August 1, 2005

Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT 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 Nuclear Regulatory Commission (NRC) staff for review. The TR presents the Westinghouse Electric Company methodology for the analysis of non-LOCA transients and accidents for pressurized-water reactors using a three-dimensional core kinetics model. The staff has concluded that use of the subject report as part of the Westinghouse design code system is acceptable. Enclosed for Westinghouse review and comment is a copy of the NRC staff's draft safety evaluation (SE) for the TR.

Pursuant to 10 CFR 2.390, we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10 working days, the draft SE will be made publicly available, and an additional 10 working days are provided to you to comment on any factual errors or clarity concerns contained in the draft SE. The final SE will be issued after making any necessary changes and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Brian Benney at 301-415-3764.

Sincerely,

/RA/
Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Draft Safety Evaluation

cc w/encl:
Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

If you have any questions, please contact Brian Benney at 301-415-3764.

Sincerely,
/RA/
Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Draft Safety Evaluation

cc w/encl:
Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

DISTRIBUTION:
PUBLIC **(No DPC for 10 working days)**
PDIV-2 Reading
RidsNrrDlpmLpdiv (HBerkow)
RidsNrrDlpmLpdiv2 (RGramm)
RidsNrrPMBBenney
RidsNrrLALFeizollahi
RidsOgcRp
RidsAcrsAcnwMailCenter
FAkstulewicz
AAttard WJensen

ADAMS Accession No.: ML051710455 *SE input NRR-043

OFFICE	PDIV-2/PM	PDIV-2/LA	SRXB/SC*	PDIV-2/SC	PDIV/D
NAME	BBenney	LFeizollahi	FAkstulewicz	RGramm	HBerkow
DATE	7/15/05	7/15/05	7/22/05	7/15/05	8/1/05

OFFICIAL RECORD COPY

DRAFT 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 license 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 regulations 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 a licensed 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.

A potential cycle history factor is utilized to account for 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.

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, demonstrate 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 apply to any 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 steam line 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-14545 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-14545-P-A (Reference 8).

With the coupled computer code RAVE methodology, Westinghouse will use three types of VIPRE models. 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. Additionally, to the whole-core model, Westinghouse will continue to use stand-alone VIPRE models described in WCAP-14545-A to calculate DNBR. In addition, Westinghouse 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 need 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. 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 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 steam line. 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 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 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 steam line 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. 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 fluid 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 steam line 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 Steam Line 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 steam line 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-800.

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 4 and 5), VIPRE-01 (References 6 and 7), 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

1. Letter from Gresham, J. A. (Westinghouse) to Wermiel, J. S. (NRC) "WCAP-16259-P/WCAP-16259-NP, 'Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis '," April 29, 2004.

2. Letter from Gresham, J. A. (Westinghouse) to Wermiel, J. S. (NRC), "Responses to Request for Additional Information to Topical Report WCAP-16259-P 'Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis '," December 16, 2004.

3. Letter from Gresham, J. A. (Westinghouse) to Wermiel, J. S. (NRC), "Responses to Request for Additional Information to Topical Report WCAP-16259-P 'Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis '," March 22, 2005.

4. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

5. Chao, Y. A., et al., “SPNOVA - A Multidimensional Static and Transient Computer Program for PWR Core Analysis,” WCAP-12394-A (Proprietary) and WCAP-12983-A (Nonproprietary), June 1991.

6. Letter from Liparulo, N. J. (Westinghouse) to Jones, R. C. (NRC), “Process Improvement to the Westinghouse Neutronics Code System,” NTD-NRC-964679, March 29, 1996.

7. Stewart, C. W., et al., “VIPRE-01: A Thermal/Hydraulic Code for Reactor Cores,” Volumes 1, 2, 3 (Revision 3, August 1989), and Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute, Palo Alto, California.

8. Sung, Y. X., Schueren, P. and Meliksetian, A., “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), October 1999.

9. “RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,” Volumes 1-4, NP-1850-CCM, Electric Power Research Institute, May 1981.

10. Huegel, D. S., et al., “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” WCAP-14882-P-A (Proprietary), April 1999.

11. Beard, C. L., et al., “Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics,” WCAP-15806-P-A, October 2003.

12. Davidson, S. L., et al., “Westinghouse Reload Safety Evaluation Methodology,” WCAP-9272-P-A (Proprietary) and WCAP-9273-NP-A (Nonproprietary), July 1985.

13. Nguyen, T. Q., et al., “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” WCAP-11596-P-A (Proprietary) and WCAP-11597-A (Nonproprietary), June 1988.

14. Liu, Y. S., et al., “ANC - A Westinghouse Advanced Nodal Computer Code,” WCAP-10965-P-A (Proprietary) and WCAP-10966-A (Nonproprietary), September 1986.

15. Liu, Y. S., “ANC - A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery,” WCAP-10965-P-A, Addendum 1 (Proprietary) and WCAP-10966-A Addendum 1 (Nonproprietary), April 1989.

16. Letter from Liparulo, N. J. (Westinghouse) to Jones, R. C., (NRC), “Notification to the NRC Regarding Improvements to the Nodal Expansion Method Used in the Westinghouse Advanced Nodal Code (ANC),” NTD-NRC-95-4533, August 22, 1995.

17. Letter from Thomas, C. O. (NRC) to Schnatz, T. W., (UGRA), “Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, ‘RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems,’ and EPRI NP-1850-CCM, ‘RETRAN-02 A Program for Transient Thermal-Hydraulic Analysis for Complex Fluid Flow Systems,” September 2, 1994.

18. Letter from Thadani, