rod is assumed to be in a stuck-out position, skewed radial power profiles could be produced which cannot be adequately addressed by the point kinetics model in RETRAN. With the previous methodology the reactivity coefficients (moderation, boron, and power) were calculated separately and input into RETRAN from a more sophisticated multidimensional neutronics compilation. Iteration was performed until the total reactivity change during the accident was conservatively predicted by RETRAN in comparison to the multidimensional neutronics code. Using this methodology, conservative predictions of reactor power were obtained by the RETRAN code point kinetics model. With the RAVE methodology, reactor power will be directly calculated using the SPNOVA 3-dimensional neutronics computer code using core thermal/hydraulic information from RETRAN and VIPRE so that this iterative procedure will no longer be necessary.

3.1.3 Use of VIPRE in Westinghouse RAVE Methodology

VIPRE is a subchannel thermal/hydraulic computer code that is typically used to describe the reactor core of a nuclear power plant. The code requires that users enter the boundary conditions describing the coolant entering the core, the power generation, and the dimensional and material properties of the nuclear fuel. The boundary conditions for the coolant entering the core include the inlet flow rate, enthalpy, and pressure or the pressure, inlet enthalpy, and differential pressure from which the inlet flow rate can be derived. The core power generation input includes spatial as well as temporal variations. The code input is versatile and flexible, providing the user with numerous options. These include choices among correlations for heat and mass transfer that are built into the code. Multiple channels can be described and cross flow is calculated based on user supplied input.

VIPRE was developed by Battelle Pacific Northwest Laboratories under the sponsorship of the EPRI and submitted to the NRC for generic review (Reference 7). The staff's generic review as discussed in our SER (Reference 21) was limited to PWR applications and to heat transfer regimes up to the critical heat flux. The review included an audit calculation using the COBRA-IV code (Reference 22) and the comparison of VIPRE results to experimental test data. The review was stated to consist primarily of an evaluation of the internal program, including the governing conservation equations and constitutive equations, including the two-phase flow and heat transfer models and the numerical solution techniques. The staff required each VIPRE user to submit documentation describing the proposed use for the code, other computer codes with which it will interact, the source of each input variable, and the selected correlations, including justification for using the selected correlations. In particular, it was required that any new critical heat flux (CHF) correlations that are to be used within VIPRE be evaluated against their experimental database to determine the appropriate DNBR safety limit.

In April 1997, Westinghouse submitted topical report WCAP-14565 describing use of VIPRE for departure from nucleate boiling (DNB) analysis for those FSAR Chapter 15 transients and accidents for which DNB might be of concern.

Use of VIPRE for this type of analysis replaced the THINC-IV (Reference 23) and FACTRAN (Reference 24) codes, both of which were previously approved by the NRC staff. The THINC-IV code performs thermal/hydraulic calculations within the fuel channels, including DNBR evaluation at the fuel pin surface. For calculations in which transient heat conduction within the fuel pins is important, this calculation is performed by FACTRAN. FACTRAN describes the conductive heat transfer within the fuel pin interior and the convective heat transfer at the surface. Iteration may be required between the two codes. Both the thermal/hydraulic and the conduction/convection calculations are performed simultaneously in VIPRE. The NRC staff approved use of VIPRE for Westinghouse use in making DNBR calculations as described in the SER included with WCAP-14565-P-A (Reference 8).

Westinghouse will use three types of VIPRE models with the coupled computer code RAVE methodology. To describe the detailed thermal hydraulics conditions in the reactor core for use by the SPNOVA neutronics computer code, Westinghouse has developed a whole-core VIPRE model. Each node in the whole-core VIPRE model will communicate to a corresponding node in the SPNOVA reactor physics model. In addition to the whole-core model, Westinghouse will continue to use stand-alone VIPRE models described in WCAP-14565-P-A to calculate DNBR. Westinghouse also plans to use stand-alone VIPRE to calculate post- CHF core heat-up in a manner similar to that which the NRC staff has approved using FACTRAN.

These stand-alone models differ in that, for DNBR and core heat-up evaluation, only a portion of the core needs to be described in the simulation, as opposed to the entire core for coupling to SPNOVA. Furthermore, core heat transfer is made to be conservative in the heat-up and DNBR simulations, whereas the whole-core model uses more realistic assumptions to calculate core heat transfer. Since the purpose of the whole-core VIPRE model is to provide fuel and coolant conditions to evaluate reactivity in the reactor physics calculation, the selection of conservative heat transfer assumptions is not obvious and the use of more realistic assumptions is appropriate.

VIPRE does not model the effects of burn-up within the fuel rods. These effects include fuel pellet swelling, clad shrinkage, and increased internal gas pressure. The Westinghouse fuel design computer codes, which do evaluate these effects, are utilized to calibrate the VIPRE fuel rod input over the range of burn-up needed to represent the fuel. The initial gap size and fuel conductivity are adjusted until the resulting VIPRE calculated fuel and cladding temperatures compare with the temperatures from the design model at all power levels. Sample temperature calibration results showing agreement between the VIPRE predictions and those of the design models are presented in Reference 2.

3.1.4 Staff Review of Whole-Core VIPRE Model

As a part of the RAVE methodology, Westinghouse has developed a whole-core thermal/hydraulic model to continuously provide core moderator densities and fuel temperatures for the purpose of determining local reactivity feedback with the SPNOVA

neutronics computer code. The whole-core VIPRE model will continuously receive core inlet flows, and temperature and exit pressures from the RETRAN reactor system model. Because the entire core is modeled, a burn-up specific rod type can be described for each bundle in the core. The VIPRE initial temperature for each rod type is calibrated against Westinghouse's fuel design codes as a function of power level.

There will be a one-to-one correspondence between the nodes within the SPNOVA simulation of the core and that of VIPRE. Therefore each neutronics node in the SPNOVA model will receive moderator density and fuel temperature information from the corresponding VIPRE thermal/hydraulic node. The NRC staff agrees that this degree of noding detail is appropriate since interpolation errors that could occur, if the thermal/hydraulic and neutronics noding schemes were different, are avoided. The noding detail for the whole-core model provides additional detail from the DNBR stand-alone model which was previously shown to be adequate. The staff concludes that the whole-core model noding detail as proposed by Westinghouse is adequate.

Other input assumptions which the staff questioned as part of the whole-core model review were the modeling of the fuel-to-cladding gap, core voiding and core inlet flow mixing. The initial fuel-to-cladding gap thickness is a function of fuel burn-up. This is an input to the VIPRE code. During a transient, the gap thickness will change as a function of the fuel temperature and differential pressure across the cladding. In the DNBR stand-alone model, Westinghouse uses bounding values of the gap conductance. Use of bounding values for providing input to SPNOVA is not appropriate since assumptions which are conservative for predicting fuel or cladding temperatures may not be conservative for the neutronics calculation. In the whole-core model, Westinghouse will use the dynamic gap model in VIPRE. This model was approved by the NRC staff in the generic review of VIPRE. The model was found to be similar to the NRC staff-developed computer codes GAPCON and FRAP and to be extensively benchmarked to experimental data. Changes in gap width caused by elastic and thermal stresses are evaluated. The staff concluded that the fuel rod heat-conduction model including the dynamic gap conduction model is acceptable for licensing analysis.

Voiding in the coolant provides a negative reactivity contribution for the reactor cores designed by Westinghouse. It will therefore be conservative to minimize the calculated core voiding in the whole-core VIPRE model used to provide values of coolant density to SPNOVA. In the DNBR stand-alone VIPRE model which the staff has already approved, Westinghouse uses assumptions which underpredict the rate of steam separation from the water in the core and hence tend to overpredict the amount of core voiding. This is conservative for calculating DNBR. For the whole-core model, Westinghouse will use assumptions which minimize the reactivity feedback from core voiding while the coolant is below the boiling temperature. For core channels in which the bulk coolant temperature reaches saturation, bulk boiling will occur. Comparisons of VIPRE predications with experimental data (Reference 7) did not show any significant deviation from the measured void fraction for low steam qualities regardless of the steam/water separation model used to predict core voiding. Westinghouse performed sensitivity studies which demonstrate that the reactor power calculated by SPNOVA is insensitive to the core voiding model used in the whole-core VIPRE model up to a steam void fraction of 30 percent. Westinghouse does not believe that steam voiding will be an issue for calculating reactivity feedback using the RAVE methodology. If the maximum void fraction in

any RAVE reactivity feedback calculation exceeds 30 percent, additional justification will be provided by the licensees in plant-specific applications for the steam/water separation model utilized in the VIPRE whole-core model.

During the evaluation of the Westinghouse application for RETRAN (Reference 10), the NRC staff reviewed the assumptions available to the user for the amount of mixing that occurs in the coolant entering and exiting the core. Coolant mixing is important for analysis of transients and accidents such as the asymmetric cooldown that would occur as the result of the break of a single main steamline. Cooler water entering the core from the affected loop will cause space-dependent reactivity changes in the core, which will affect the calculation of power. For analysis of thermal asymmetry within the coolant loops, Westinghouse uses mixing inputs previously approved for use with LOFTRAN. These inputs are called "design mixing" and are based on scale mixing tests for the Indian Point 2 reactor vessel. The tests were set up to simulate 2-, 3-, and 4-loop plants. The mixing coefficients were confirmed by comparison with data from 3-loop scale model reactor vessel experiments in Europe. The data used to verify "design mixing" assumptions were all taken at flow conditions designed to simulate reactor coolant pump operation. Westinghouse uses the "design mixing" assumptions with RETRAN to analyze most of the transients and accidents for which the reactor coolant pumps are assumed to be in operation. For other transients, such as loss of offsite power, Westinghouse assumes perfect mixing of the fluid entering and exiting the core. This is acceptable since asymmetric cold-leg temperatures will not occur for these transients.

For natural-circulation conditions, data taken at a European reactor have demonstrated that perfect mixing is a valid assumption for computing the temperature of water exiting a reactor vessel and this is the assumption that Westinghouse will use for natural-circulation conditions. The NRC staff reviewed the European data and confirmed perfect mixing to be valid for reactor vessel thermal/hydraulic analysis during natural circulation (Reference 10).

Similar to the mixing assumed using RETRAN, Westinghouse will input inlet mixing assumptions to the whole-core VIPRE model. Since the whole-core VIPRE model has more core detail than does the RETRAN core model, Westinghouse will use a "fine mesh model" to describe the inlet temperature distribution. With the fine mesh model the total core mass flow and enthalpy are preserved. As a result of staff questions, Westinghouse provided validation for the fine mesh mixing model. Validation included benchmarking of the model against the original data from the reactor vessel scale model tests until the predicted local core inlet enthalpies closely matched those of the test data. Thus, the fine mesh mixing factors to be used in the whole-core VIPRE model are consistent with the measured mixing factors across the core inlet and are in good agreement with the design mixing model used with RETRAN.

3.1.5 NRC Staff Review of VIPRE for DNBR Prediction

Use of VIPRE for DNBR prediction in the hot channels of a reactor core undergoing a design-basis non-LOCA transient or accident has been previously reviewed and approved by the NRC staff (Reference 8). VIPRE input options are versatile and flexible to permit numerous applications. A number of the options are evaluated in Volume 4 of the EPRI VIPRE manual (Reference 7). By comparison with experimental data, Westinghouse chose to use options that

would make VIPRE results approximate those obtained using the previously-approved THINC-IV and FACTRAN codes. A summary of the input options chosen by Westinghouse appears in Table 3-1 of Reference 8.

Westinghouse uses a multi-channel model to determine DNBR for the hot rod so that the effect of coolant channel cross flow can be included. For cores containing only one type of fuel, a one-eighth core segment is modeled. Reference 8 contains diagrams of the radial noding for 2-, 3- and 4-loop plants. When a reactor core is loaded with more than one type of fuel element, the coolant may preferentially flow through one type of fuel, thereby reducing flow in the other. Under these conditions, a DNBR penalty is applied to account for the reduced coolant flow rate (Reference 25). Extra axial noding detail is applied to evaluate flow redistribution. Conservative assumptions are made for thermal mixing. When the DNBR penalty is applied to plant-specific transient analysis, Westinghouse will ensure that the conditions for the analysis under consideration are within the range of applicability or are bounded by conditions considered in any generic VIPRE calculation of the transition core. Westinghouse will continue to follow the current practice of assuming fuel cladding failure for any fuel rod which exceeds the DNBR limit.

Several of the transients and accidents that are part of the design basis for Westinghouse operating plants, for example a steamline break with a stuck control rod cluster, involve perturbed neutron flux distributions that cannot be assessed using the point kinetics model in RETRAN. With the current methodology, Westinghouse used a separate multidimensional neutronics computer code to determine the perturbed neutron flux shape. The resulting neutron flux shape is then input into the VIPRE stand-alone DNBR model to determine the hot-channel critical heat flux (CHF). With the RAVE methodology, the perturbed flux shape from the SPNOVA neutronics calculation will be input into the stand-alone VIPRE models for CHF or fuel rod heat-up evaluations.

3.1.6 *Staff Review of Stand-alone VIPRE for Post-CHF Fuel Heat-up Calculations*

The NRC staff considers certain design-basis accidents to be sufficiently unlikely to occur within the lifetime of a plant that a certain amount of calculated fuel failure is permitted (Reference 4). Post-CHF core heat-up is therefore evaluated to determine the extent of any fuel failure for calculation of the offsite dose and to ensure that the reactor core remains in a coolable geometry. The Staff's review of VIPRE in Reference 8 did not extend to: (1) the use of VIPRE for post-CHF heat-up calculations, and (2) the generic review of VIPRE (Reference 7) into that range. Westinghouse currently analyzes post-CHF fuel heat-up using the FACTRAN code in combination with THINC-IV or VIPRE.

As part of the RAVE review, Westinghouse has submitted additional information, which demonstrates that the post-DNBR core heat-up assumptions which Westinghouse will use with VIPRE are the same as those with the FACTRAN code which has been approved by the NRC staff. The staff evaluated the post-DNBR heat-up assumptions for VIPRE as a part of the RAVE review. The convective heat transfer and zirconium-water reaction correlations for the VIPRE model are the same as what the NRC staff previously approved for FACTRAN and are, therefore, acceptable. Other features of the VIPRE fuel rod model include the pellet power

profile model and the pellet-clad gap conductance model. These are also the same as previously approved for FACTRAN and are also acceptable. For fuel heat-up calculations with VIPRE, Westinghouse will use the multi-channel modeling detail approved by the NRC staff in Reference 8.

In response to NRC staff questions, Westinghouse submitted analyses showing that for post-CHF core heat-up, VIPRE input as modified by Westinghouse and FACTRAN produce virtually identical results. Therefore, the NRC staff considers VIPRE to be equivalent to FACTRAN for performing post-CHF core heat-up calculations. As is permitted for FACTRAN, VIPRE can be used to show compliance with acceptance criteria for peak cladding temperature for a locked rotor event, fuel melting, and pellet enthalpy criteria as well as for DNBR evaluation. Neither VIPRE nor FACTRAN includes the time-dependent physical changes that may occur in a fuel rod at elevated temperatures. Therefore VIPRE cannot be used to predict such failures and another fuel code should be used to predict mechanical behavior.

3.2 Coupling Issues (Sensitivity Studies and Convergence)

In using the RAVE methodology Westinghouse will retain the basic conservatisms of current safety analyses. In calculation of reactor power, uncertainty allowances will be applied to the Doppler and moderator feedback as well as to the delayed neutron fraction. Shutdown reactivity will be reduced by the assumption that a single rod cluster or shutdown bank fails to insert. Control rod insertion rates and reactor trip set points will be applied using technical specification conservatisms. Initial thermal/hydraulic conditions will be determined using existing approved methodology which account for uncertainty using statistical methodology or by applying the maximum steady-state allowances. The uncertainty values are determined on a plant-specific basis and will not be affected by use of the RAVE methodology. Assumptions for local peaking factor uncertainty, local engineering peaking factor penalties, and core calorimetric uncertainty will also remain unchanged. These and other conservatisms that Westinghouse will use with the RAVE methodology and which are unchanged from the current methodology of running the neutronics and thermal/hydraulic codes separately are described in Sections 2.5 and 2.6 of WCAP-16259-P.

Using the existing methodology for which SPNOVA, RETRAN, and VIPRE were run separately, assumptions were made which lead to conservative results for each code. For example, in running VIPRE for hot channel DNBR analysis it is usually conservative to assume an upward tilted power shape in the core so that the hottest fuel region will be adjacent to coolant that has been heated by traveling up most of the core length. For the neutronics calculations, it is conservative to assume a bottom tilted flux shape so that following a reactor trip the maximum time will be required for the control rods to reach the location of peak power. With the RAVE coupled code methodology, the same power shape will be assumed for both the DNBR and the neutronics calculations. Because of competing effects between the coupled computer codes, the most conservative assumptions will in many cases no longer be obvious. Sensitivity studies will need to be performed in which input assumptions are varied to enable the most conservative plant conditions to be determined. Appendix C to WCAP-16259-P describes sensitivity studies performed by Westinghouse for the postulated complete loss of forced coolant flow, locked reactor coolant pump rotor, and main steamline break events. These analyses were for a typical operating plant designed by Westinghouse with three reactor

coolant loops. Westinghouse recognizes that different core designs may exhibit different sensitivities. Therefore, Westinghouse will perform sensitivity studies for every new reactor type, core type or fuel combination to which the RAVE methodology is applied to ensure that the limiting conditions have been identified.

With three computer codes running simultaneously and constantly transferring information, it is important that convergence among the three codes be maintained. The RAVE methodology provides many warning messages and error checks to help ensure that the code set is being used correctly. If during an analysis using the RAVE methodology certain key parameters begin to diverge, a warning message is generated. The code analyst will then be required to determine the cause of the imbalance and take corrective action. The NRC staff reviewed the error checks to be performed and agrees that the code convergence checks and remedial actions proposed by Westinghouse are sufficient.

3.3 Comparison of RAVE Results with NEA Main Steamline Break Benchmark

Following a main steamline break (MSLB) the cooling of the reactor core by the increased steam flow might cause a return to power even after the control rods are tripped. The assumption that one control assembly did not insert would cause significant perturbations within the reactor core which could only be adequately addressed by a 3-D neutronics code coupled with thermal/hydraulic methodology. In an international cooperative program sponsored by the Nuclear Science Committee of the Nuclear Energy Agency (NEA), the NRC staff with the assistance of Penn State University developed a PWR main steam break test problem (MSLB-TP). The purpose of the MSLB-TP is to compare the results from international participants using different methodologies so that deviations in the calculated predictions can be evaluated. Since the MSLB-TP does not utilize an experimental test facility, no definitive conclusions can be made for the accuracy of the predictions. The comparisons do provide opportunity for examination of deviations between the predicted results which may aid in the identification of code or modeling errors.

Westinghouse provided the staff with comparisons of their predictions with those of the other participants. In general the Westinghouse predictions are within one standard deviation from those of the other participants for break flow rate, cold leg temperature, and core power versus time. Late in the analysis, the Westinghouse predictions deviate from the responses of most of the other participants. This is because Westinghouse modeled the once-through steam generators (OTSGs) in the test problem as having homogenous flow with the steam and water having the same velocity. The OTSGs did not have internal steam separating equipment. However, the homogeneous flow assumption used by Westinghouse predicted excessive water to be discharged from the break and reduced the calculated reactor system cooling from that predicted by most of the other participants. The Westinghouse 2-, 3-, and 4-loop operation plants, for which Westinghouse has requested NRC staff approval for the RAVE methodology, do not have OTSGs and instead have U-tube type steam generators which have internal steam separation equipment. For analysis of the operating Westinghouse plants, Westinghouse will assume perfect steam separation within the steam separation equipment so that the steam generator water will remain in the steam generators for maximum heat removal. This assumption is conservative for predicting reactor system cooldown following a main steamline break and has been accepted by the NRC staff (Reference 10).

4.0 CONDITIONS AND LIMITATIONS

The NRC staff accepts the methodology described in WCAP-16259-P, subject to the following conditions and limitations:

1. Consistent with the guidance contained in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," a methodology that is used in the evaluation of the cycle-specific safety limits and plant safety analyses needs to be incorporated into the technical specification (TS) list of references. Therefore, the implementation of RAVE on a plant-specific basis requires a TS amendment by the plant when the RAVE methodology is first implemented for that plant.

2. Because of competing effects between the coupled computer codes, the most conservative assumptions will, in many cases, no longer be obvious. Sensitivity studies will need to be performed to determine the most conservative plant conditions. Since different core designs may exhibit different sensitivities, the first implementation of the RAVE sensitivity studies should be performed to ensure that the limiting conditions have been identified. The sensitivity results will accompany the analyses using the RAVE methodology whenever the RAVE methodology is first implemented for a plant and must be presented to the NRC staff for review and approval.

3. As support for the TS amendment, licensees implementing RAVE should provide justification that SPNOVA, VIPRE, and RETRAN computer codes and methodology are approved for use in compliance with the conditions identified in the NRC staff SEs. The methodology for use of the VIPRE code shall be considered to be reviewed and approved for use in the RAVE methodology if all three applications of VIPRE have been reviewed and approved by the NRC staff. The three applications of VIPRE are the whole-core model, the DNBR model, and the post-CHF fuel heat-up model.

   If a specific plant has not been licensed for the use of the computer codes and methodology that are utilized by RAVE then that licensee will need to take appropriate licensing action for application of these computer codes. Licensees will need to verify that the conditions and limitations imposed on each of the three NRC approved codes (SPNOVA, RETRAN, and VIPRE), encompassing the RAVE methodology, will continue to be satisfied each time the RAVE methodology is utilized.

4. Westinghouse submitted analyses showing that for post-CHF core heat-up, VIPRE input, as modified by Westinghouse and FACTRAN, produce virtually identical results. Therefore, the NRC staff considers VIPRE to be equivalent to FACTRAN for performing post-CHF core heat-up calculations. As is permitted for FACTRAN, VIPRE can be used to show compliance with acceptance criteria for peak cladding temperature for a locked rotor event, fuel melting, and pellet enthalpy criteria as well as for DNBR evaluation. Neither VIPRE nor FACTRAN include the time-dependent physical changes that may occur in a fuel rod at elevated temperatures. Therefore, VIPRE cannot be used to predict such failures and another fuel code should be used to predict mechanical behavior.

5. The code option selected for use with whole-core VIPRE model may not be conservative for calculation of reactivity feedback for elevated steam void fractions. Westinghouse performed sensitivity studies which demonstrated that the reactor power calculated by the RAVE methodology is insensitive to assumptions for core voiding up to a maximum steam void fraction of 30 percent. If the maximum void fraction in any RAVE reactivity feedback calculation exceeds 30 percent, additional justification will need to be provided for the steam/water separation model utilized in the VIPRE whole-core model to the staff for additional review of that application of RAVE.

5.0 CONCLUSION

Based on NRC's review of WCAP-16259-P and its analyses and supplements, the staff concludes that the information and data presented provide the basis for its approval as a methodology for the analysis of non-LOCA transients and accidents. All issues associated with the review of this submittal were resolved by Westinghouse and the NRC staff. In addition, the April 29, 2004, submittal, as supplemented, is in accordance with 10 CFR 50.34, "Contents of applications; technical information" and the applicable sections of NUREG-0800.

In addition, the staff considers the methodology as described in topical report WCAP-16259-P as more realistic and consistent with present core behavior and management, but also still conservative. The methodology utilizes the NRC-approved codes SPNOVA (References 5 and 6), VIPRE-01 (References 7 and 8), and RETRAN-02 (Reference 9 and 10), 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 within 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 11).

Therefore, on the basis of the above review and justification, the staff concludes that the proposed methodology presented in WCAP-16259-P is acceptable, subject to the above discussed conditions and limitations.

6.0 REFERENCES

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Date: September 15, 2005

SECTION B

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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.

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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E. Kendrick, 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.

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ACKNOWLEDGEMENTS

The authors would like to thank Dr. Kostadin N. Ivanov from The Pennsylvania State University for his assistance on OECD Main Steamline Benchmark problem, and Messrs. J. E. Moon, H. S. Woo and Song K. Sung from KNFC for their contribution to the sensitivity analyses.

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)

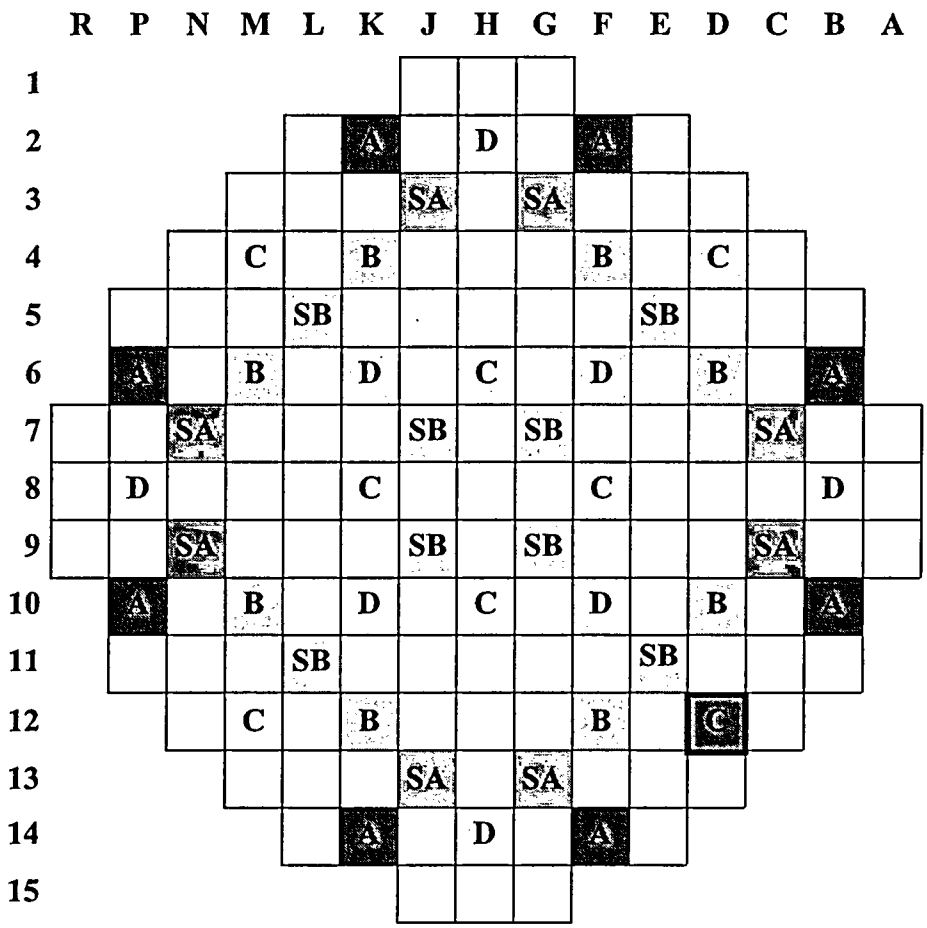
The reactor core model used in the updated 3-D core transient analysis methodology is identical to the core design model approved for use in WCAP-15806-P-A. No new models were developed for this analysis. The models used are described below.

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

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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.

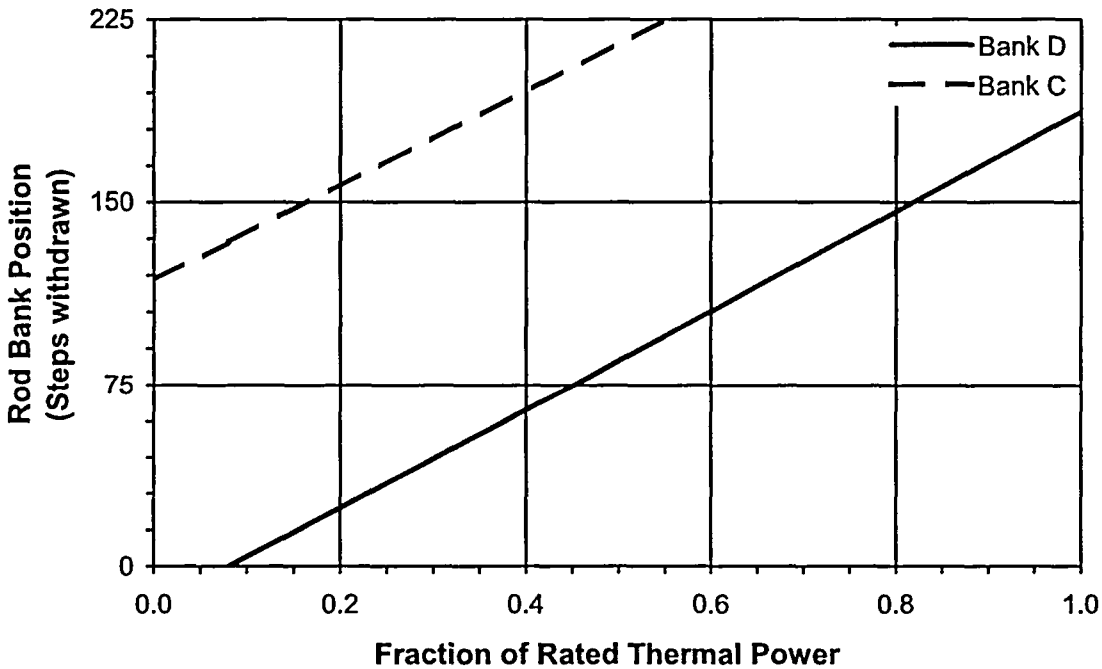
In the power operating range, there is a nominal operating range in which the reactor is allowed to operate. This band of operation is typically defined by axial flux difference (AFD) limits as a function of power level (Reference 16). Note that AFD is identical to axial offset (AO) at hot full power. 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

a. c

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

a. c.

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		2	2	2	1	7	7	7	13		
6		15	15	9	9			2	2	2	1	1	7	7	13	13	
7		15	15	9	9				2	1	1	1	7	7	13	13	
8		16	15	9	10	4		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

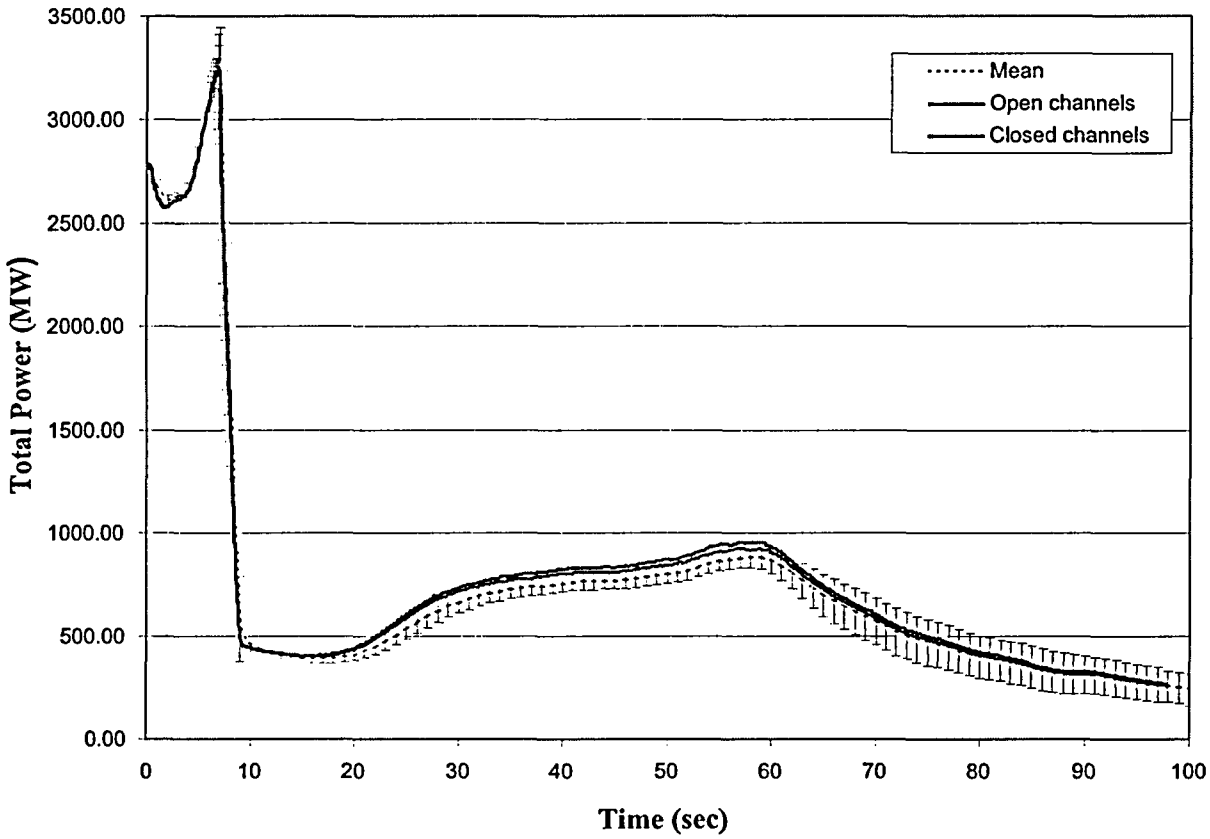

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Figure B.3-3
MSLB Benchmark Phase II Scenario 2: Core-average Coolant Density vs. Time

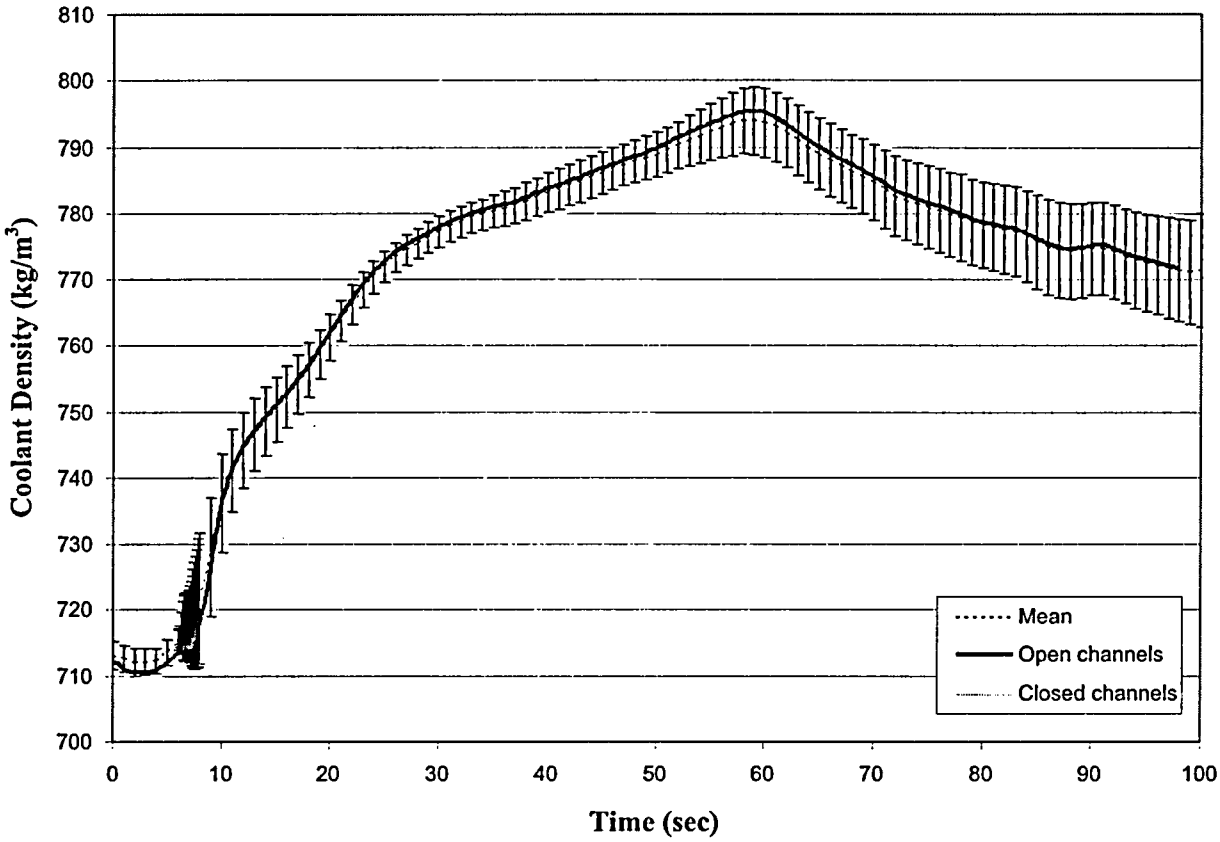

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Figure B.3-4
MSLB Benchmark Phase II Scenario 2: Core-averaged Doppler Temperature vs. Time

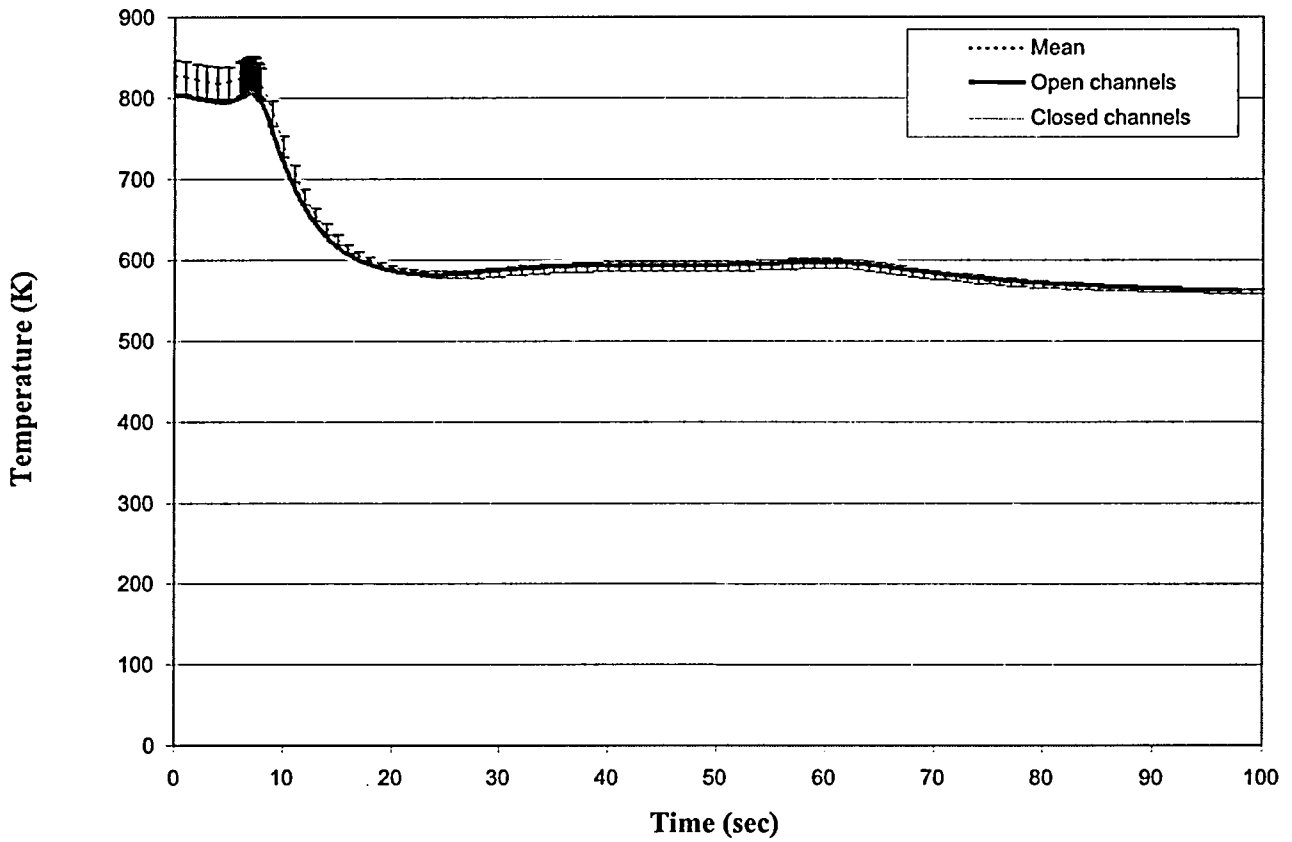

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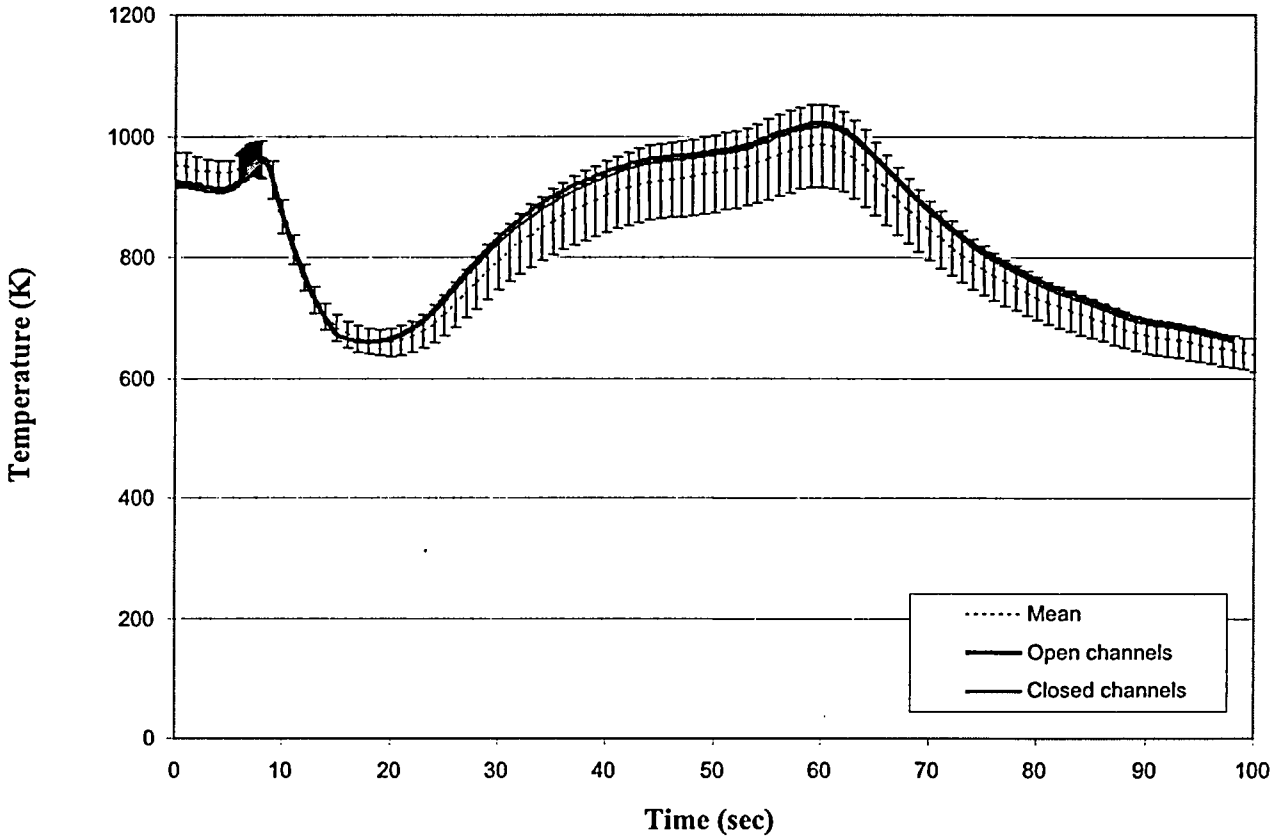
Figure B.3-5
MSLB Benchmark Phase II Scenario 2: Maximum Core Doppler Temperature vs. Time



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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

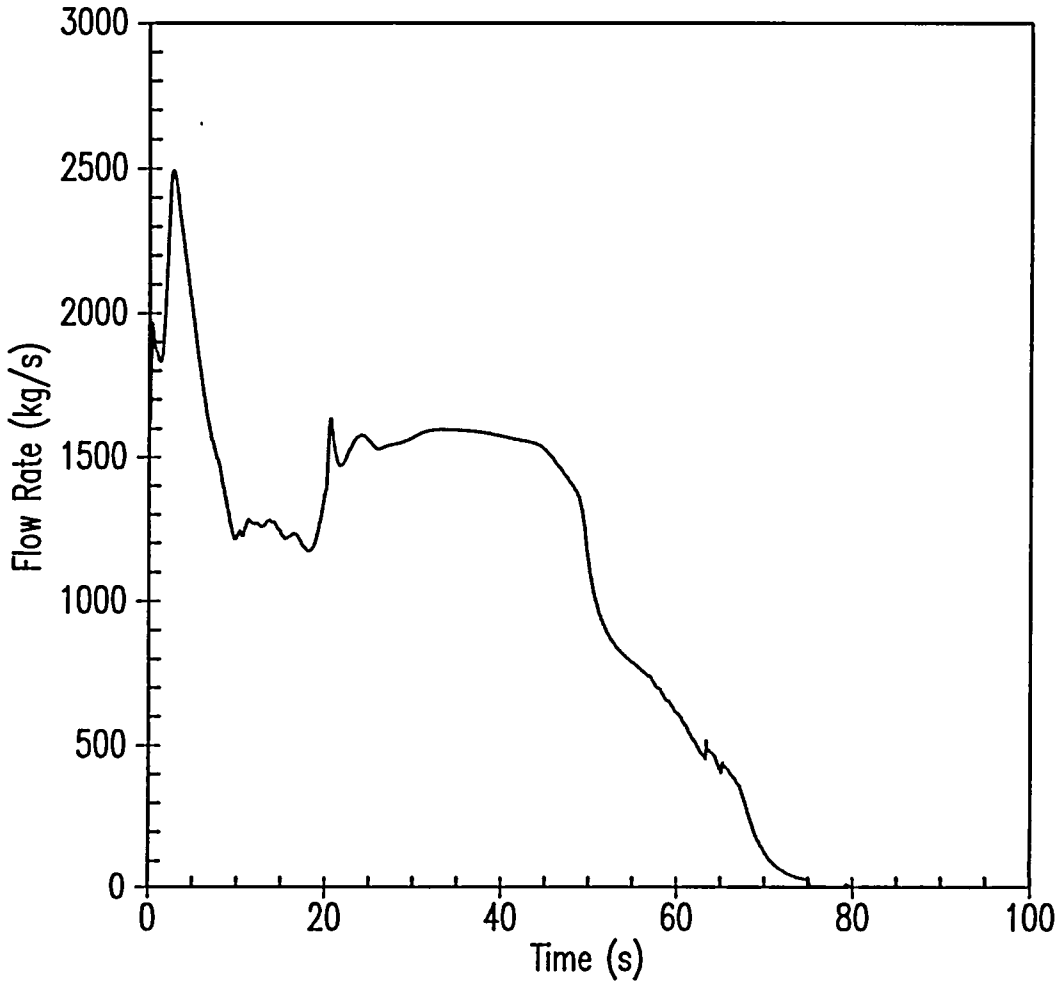

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Figure B.4-2
MSLB Benchmark Phase III Scenario 2:
Broken Loop Cold Leg Temperature vs. Time

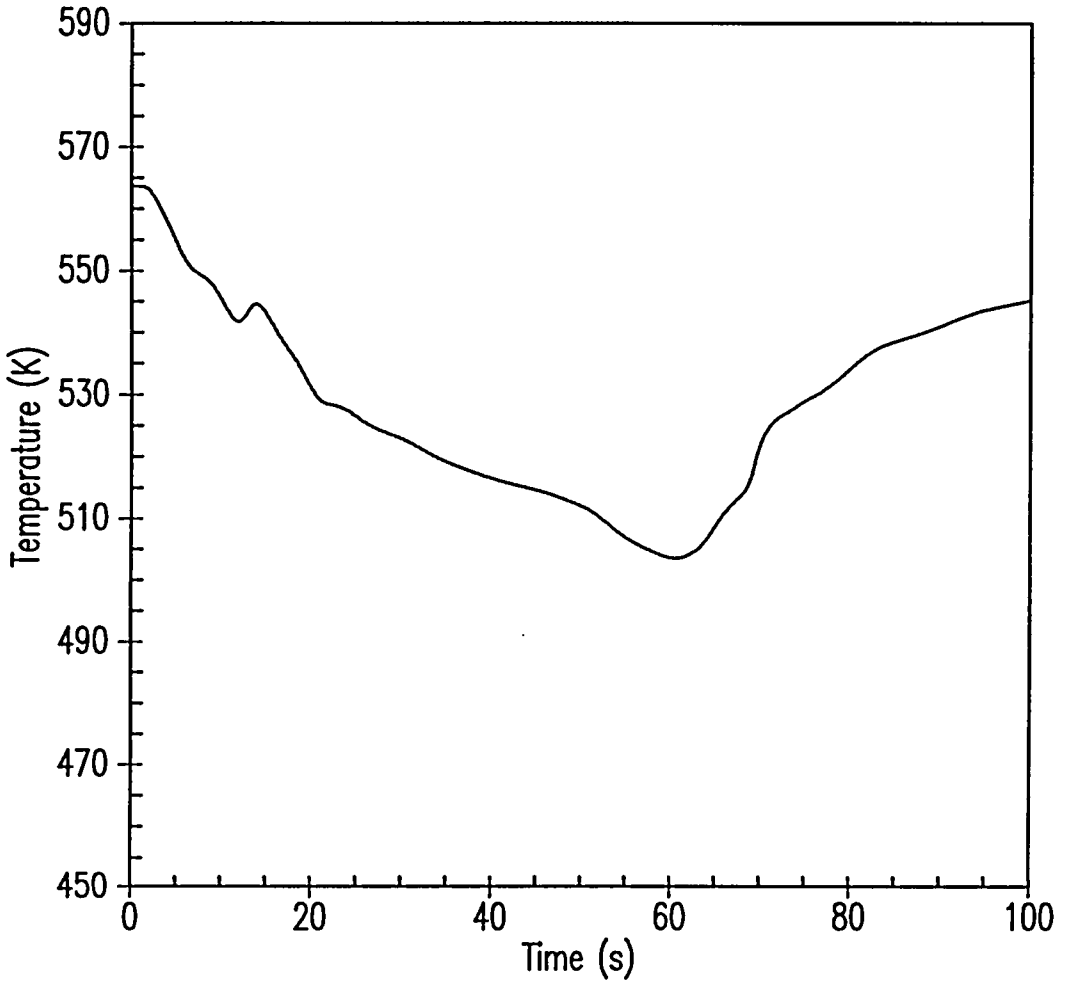

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Figure B.4-3
MSLB Benchmark Phase III Scenario 2:
Intact Loop Cold Leg Temperature vs. Time

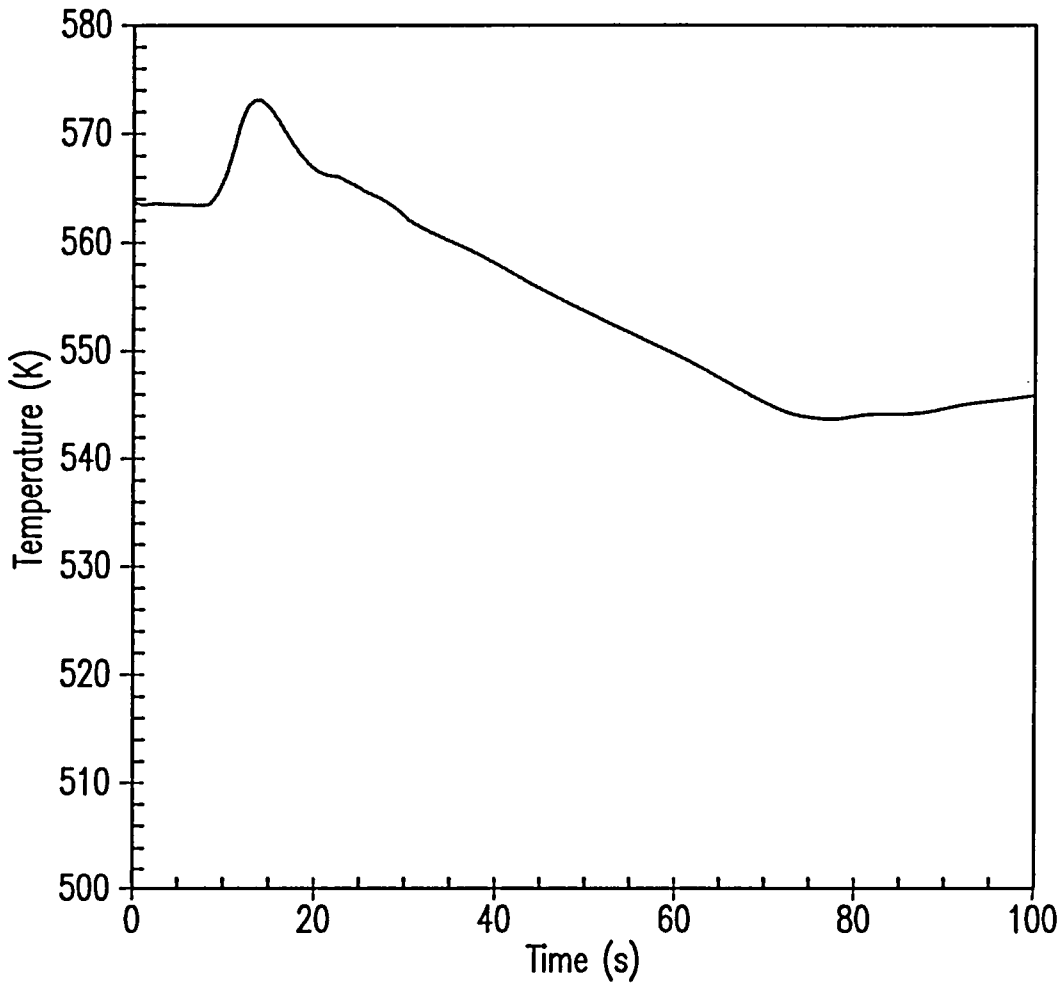

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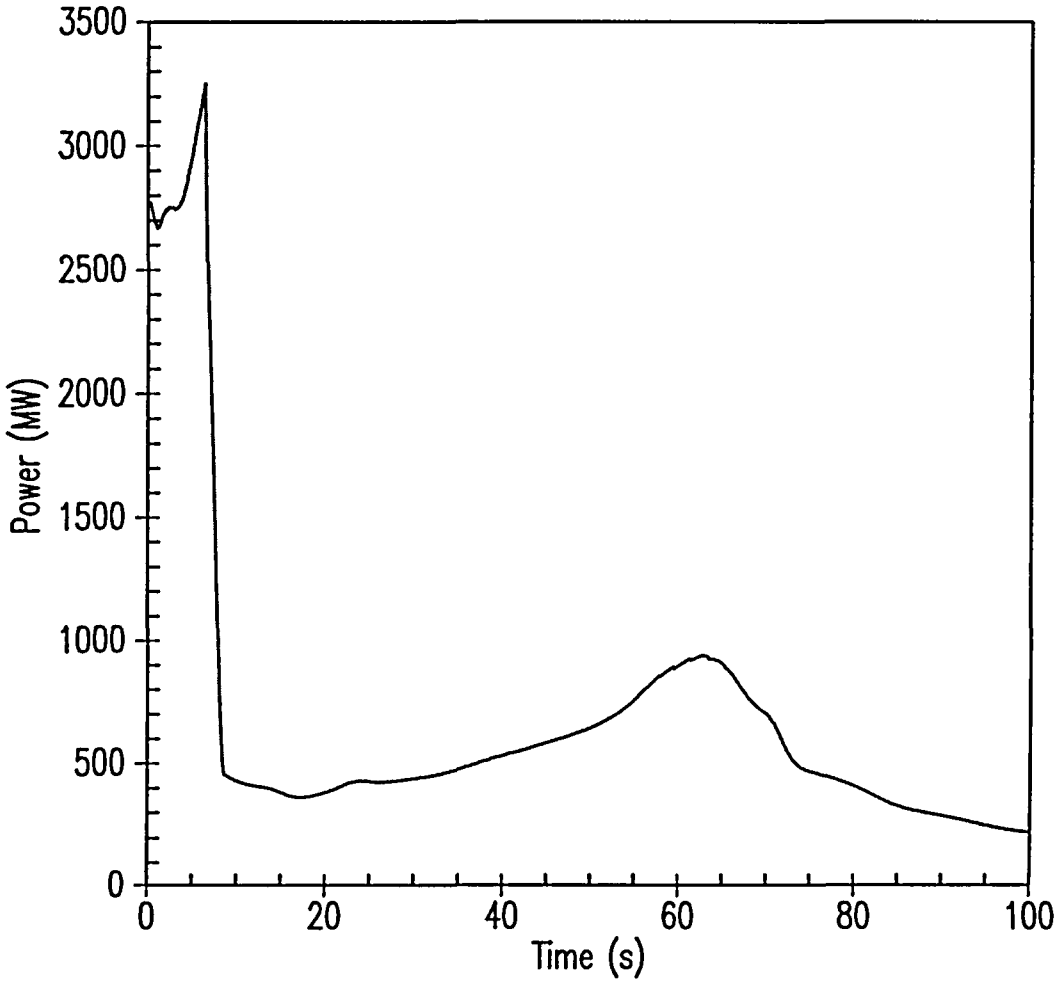
Figure B.4-4
MSLB Benchmark Phase III Scenario 2:
Total Core Power vs. Time



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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

[ ]

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SECTION C

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Our ref: LTR-NRC-04-71

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

December 16, 2004

Subject: "Responses to Request for Additional Information of Topical Report WCAP-16259-P Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis" (Proprietary/Non-Proprietary)

Reference: LTR-TA-04-268 Rev. 2, Attachments B and C

Dear Mr. Wermiel:

Enclosed is a copy of "Responses to Request for Additional Information of Topical Report WCAP-16259-P Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis" (Proprietary/Non-Proprietary). The response to this request for additional information was discussed with the NRC in September, 2004.

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-04-1936 (Non-Proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

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.

Correspondence with respect to this affidavit or Application for Withholding should reference AW-04-1936 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: F. M. Akstulewicz/NRR
A. Attard/NRR
B. J. Benney/NRR
L. M. Feizollahi/NRR

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e-mail: greshaja@westinghouse.com

Our ref: AW-04-1936

December 16, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Responses to Request for Additional Information of Topical Report WCAP-16259-P Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis" (Proprietary)

Reference: Letter from J. A. Gresham to J. S. Wermiel, LTR-NRC-04-71, dated December 16, 2004

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-04-1936 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-04-1936 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

AW-04-1936

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. 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:

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 16th day
of December, 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 rule making 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 constitutes Westinghouse policy and provides the rational basis required.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, 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 in, "Responses to Request for Additional Information of Topical Report WCAP-16259-P Westinghouse Methodology for Application of 3-D Neutronics to Non-LOCA Accident Analysis", for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-04-71) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse 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) Westinghouse and defense ofcan 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.

Further the deponent sayeth not.

Responses to Request for Additional Information of
Topical Report WCAP-16259-P
Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis

NEUTRONICS related RAIs

Section 1.0 Introduction

1. **The 1st and 2nd paragraphs on page 2 require additional discussion regarding applicability of this 3-D methodology. (At the audit).**

Response: The purpose of this report (WCAP-16259-P) is to provide the documentation for NRC review and approval of the Westinghouse Methodology for Application of the 3-D transient Neutronics to Non-LOCA Accident Analysis (RAVE methodology). The methodology is applicable to the events listed in Table 3.6-1 of WCAP-16259-P.

Specifically; Westinghouse is seeking generic approval from the NRC for the use of the RAVE methodology with the linked NRC-approved core neutron kinetics code (SPNOVA (ANC)), NRC-approved core thermal-hydraulics code (VIPRE), and NRC-approved RCS loop thermal-hydraulics code (RETRAN). The generic approval of the RAVE topical should be applicable to any PWR where SPNOVA, VIPRE and RETRAN codes and models are approved for use in compliance with the conditions identified in the NRC SERs. The RAVE methodology is applicable to versions of SPNOVA, VIPRE, and RETRAN that are licensed for plant application and have the appropriate external communications interface.

The RAVE Methodology should be approved for use at all PWRs where SPNOVA, VIPRE and RETRAN are approved for use. Westinghouse has demonstrated that the RAVE methodology can successfully link SPNOVA, VIPRE and RETRAN.

For those plants which have licensed the use of SPNOVA, VIPRE and RETRAN, no additional regulatory action on the external linkage is required; the applicability of the RAVE methodology is addressed by reference to WCAP-16259-P.

If a specific plant has not licensed the use of the codes and models for which RAVE has been approved, then the plant will need to take appropriate licensing action for application of these codes. As a part of the licensing action, any changes to the Chapter 15 analyses would need to be addressed; however, if SPNOVA, VIPRE and RETRAN have been approved for use at

the plant, no specific licensing action should be required to apply the RAVE methodology.

2. **Regarding the assumptions in the last paragraph of page 3, how do you ensure/verify that the "key safety parameters" themselves remain the same for the 3-D methodology and that the "values" of these key parameters remain bounded by the "reference safety analysis"? i.e. for each code, please provide the assumption used and show how these codes still meet imposed conditions and limitation on valid ranges.**

Response: The "key safety parameters" with respect to a reload safety analysis are those parameters which have been found to have a significant influence on the event, and could become changed as a result of a reload. Typically, these are [

] [a,c]. These parameters may vary as a result of the reload core design and with cycle burnup.

Typically, it is expected that the reload core design will be very similar to the previous cycle, and these parameters will be unchanged; however, they are checked for each cycle in accordance with the NRC-approved reload methodology in WCAP-9272-P-A (Reference 15 of WCAP-16259-P).

[

] [a,c]

The 3-D analyses for the reloads will be limited and bounded by the reference analyses provided in the safety analysis reports. This process is the same as that currently applied under WCAP-9272-P-A. For each plant, the RAVE methodology would be used during the initial application to identify and analyze the base or reference case. The key parameters would then be confirmed to remain valid during the reload. Since the overall process philosophy has not changed (analyzing a reference case and checking key parameters during the reload), WCAP-9272-P-A remains valid.

Section 2.0 Generic Models

3. **The 2nd paragraph on page 5 and following, describes a W-3-loop plant as the sample application. Is this considered a limiting plant type for any reason?**

Response: A 3-loop core may not be the most limiting for the transients, but it represents [

] a,c

4. **The 2nd paragraph on the same page refers to an "input multiplier" on the Doppler feedback. Please provide additional information/discussion.**

Response: [

] a,c

For example, a multiplier of [ ] a,c is used to reduce the Doppler feedback for events resulting in a power increase (Rod Withdrawal, RCS cooldown events). The reduced Doppler feedback results in a faster rise in the reactor power and a higher peak power level for a given positive reactivity insertion. A multiplier of [ ] a,c is used to increase the Doppler feedback for events resulting in a power decrease (Loss of Flow, Locked Rotor, RCS heatup events). Due to the negative Doppler power/temperature coefficient, the power reduction causes a positive reactivity insertion. The use of maximum Doppler feedback reduces the rate of the power decrease prior to or after a reactor trip. The use of maximum or minimum Doppler feedback is consistent with the current (point-kinetic) analysis methodology.

5. **How does Westinghouse assure that all the staff conditions and limitations, associated with the approval of all the codes involved in this methodology, have been satisfied and will continue to be satisfied when the pertinent codes are coupled?**

Response: Westinghouse has addressed the SER conditions and limitations on NRC-approved RETRAN, SPNOVA and VIPRE codes involved in the new methodology as summarized in Appendix A of WCAP-16259-P.

The RAVE code provides many warning messages and error checks to help ensure that the code set is being used correctly. For example, the RAVE interface utilizes the [ ] a,c to provide information on whether the core conditions between VIPRE and RETRAN are consistent. The [ ] a,c are checked to be within [ ] a,c between the VIPRE model and RETRAN model, for every time step. If any of these values differ by more than [ ] a,c a warning message is output. The user must then disposition these warning messages. These warning messages are generated if RETRAN and VIPRE are not initialized consistently, such as the codes do not have the same [ ] a,c in RETRAN is significantly different from that in VIPRE. If warning messages on the [ ] a,c are generated for these reasons, the user would be required to rerun the case with the correct initialization. These warning messages can also be seen if a too large time step is used when the core conditions are changing significantly. The user would then be required to rerun the case with a smaller time step over the interval where the warning messages were printed out.

The analysts performing the analyses receive training to understand and correctly address the system messages. Westinghouse maintains training guidelines that assures only qualified analysts perform and verify the analyses being performed.

With respect to the uncertainties, the uncertainties utilized in transient applications of the SPNOVA code are consistent with the uncertainties utilized in the static calculations and are consistent within all the transients. The peaking factor uncertainties and the control rod worth uncertainties are the same as applied in the static analyses. Conservatisms in the reactivity coefficients and transient parameters are used consistently across all the transients, but they are used in conservative direction for each transient. These parameters are also consistent with those defined in the 3D Rod Ejection Report (WCAP-15806-P-A). These conservatisms are summarized in the table below:

[ ] a, c

6. **The 4th paragraph on page 9 alludes to a "potential cycle history factor". Is this based on a calculation/prediction or are the input parameters to the calculation based on the available data from the last cycle of the plant?**

Response: The range of previous cycle length is defined by the operator prior to the safety analysis being performed. [

] a,c

7. **The last paragraph on page 9 discusses the Westinghouse methodology regarding the reload process.**

   a. **Please provide a one or two-page outline of this process, showing the key parameters involved. Discuss how and why they are determined to be valid for the current cycle.**

   Response: The reload evaluation process and key parameters for each event considered in the FSAR are described in the NRC-approved bounding analysis approach for reload in WCAP-9272-P-A (Reference 15 of WCAP-16259-P). The key parameters are then confirmed using static calculations.

   b. **Also, please provide technical justification as to why the topical report, which was approved by the staff in 1985, is verified to still be valid/applicable to analyzing modern cores.**

   Response: The reload evaluation process and methodologies have not been changed. However, the neutronic codes and methods described in the original topical report (WCAP-9272-P-A) have been supplemented by NRC-approved topical reports (WCAP-10965-P-A, WCAP-11596-P-A and WCAP-12394-A for the core Nuclear/Thermal-Hydraulic design, and WCAP-14882-P-A for the RCS loop analysis) as applied to current reload safety evaluations. The reload safety analysis process verifies that the behavior of the core is still bounded by the parameters that were used for the reference safety evaluation.

8. **On page 20, Section 2.6 describes the application of "conservative allowances". It is stated that only the deterministic method will be used in the updated methodology, to determine the necessary uncertainties. Are these uncertainties more conservative when they are obtained in the deterministic manner?**

Response: The "deterministic" method of applying uncertainties refers to the method in which the uncertainty for each of the key analysis parameters is applied simultaneously in the most limiting direction in the analysis. The use of the deterministic method, and the magnitude of the uncertainty allowances used, is the same as in the current (point-kinetics) methodology. The deterministic method is more conservative than the "statistical" approach, in which the total uncertainty is determined by the square root of the sum of the squares (SRSS) of the effect of the individual uncertainty allowances on the analysis results.

Section 3.0 Sample Application of 3-D Methodology

9. On page 27, under section 3.1.5, (b), regarding assumptions used in reactor core calculations, it is stated that BOC HFP equilibrium xenon conditions lead to the "least negative" MTC. Are plants with heavy burnable poison loadings always least negative at BOC?

Response: The most positive MTC may occur later in the cycle than at BOL due to the soluble boron concentration increase with the burnable absorber depletion. Traditionally, the most limiting MTC was at BOL, and many times the wording reflects that traditional approach. The reference bounding case will address the actual Technical Specification limits on MTC.

10. In the same paragraph, the choice of AOs is discussed. Was a search conducted to determine the most limiting AO for these circumstances?

Response: The sensitivity study evaluated the impact of different axial offset preconditions for the transient. [

] a,c

11. On page 28, 1st paragraph the statement is made that a multiplier was used on the Doppler feedback cross-sections, as an adjustment. Please explain further.

Response: See the response to question 4.

In addition, on page 51 it is stated that the same beta (delayed neutron fraction) was used in the 3-D analysis. Why was a more pessimistic value of beta not used, such as the approach used for the Doppler coefficient.

Response: The value of beta used for the 3-D analysis was the same pessimistic value as was used in the current (point-kinetics) analysis. Typically, accidents are not very sensitive to beta (except for the RCCA Withdrawal from Subcritical and RCCA Ejection events); therefore, beta is assumed to be either a maximum or minimum constant bounding value over the entire cycle, which includes a variable amount of conservatism depending on the time in life. The value chosen for this analysis was a [ ] a,c value, and already includes [ ] a,c

12. Figure 3.1-2 on page 37, shows the 3-D results crossing over the point kinetics results. Please explain the phenomena affecting this result.

Response: The 3-D calculations use a detailed VIPRE fuel rod and thermal-hydraulics model for the fuel temperature and moderator density feedback calculations. To be consistent with the steady-state ANC neutronics feedback calculations, the VIPRE fuel temperature model is calibrated to match the design model used with ANC, and the Doppler feedback is based on the calculated fuel temperature vs. time. In contrast to the 3-D method, the point kinetics method uses a relatively simple average core heat transfer model which is calibrated against a conservative design model to obtain either a maximum or a minimum heat transfer model depending on the application. For the Loss of Flow event described in Section 3.1, the point kinetics fuel rod model is based on a [ ] a,c heat transfer model in order to [

] a,c

It should be noted that in both the 3-D and point kinetics methods, the average core heat flux vs. time is not used in the DNBR evaluation. Instead, a separate hot rod calculation is performed with the VIPRE code using the core nuclear power transient as input. For the hot rod calculation, both the 3-D and point kinetic methods assume a conservative [ ] a,c which yields a conservative minimum DNBR.

13. Figure 3.1-4, shows comparisons for the pressurizer pressure. Please explain the phenomena being modeled in 3-D that lead to the large difference.

Response: [

] a,c

14. Figure 3.1-5, page 43, please provide additional discussion

Response: [

] a,c

15. The 1st paragraph on page 49 states that DNBR will be calculated with the WRB-2 correlation. On the same page, the 2nd paragraph from the bottom makes reference to a limiting DNB axial power shape. What is that shape and how is it determined?

Response: [

] a,c The NRC-approved RAOC methodology is described in WCAP-10216-P-A, Rev. 1-A (Reference 16 of WCAP-16259-P).

16. Figure 3.3-2 on page 81 shows no cross-over as in the previous runs. Please explain.

Response: As described in the response to RAI #12, the 3-D calculations use a detailed VIPRE fuel rod and thermal-hydraulics model for the fuel temperature and moderator density feedback calculations, consistent with the ANC design

model, whereas the point kinetics method uses a simpler model which is pessimized to obtain either a maximum or a minimum heat transfer model. For the Locked Rotor-Peak pressure event described in Section 3.3, the point kinetics model used a conservative [

] a,c

17. What are the conservative values used for the Doppler coefficient and the EOC MTC (also page 93, 3rd paragraph) as stated in the 2nd paragraph from the bottom of page 90?

Response: [

] a,c

18. Please discuss Figure 3.4-5.

Response: Figure 3.4-5 is discussed in more detail in Section 3.4.6 of the report (see paragraph 4 on page 94). The figure shows the transient variation in the radial power peaking factor ($F_{\Delta H}$) and the core average axial offset (A.O.) during the event. The purpose of the figure is to show the variation in the $F_{\Delta H}$ and A.O. during the transient, since these are related to the calculated DNBR vs. time. [

] a,c

19. What is the source of oscillations in Figure 3.5-1 and 3.5-2 for the 3-D case at the cross-over point for the point kinetics case?

Response: They are caused by the activation of the accumulators which are actuated when the pressure falls below 600 psia. [

] a,c

THERMAL-HYDRAULIC Related RAIs

1. **Page 6 notes that there are two types of analyses using the VIPRE code. A VIPRE model of the entire core is used to calculate the transient local coolant density and fuel effective temperature for reactivity feedback calculations by SPNOVA. VIPRE is also utilized in a separate calculation to determine the hot rod minimum DNBR vs. time and the fuel cladding temperature vs. time. The hot rod VIPRE model is described in Section 2.4 which references WCAP-14565-P-A for the approved methodology. The total core VIPRE model is not described in detail. Please describe this model including the following information:**

   a. **Noding diagrams applicable to all reactor types for which you seek approval using the methodology of WCAP-16259-P.**

   Response: [

   ] a,c

   b. **Discuss sensitivity studies used to establish the proper noding, both radial and axial. Discuss noding requirements to determine thermal/hydraulic conditions in the vicinity of a stuck out control rod for N-1 analysis. What different noding is required to analyze the condition of control bank A and shutdown banks A and B not tripping to obtain local thermal/hydraulic conditions to transfer to SPNOVA?**

   Response: [

   ] a,c The individual code models are consistent with the SPNOVA and VIPRE topical reports previously approved by the NRC.

   [

   ] a,c

a, c

Figure A: Sample VIPRE Whole Core Nodalization for a Westinghouse-design 3-loop Core

[

] a,c

[

] a,c

c. **In response to Question 5 which was asked during the NRC staff's review of VIPRE as described in WCAP-14565-A it was stated that the void fractions models selected for use in VIPRE over-predict the actual void fraction and this is conservative for calculation of DNBR. In calculating the coolant voiding for input into the neutronics calculations in SPNOVA perhaps it would be conservative to minimize void fraction. Please justify that the void fraction models selected to use in the VIPRE total core model are conservative for the neutronics calculations. Please consider all the transients and accidents for which the coupled code model will be utilized.**

Response: [

] a,c

Additional justification on the void fraction model is provided in Supplement 1 to WCAP-16259-P.

2. **Page 19 describes use of VIPRE to perform calculations of post DNBR fuel performance. In the staff's SER for use of VIPRE by Westinghouse (WCAP-14565), Condition 4 in the conclusion stated that the staff's review did not extend to use of VIPRE for post DNBR calculations. Therefore, justification for use of the code calculations of this type would have to be submitted with each application. Please provide this supporting information for the post DNBR calculations described in Section 2.4.2.**

Response: Condition 4 of the SER on WCAP-14565 states that "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."

Westinghouse VIPRE post-CHF (or post-DNBR) applications are limited to FACTRAN replacement in non-LOCA Condition IV events. The VIPRE model retains the same conservatism as the current design model using the FACTRAN code. The conservative modeling assumptions in WCAP-7908-A (Ref. 21 of WCAP-16259-P) remain unchanged for the VIPRE post-CHF applications. The VIPRE applications are in compliance with the conditions in the FACTRAN SER. Therefore, the VIPRE code with Westinghouse modeling method is justified to replace the FACTRAN code for non-LOCA post-CHF transient analysis.

The VIPRE conservative post-CHF modeling assumptions from FACTRAN are summarized in the table below.

[ ] a, c

[ ] a, c

In order to demonstrate that the VIPRE post-CHF model is equivalent to the FACTRAN code, a Locked Rotor hot spot fuel temperature comparison between FACTRAN and VIPRE is presented for a 2-loop Westinghouse plant. The reactor core and fuel parameters for the locked rotor (LR) calculations are provided in the table below.

Parameter Description	Value Used for LR Calculation
Fuel Rod OD, inches	[ ] a,c
Pellet OD, inches	[ ] a,c
Clad Thickness, inches	[ ] a,c
Densified Heated Length, inches	[ ] a,c
Initial Core Power, MWt	[ ] a,c
Initial Core Pressure, psia	[ ] a,c
Initial Core Inlet Enthalpy, BTU/lbm	[ ] a,c
Initial Core Effective Flow, ft/s	[ ] a,c
Hot Spot Power Peaking Factor F_Q	[ ] a,c

The results of VIPRE and FACTRAN comparison are consistent with the results presented for the 3- and 4-loop plants in WCAP-14565-P-A. Both codes predict similar temperature changes during the transient as depicted in Figure 1. The difference in the maximum clad temperature is about [ ] a,c. The VIPRE temperature values are [ ] a,c the FACTRAN values because the VIPRE calculation accounted for [ ] a,c. The FACTRAN calculations incorporated additional conservatism to account for [ ] a,c in calculating the film boiling heat transfer coefficients. The third case presented in Figure 1 corresponds to a FACTRAN case [ ] a,c. This FACTRAN case also predicts similar temperature changes during the transient with [ ] a,c maximum cladding temperature. The difference is about [ ] a,c.

These results of the code comparison confirm that the VIPRE post-CHF heat transfer model is consistent with the FACTRAN code. The difference in the maximum cladding temperature is due to the simplified assumptions used in the FACTRAN film boiling heat transfer coefficient calculations, as explained above.

a, c

Figure 1. Comparison of VIPRE and FACTRAN Peak Cladding Temperature vs. Time for a 2-Loop Plant Locked Rotor Hot Spot Analysis

a, c

Figure 2. Comparison of VIPRE and FACTRAN Film Boiling Heat Transfer Coefficient vs. Time for a 2-Loop Plant Locked Rotor Hot Spot Analysis

a. **VIPRE may not model all the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. If the code were to be run beyond the conditions for which physical changes in the fuel might occur, the results would no longer be valid. Please provide the physical limits beyond which VIPRE results would no longer be acceptable. What checks are made within the code to ensure that it is not used beyond its range?**

Response: The Westinghouse version of VIPRE is used in place of the USNRC-approved FACTRAN code for the hot rod analysis. The VIPRE results are used to confirm that the fuel acceptance criteria [

] [a,c] are met. Since the fuel acceptance criteria are all within the conditions of applicability for the VIPRE code, the VIPRE code is not used beyond its validity range.

The VIPRE code will not be used for LOCA analysis.

b. **Page 19 of WCAP-16529 states that the dynamic gap conductance model in VIPRE will be used to account for changes in the fuel dimensions and fill gas pressure. Section 3.3.3 of WCAP-14565-P-A indicates that a constant gap conductance will be used or the gap conductance will be adjusted to a high value to simulate clad collapse onto the fuel pellet similar to as is done using FACTRAN. Please justify that Westinghouse will utilize the dynamic gap conductance model in VIPRE in such a manner so that conservative results will be obtained.**

Response: [

] [a,c]

c. **Discuss fuel rod failure models to be used with the VIPRE hot rod model. Provide and justify as conservative assumptions for cladding collapse,**

overheating of cladding, overheating of fuel pellets, excessive fuel enthalpy, pellet cladding interaction and bursting. Compare the assumptions you will use to the guidance of SRP 4.2. Will cladding failure be assumed immediately after DNBR limits are exceeded? If not please provide appropriated justification.

Response: The following fuel rod failures in SRP 4.2 are addressed through design analysis using the fuel performance code such as PAD: hydriding, cladding collapse, fretting, overheating of fuel pellets, and pellet/clad interaction. The VIPRE hot rod model is not used for such analysis.

The following fuel rod failures are part of the LOCA analysis for which the VIPRE code is not applicable: bursting and mechanical fracturing.

Overheating of cladding is protected by the DNB design criterion using NRC-approved DNB correlations in the VIPRE code. Cladding failure is assumed immediately if the DNBR limit is exceeded.

The VIPRE code is used for predicting fuel and clad temperatures and fuel enthalpy for a Condition IV reactivity initiated accident (RIA). The fuel failure criteria are discussed in WCAP-15806-P-A (Reference 7 of WCAP-16259-P).

d. **Following transients and accidents which are calculated to produce fuel damage, fuel coolability must be demonstrated. Acceptance criteria for fuel coolability are described in SRP 4.2. Please address Westinghouse assumptions for fuel coolability as a result of cladding embrittlement, violent expulsion of fuel, generalized cladding melting, and fuel rod ballooning. Does Westinghouse propose to use fuel coolability acceptance criteria different from those of SRP 4.2? If so, please provide appropriate justification.**

Response: The submittal of WCAP-16259-P does not result in fuel coolability acceptance criteria different from SRP 4.2. There is also no change to the currently NRC-approved Westinghouse evaluation methodology for addressing those criteria in safety analysis as described in WCAP-12488-A and WCAP-15806-P-A, except that the FACTRAN code is replaced by the VIPRE code.

Reference:

2-1 Davidson, S. L., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994.

3. **Page 27 states that for the initial core condition using the new methodology that an initial axial power distribution of +10% was used since that represents the most positive limit of APD at the 3-loop plant for which the analysis was performed. On page 29 it is stated that using the current methodology which uses point kinetics that an axial offset (AO) of approximately +10% is assumed. Page 30 states that the difference in initial DNBR between the two methods (~1.8 vs. ~2.2) in Figure 3.2-6 is caused by the conservative reference axial power shape in the current method. Since the initial AO is the same for the two methods, DNBR is apparently reduced in the proposed method by some other cause. Please identify the initial DNBR reduction in the proposed methodology and justify the validity.**

Response: [

] a,c

4. **Page A-4 states that [ ] a,c axial nodes are used to describe the core in the RETRAN model. Figures 2.3-1 and 2.3-2 show but [ ] a,c axial nodes in the RETRAN models.**

   a. **Please provide the corrected figures.**

   Response: Corrected figures are provided below.

   b. **What sensitivity studies were performed to ensure that [ ] a,c nodes in the RETRAN model were adequate?**

   Response: The WCAP-14882-P-A RETRAN core model with [ ] a,c has already been approved by the NRC. The functionality of the RETRAN core in terms of the thermal-hydraulic calculations is not changed in the 3-D neutron kinetics methodology. [

] a,c

The RAVE code provides many warning messages and error checks to help ensure that the code set is being used correctly. For example, the RAVE interface utilizes the [ ]a,c to provide information on whether the core conditions between VIPRE and RETRAN are consistent. The [

]a,c are checked to be within [ ]a,c between the VIPRE model and RETRAN model, for every time step. If any of these values differ by more than [ ]a,c a warning message is output. The user must then disposition these warning messages. These warning messages are generated if RETRAN and VIPRE are not initialized consistently, such as the codes do not have the same [

]a,c in RETRAN is significantly different from that in VIPRE. If warning messages on the [ ]a,c are generated for these reasons, the user would be required to rerun the case with the correct initialization. These warning messages can also be seen if a too large time step is used when the core conditions are changing significantly. The user would then be required to rerun the case with a smaller time step over the interval where the warning messages were printed out.

c. **It is stated that the increased core noding was done to facilitate data transfer between the VIPRE core model to the RETRAN core model. Compare the axial core noding in VIPRE model to that of the RETRAN core model. If the noding is different, discuss how the pressure, temperature and flow data from RETRAN is manipulated to accommodate the different noding between the two computer codes.**

Response: [

]a,c

[

]a,c

[ ]a, c

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

a, c

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

5. Page A-5 describes 4 methods by which 2-, 3- or 4-channel conditions of the RETRAN model are applied to the inlet for each of the SPNOVA/VIPRE model core channels. These are 1) the average model, 2) the core sector model, 3) the currently licensed model, and 4) the fine mesh model.

a. Please list the postulated transients and accident for which each of these models will be used and justify that each usage is conservative for the analysis being performed.

Response: [

] a,c

[

] a,c

[

] a,c

b. With the exception of the fine mesh mixing model, the mixing models discussed on page A-5 have previously been reviewed by the NRC staff. So that the staff may review the fine mesh mixing model please provide the details of this model and discuss how the model had been validated by comparison to applicable experimental data.

Response: [

]a,c

a, c

[

]a,c

[

]a,c See Supplement 1 to WCAP-16259-P.

6. **Appendix B describes results from Westinghouse participation in the OECD MSLB benchmark. The benchmark was structured into 3 phases. In Phase I, the ability of the RETRAN code to model the reactor system performance was investigated. Phase II investigated the ability of SPNOVA/VIPRE to model the reactor core. Phase III investigated the ability of the three combined codes to analyze the entire transient. The results for Phase I are reported to be in excellent agreement with the other participants however, the results are not shown since the conclusions would be common to Phase III. For Phase II, the results are shown and graphs are provided showing the comparison with the results of other participants. For Phase III, graphs of the results are provided and the results are stated to be in excellent agreement with those from the other participants. No comparisons are provided, however. Please provide the results**

from the other benchmark participants for Figures B.4-1 to B.4-4 in a manner similar to that which was done for Phase II. Discuss the reasons for any disagreements between the Westinghouse results and those of the other participants.

Response: Revised Figures B.4-1 to B.4-4 are provided. The revised figures include a comparison between the Westinghouse results and the average benchmark participants solution in a manner similar to that which was done for Phase II. Appendix B.4 includes a discussion of the differences between the Westinghouse results and those of the other participants. This discussion remains valid, as confirmed by the provided revised figures.

The deviation in power from the average benchmark results in the final part of the transient is directly related to the deviation in Broken Loop Cold Leg temperature (where temperature is high, the power is low and vice-versa). When the power drops below the standard deviation range, the temperature is deviating high out of the standard deviation range, which follows the SG dryout. The Westinghouse model dryout occurs earlier than the average of the benchmark participants due to the over predicted entrainment in the early part of the transient.

7. **Table C.5-1 provides the results of a sensitivity study for main steam line break from hot zero power. The case of loss of offsite power (LOOP) is shown not to challenge minimum DNBR limits. With the current point kinetics model, main steam line break and LOOP is calculated to produce return to power which at low flow could lead to the occurrence of DNB. Please provide a comparison of the results from the current model with those from the proposed model showing reactor system pressure, flow, reactivity, and core power. Provide a discussion for the cause for the difference in results.**

Response: An evaluation of the main steam line break with loss of offsite power (LOOP) using the current method is described in Section 3.1.1.14 of WCAP-9226-P-A (Ref. 7-1, below). The evaluation concluded that the LOOP case [ ] [a,c] The results of the sensitivity study for the LOOP case in Table C.5-1 of WCAP-16259-P are consistent with the conclusion in WCAP-9226-P-A. This is not a change from the current methodology.

Reference:

7-1 Scherder W. J. and McHugh C. J. (Editors), "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226-P-A Revision 1, February 1998.

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

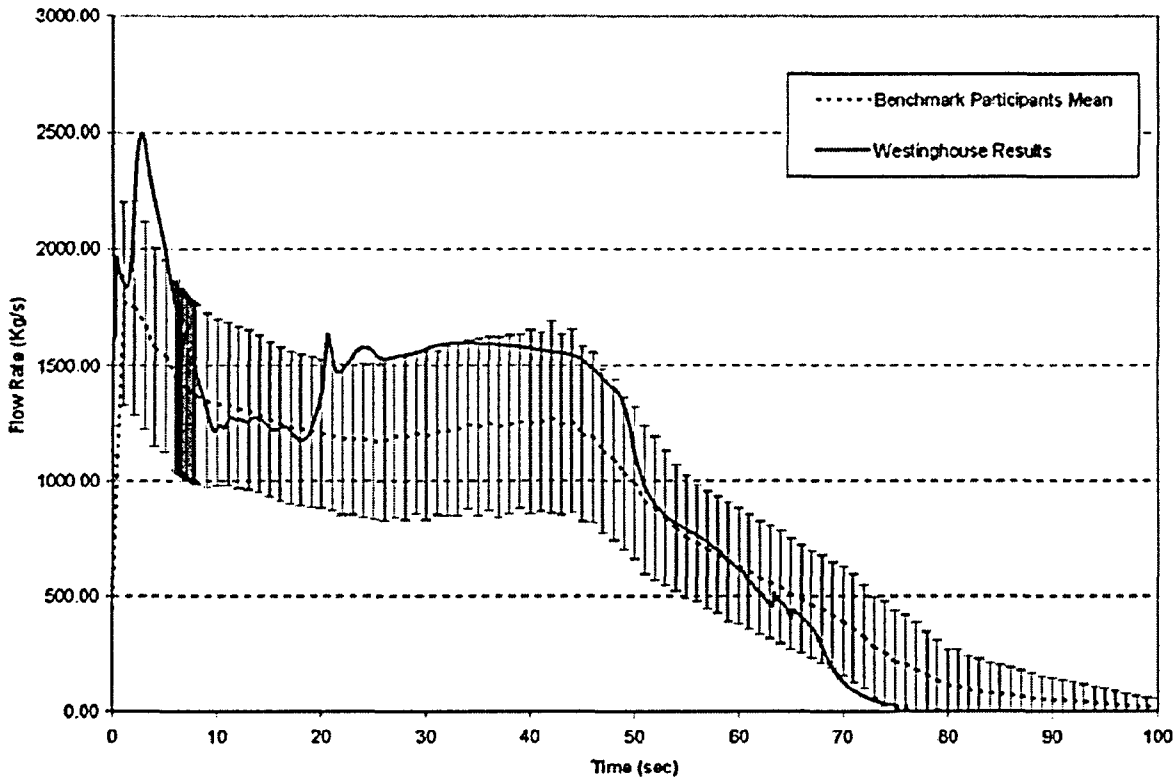


Figure B.4-2 MSLB Benchmark Phase III Scenario 2: Broken Loop Cold Leg Temperature vs. Time

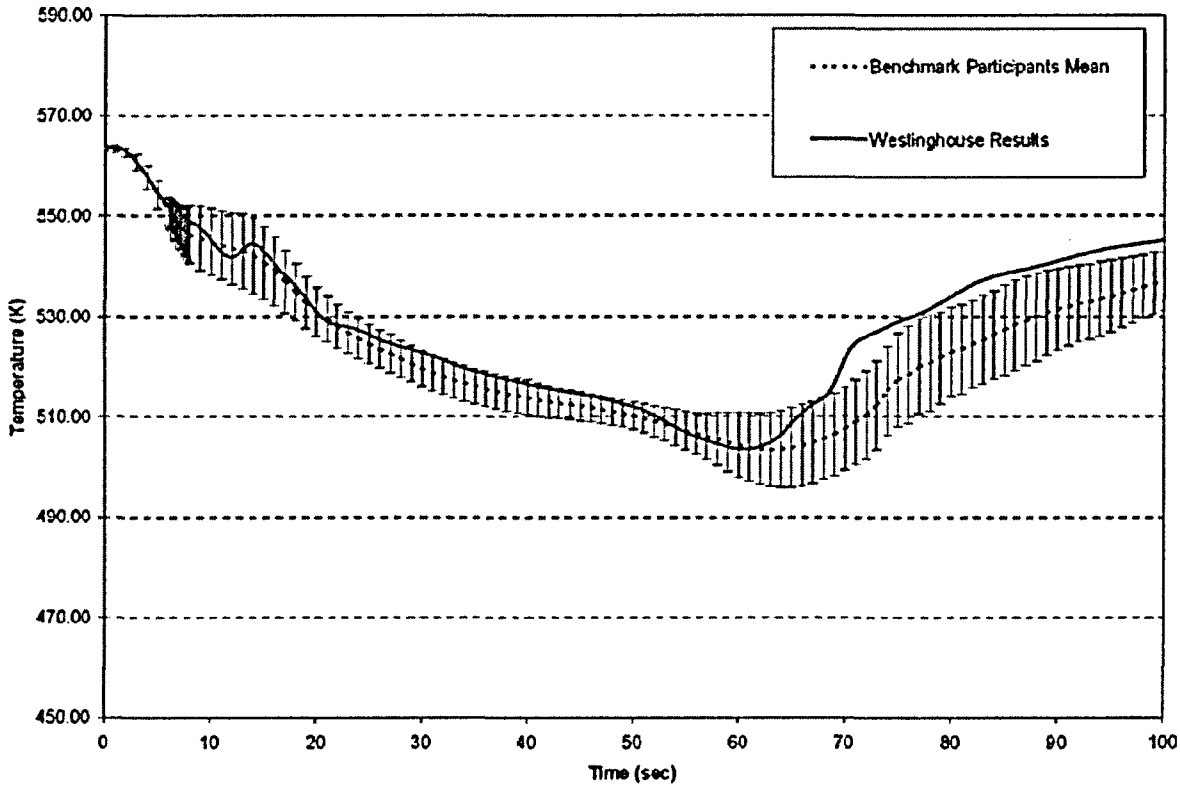


Figure B.4-3 MSLB Benchmark Phase III Scenario 2: Intact Loop Cold Leg Temperature vs. Time

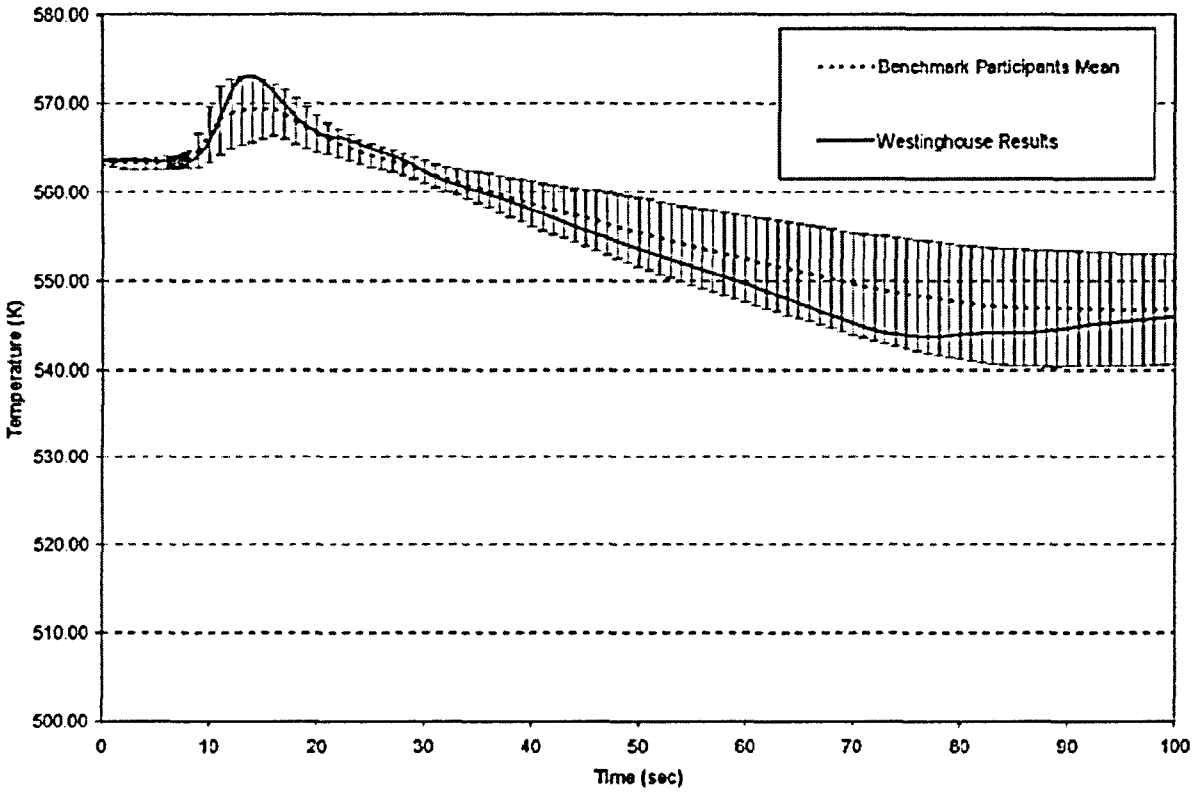
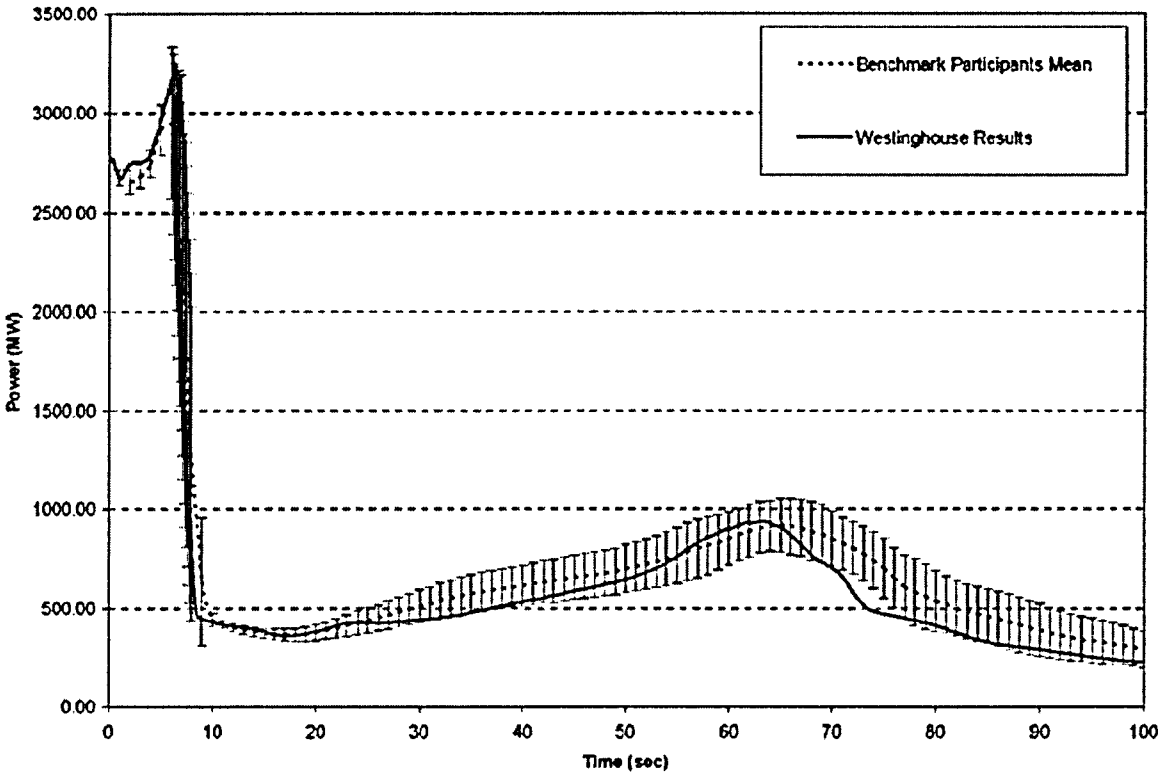


Figure B.4-4 MSLB Benchmark Phase III Scenario 2: Total Core Power vs. Time



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Supplement 1 to WCAP-16259

This is a supplement to WCAP-16259-P. This supplement includes information on the following:

S-1 Fine Mesh Mixing Model

With the implementation of a 3D core model for the evaluation of the event transients, the core inlet mixing model was updated to provide a consistent, but more detailed, distribution compared to the existing model which utilizes a few uniform inlet zones.

S-1.1 Model Description

The fine mesh model is designed to permit much more flexibility with respect to variable loop inlet temperatures and flow rates. It provides a more realistic inlet temperature distribution across the core. It is based on a smooth functional relationship which locally averages the loop enthalpies based on the inverse of the effective distance from the source. [

] a,c

[ ] a, c

The total core mass flow and enthalpy are preserved. This provides a smooth gradient across the core with the core periphery being near the associated inlet enthalpy and the core center being a mass flow averaged enthalpy. [

] a,c

S-1.2 Model Qualification

The core inlet enthalpy distribution function was fit to data from the IPP 1/7 scale model tests for Westinghouse standard 4, 3, and 2-loop core configurations and was evaluated for consistency with the coarse zone model used previously.

The comparison to the IPP 1/7 scale model are shown in figures S-1.1 through S-1.3. These correspond to flows through 4, 3 and 2 loops. In the reduced loop cases, the inactive loops were valved out of operation. The focus of the tests was to determine the mixing factors – the fraction of the cold loop flow – across the lower core support plate and in the loop outlets. The tests were run several times, and the measured data presented

are the maximum mixing factor, the minimum mixing factor and the average mixing factor in the instrumented locations. The numbers outside of the core on the diagonals are the mixing fractions for the loop outlets. The bottom data item in each set is the mixing factor calculated by the algorithm. As one can see, there is considerable variability in the measured mixing factors, but the fine mesh mixing algorithm provides a reasonable distribution which agrees well with the measured data.

Previous methods utilized a coarse mixing distribution with six to eight regions across the core inlet. It is more consistent with the new methodology which uses explicit assembly-wise thermal-hydraulic and nuclear calculations to have a more refined inlet temperature distribution. The new fine mesh model was developed to be consistent with the coarse model used previously, but to provide the greater detail for the inlet mixing. A comparison of the inlet mixing distributions and coarse region values are provided in Figures S-1.4 through S-1.6 for Westinghouse 4, 3 and 2-loop cores. In these figures the coarse mesh regions are identified by the dashed lines separating the assemblies. The colored contours are associated with the fine mesh mixing factors.

S-1.3 Conclusion

The comparisons demonstrate that the fine mesh mixing factors are consistent with the measured mixing factors across the core inlet, and they are also in good agreement with the coarse region data used previously.

S-1.4 References

S.1.1 Hollingsworth, S. D., et al., *Reactor Core Response to Excessive Secondary Steam Releases*, WCAP-9226-P-A, Revision 1 (Proprietary), February 1998.

Figure S-1.1 Mixing Fraction Comparison for IPP 4-Loop Data

a, c

Figure S-1.2 Mixing Factor Comparison for 3-Loop Data

a, c

Figure S-1.3 Mixing Factor Comparison for 2-Loop Data

a, c

Figure S-1.4 Comparison of Coarse Region to Fine Mesh Mixing – 4 Loop Core

a, c

Figure S-1.5 Comparison of Coarse Region to Fine Mesh Mixing – 3 Loop Core

a, c

Figure S-1.6 Comparison of Coarse Region to Fine Mesh Mixing – 2 Loop Core

a, c

S-2 Fuel Temperature Calibration

The thermal-hydraulic model used in the RAVE method is based on the use of the NRC-approved VIPRE code. As described in Sections 2.2.2 and 2.4 of WCAP-16259-P, the VIPRE code is used in two ways in the RAVE methodology:

1) Section 2.2.2 describes the VIPRE fuel rod model used in the reactivity feedback model calculations. In this model, a fuel rod is represented in each SPNOVA neutronics node to calculate the fuel temperature and local water density for the nuclear feedback calculations. [

   ] [a,c] This calculation is performed simultaneously with the neutronics and RCS loop model calculations.

2) Section 2.4 describes the "hot" fuel rod model in which either a single fuel rod (for peak fuel temperature/enthalpy calculations) or a [

   ] [a,c] is represented. The "hot" rod model uses the peak nuclear power reconstructed by SPNOVA in each neutronics node to calculate the hot-spot peak fuel/clad temperature, peak fuel enthalpy, or the minimum DNBR. This calculation is currently performed after the neutronics/RCS loop model calculations, using stored values of the local 3-D peak nuclear power vs. time.

Since the VIPRE code does not model the effects of burnup on the fuel rod (e.g. pellet swelling, clad shrinkage, increased internal gas pressure), it is not used by itself to predict the fuel temperature as a function of burnup. Instead, the fuel temperature vs. power and burnup is calculated by a fuel rod design model, and the VIPRE fuel rod model is then "calibrated" against the fuel rod temperatures calculated by the design model over the entire range of power and burnup needed to represent the fuel (typically 0-400% power and 0-80,000 MWD/MTU burnup range). During the temperature calibration process a separate VIPRE fuel rod type is used for each burnup and this rod type is applicable for all power levels. The initial gap size and fuel thermal conductivity are adjusted and the resulting VIPRE temperatures (fuel centerline, average, surface and cladding average temperatures) are compared against the temperatures from the design model at all power levels. Sample temperature calibration results for 0 MWD/MTU and 24,000 MWD/MTU are presented in Figures S-2.1 and S-2.2.

[

]a,c

[ ]a, c

Figure S-2.1. VIPRE Temperature Comparison against Design Model (0 BU)

a, c

Figure S-2.2. VIPRE Temperature Comparison against Design Model (24K BU)

a, c

Figure S-2.3. Sample Layout of a 3-Loop Plant Full Core (eighth core symmetric

a, c

Figure S-2.4. Eighth Core Layout of a 3-Loop Plant

a, c

Figure S-2.5. 3-Loop Plant (BOC) 1/8 Core Layout and VIPRE Rod Types

a, c

Figure S-2.6. 3-Loop Plant (EOC) 1/8 Core Layout and VIPRE Rod Types

S-3 Void Model for Whole Core Feedback Calculation

In calculating the coolant conditions for input to the neutronic feedback calculations in SPNOVA, the following VIPRE void model is selected:

1) No subcooled void model is used in the whole core reactivity feedback calculation. However, the subcooled void model discussed in WCAP-14565-P-A is continuously used for the hot rod DNBR calculation.

2) A homogeneous void model is used under the bulk boiling conditions. The VIPRE code qualification (Ref. S.3.1) shows that the void fractions predicted with the homogeneous model were in good agreement with the high pressure (1565 to 1991 psia approximately) void data with the fluid quality less than 15%.

S-3.1 Void Model Sensitivity Study

A RAVE sensitivity study was performed with different void models to evaluate effects of bulk boiling on the reactivity feedback and DNBR during a locked rotor and the HZP steamline break accidents. Because of severity of those accidents (both of them are Condition IV events), the hot assembly and the surrounding assemblies are more likely to be in bulk boiling conditions. The plant design parameters and initial core conditions are summarized in Table S-3.1.

Table S-3.1
Plant Design Parameters and Conditions for RAVE Sensitivity Study

Parameter Description	Value Used for LR Calculation
Fuel Rod OD, inches	[ ] a,c
Pellet OD, inches	[ ] a,c
Clad Thickness, inches	[ ] a,c
Densified Heated Length, inches	[ ] a,c
Plant Configuration	3-Loop
Nominal Core Power, MWt	[ ] a,c
Nominal Core Pressure, psia	[ ] a,c
Nominal Core Inlet Temperature, °F	[ ] a,c
Design Core Effective Flow, ft/s	[ ] a,c

* [ ] a,c

For the purpose of sensitivity study, the following bulk boiling void models were used in the RAVE calculations:

1) VIPRE homogeneous void model (base case),

2) VIPRE slip model with a slip ratio of 2.

The transient responses of the locked rotor and the steamline break accidents using the homogeneous void model are described in Sections 3.2 and 3.5, respectively of WCAP-16259-P. The DNBR results from the different void models are shown in Table S-3.2.
[

] a,c

Table S-3.2
DNBR Results of Void Model Sensitivity Study

Parameter Description	Homogeneous Void Model (Base Case)	Slip Model (Slip Ratio = 2)
Locked Rotor Event:		
DNB Limiting Time Step, seconds	[ ] a,c	[ ] a,c
Minimum DNBR (WRB-2)	[ ] a,c	[ ] a,c
% DNBR Change	--	[ ] a,c
HZP Steamline Break Event (1 %Δρ Shutdown Margin*):		
DNB Limiting Time, seconds	[ ] a,c	[ ] a,c
Minimum DNBR (W-3)	[ ] a,c	[ ] a,c
% DNBR Change	--	[ ] a,c

*The base case of the HZP steamline break event is the same as Case 1 in Table C.5-1 of WCAP-16259-P.

S-3.2 Reference:

S.3.1 Stewart, C. W. et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, Volume 4: Applications," EPRI, April 1987, pp. 3-57 through 3-65.

S-4 Post-CHF Modeling for Whole Core Feedback Calculation

The VIPRE heat transfer correlations for the whole core feedback calculation consist of the Dittus-Boelter single phase heat transfer correlation and the Thom nucleate boiling heat transfer correlation. In order to minimize the Doppler feedback, heat transfer is limited to single phase and nucleate boiling regimes without post-DNB heat transfer in the VIPRE whole core model. A sensitivity study was performed for the locked rotor accident with the full boiling curve by adding the Bishop-Sandberg-Tong correlation (Ref. S.4.1) for the post-DNB transition and film boiling regimes. The locked rotor initial condition is similar to those in Table S-3.1 with an assumed 120% power uprate and the hot assemblies assumed to be in DNB during the transient. The DNBR results of the sensitivity study are summarized in Table S-4.1. The sensitivity study shows that the DNBR results are relatively insensitive to the change of post-DNB heat transfer modeling for the whole core feedback calculation.

Table S-4.1
DNBR Results of Post-DNB Heat Transfer Sensitivity Study

Parameter Description	No Post-DNB Heat Transfer (Base Case)	Post-DNB Heat Transfer Using Bishop-Sandberg-Tong Correlation
Locked Rotor Event:		
DNB Limiting Time, seconds	[ ]a,c	[ ]a,c
Minimum DNBR (WRB-2)	[ ]a,c	[ ]a,c
% DNBR Change	--	[ ]a,c

S-4.1 References

S.4.1 A.A. Bishop, et al., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME-65-HT-31, 1965.

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SECTION D

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Our ref: LTR-NRC-05-17

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

March 22, 2005

Subject: Responses to NRC Request for Additional Information on WCAP-16259-P, Rev. 0, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis" (Proprietary/Non-Proprietary) dated March 2005, TAC No. MC3036

Dear Mr. Wermiel:

Enclosed is the Proprietary/Non-Proprietary copy of Responses to NRC Request for Additional Information on WCAP-16259-P, Rev. 0, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis" dated March 2005, TAC No. MC3036.

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-05-1971 (Non-Proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

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.

Correspondence with respect to this affidavit or Application for Withholding should reference AW-05-1971 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: F. M. Akstulewicz/NRR
A. Attard/NRR
B. J. Benney/NRR
L. M. Feizollahi/NRR
E. Kendrick/NRR

Westinghouse

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Our ref: AW-05-1971

March 22, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Responses to NRC Request for Additional Information on WCAP-16259-P, Rev. 0, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis" (Proprietary) dated March 2005, TAC No. MC3036

Reference: Letter from J. A. Gresham to J. S. Wermiel, LTR-NRC-05-17, dated March 22, 2005

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-05-1971 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-05-1971 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

AW-05-1971

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. 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:

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 22nd day
of March, 2005

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 rule making 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 constitutes Westinghouse policy and provides the rational basis required.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, 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 in, Responses to NRC Request for Additional Information on WCAP-16259-P, Rev. 0, "Westinghouse Methodology for Application of 3-D Neutronics to Non-LOCA Accident Analysis" (Proprietary) dated March 2005, TAC No. MC3036, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-05-17) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse 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) Westinghouse can 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.

Further the deponent sayeth not.

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Responses to NRC Request for Additional Information on WCAP-16259-P, Rev. 0, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis"

TAC No. MC3036

March 2005

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

As discussed with the staff in a teleconference on February 22, 2005, clarification to the information on Westinghouse response to RAI on WCAP-16259-P (Ref. A-1) is provided below.

Additional Clarification on selection of the limiting Beta (delayed neutron fraction) (In addition to response to Question 7 of Neutronics related RAIs)

In current accident analysis methods, a limiting value of Beta is used in the analysis codes to ensure a conservative core response in the transient. As an example, for power increase events such as an excessive load increase, control rod withdrawal or rod ejection accident, a small value is typically limiting since it enhances the rate of power increase. For power decrease events, such as a loss of flow or locked rotor accidents, a large value is chosen since it results in a slower power decrease upon the reactor trip. The value of beta varies with fuel burnup, the maximum calculated value occurring at BOC (beginning of cycle), and the minimum calculated value occurring at EOC (end of cycle). The "bounding" values of beta typically used in accident analysis are therefore the maximum value at BOC plus an uncertainty allowance, and the minimum value at EOC minus an uncertainty allowance.

Since most accidents are not very sensitive to Beta, conservative bounding values are typically used for these events, irrespective of the time in life that the accident analysis represents. The limiting direction (maximum or minimum) is determined from a sensitivity study. For a rapid power increase accident, which is very sensitive to minimum values of Beta, the BOC event is typically analyzed with a BOC Beta minus an uncertainty allowance, while the EOC event is analyzed with an EOC beta minus an uncertainty allowance. For the accidents presented in WCAP-16259-P, none of the accidents is very sensitive to Beta, thus conservative bounding values of Beta over the cycle were used rather than specific BOC or EOC values with uncertainties. The sensitivity to Beta, and the selection of the maximum or minimum bounding value, was verified for each accident by the sensitivity study presented in Appendix C.

Additional Clarification on RAVE Methodology Implementation (In addition to response to Question 7 of Neutronics related RAIs)

The RAVE methodology will be implemented in accordance with the Westinghouse Quality Management System (QMS), which has been reviewed and approved by the US NRC. The Westinghouse QMS has been developed to comply with regulatory, industry, and customer quality requirements. The QMS describes the Westinghouse commitments to the quality assurance requirements of ISO 9001; ISO 9000-3; 10CFR50, Appendix B; ASME NQA-1; and IAEA 50-C-QA.

The QMS is structured around interlinked processes that provide the necessary implementation controls to ensure customer and regulatory requirements are met and continual process improvement. It provides the basis for policies and procedures that implement a comprehensive quality management system. These processes are those that define activities that are directly necessary to create the item or service, and those that provide the supporting infrastructure to enable the direct processes to operate under the required controls, and continually improve.

The QMS provides the basis for the implementation of programs such as RAVE, including work instructions. Work instructions provide detailed steps to conduct specific work activities. Work instructions are prepared as needed to supplement procedure requirements and to ensure that critical work scopes are carried out in a consistent manner.

In addition, the analysts performing the analyses receive training to understand and correctly implement new methodologies. Westinghouse maintains training guidelines that assures only qualified analysts perform and verify the analyses being performed.

Future RAVE application as part of the reload safety evaluation is discussed in Section 2.8 of the topical report. The current bounding analysis approach described in WCAP-9272-P-A (Ref. A-2) remains unchanged with the RAVE methodology. Similar to previous implementation of methodology or code changes, reload safety evaluation documents will include a reference to an approved version of WCAP-16259-P/WCAP-16260-NP in supplement to WCAP-9272-P-A/WCAP-9273-NP-A.

Technology transfer of the RAVE methodology will follow a process that meets the requirements specified in NRC Generic Letter (GL) 83-11 Supplement 1, "Licensee Qualification for Performing Safety Analyses" (Ref. A-3).

Additional Information on VIPRE Post-CHF Model in Comparison to FACTRAN (In addition to response to Question 2 of Thermal-Hydraulic related RAIs)

The FACTRAN code uses a radial fuel pellet heat transfer model for calculating the transient temperature distribution in a cross-section of a fuel rod for a single axial node in the fuel channel. [

]^{a,c} consequently the user must manually input the local coolant conditions. The user must also input the time (or heat flux) for DNB to initiate a post-DNB heat transfer calculation. FACTRAN was approved by the NRC for use by Westinghouse in calculating the hot channel average heat flux vs. time for an external DNBR evaluation model, or for calculating the hot spot fuel and clad temperature vs. time with or without assuming DNB.

In contrast, the VIPRE code includes both a radial fuel pellet heat transfer model and a detailed multi-dimensional thermal-hydraulics model. The thermal-hydraulics model has the capability of calculating the axial variation in the reactor coolant conditions in a single channel, or in multiple channels including the effects of crossflow. The fuel pellet heat transfer calculation is performed in each axial mesh. The VIPRE code therefore calculates both the local heat flux and fluid conditions in order to determine the local DNBR. In the Westinghouse version of VIPRE, [

]^{a,c} This was addressed in the US NRC-approved topical report, WCAP-14565-P-A (Ref. A-5). For the locked rotor post-DNB Peak Clad Temperature (PCT) calculation , [

]^{a,c}

In response to Question 2 of the NRC's Thermal-Hydraulic related RAIs, for the FACTRAN vs. VIPRE Locked Rotor comparison, the FACTRAN result was compared to a multi-channel VIPRE calculation. [

]^{a,c} These differences essentially disappeared when the FACTRAN calculation was re-performed [

]^{a,c} as is also shown in the figures provided with our previous response. This shows that when the VIPRE code is used for replacing the FACTRAN code in non-LOCA Condition IV accident analysis, the VIPRE post-CHF model is in compliance with the FACTRAN Safety Evaluation Report (SER) (Ref. A-4).

Additional Clarification on VIPRE Void Model for Feedback Calculations (In addition to Supplement S-3, Void Model for Whole Core Feedback Calculation)

In the sensitivity study of the void models for feedback calculations, the VIPRE homogeneous bulk void model predicted [

]^{a,c} in the same flow channels. However, the changes in the exit void fractions have [ ]^{a,c} as shown in Figures A-1 and A-2 from the locked rotor and HZP steamline sensitivity studies, respectively. [

]^{a,c},

[

]^{a,c}

Comparisons of VIPRE predictions with void test data (Ref. A-6) show that void predictions from the homogeneous model are generally in good agreement with the experimental data when the predicted void fraction is below 30%. The uncertainties in the void fractions from the homogeneous model were about 10%, which have [ ]^{a,c} as shown in the sensitivity study. If the maximum void fraction in any RAVE feedback calculation exceeds [ ]^{a,c}, additional qualification will be performed to justify the homogeneous model for such application.

Figure A-1
Comparison of Core Average Power vs. Time from Void Model Sensitivity Study
Locked Rotor Event

a, c

Figure A-2
Comparison of Core Average Power vs. Time from Void Model Sensitivity Study
HZP Steamline Break Event

a, c

Figure A-3
Comparison of Axial Power Distributions at Different Time Steps from Void Model Sensitivity Study
Locked Rotor Event

a, c

Figure A-4
Comparison of Axial Power Distributions at Different Time Steps from Void Model Sensitivity Study
HZP Steamline Break Event

a, c

Reference:

A-1 LTR-NRC-04-71, "Responses to Request for Additional Information of Topical Report WCAP-16259-P Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," December 16, 2004.

A-2 USNRC, Office of Nuclear Reactor Regulation, "NRC Generic Letter 83-11, Supplement 1, 'Licensee Qualification for Performing Safety Analysis," June 24, 1999.

A-3 Davidson, S. L., et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A / WCAP-9273-NP-A, July 1985.

A-4 Letter from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-7908, ' FACTRAN – A Fortran IV Code for Thermal Transients in UO_2 Fuel Rod," September 30, 1986.

A-5 Sung, Y. et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A, October 1999.

A-6 Cuta, J. M., et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores Volume 4: Applications," NP-2511-CCMA, Volume 4, Section 3.0, April 1987.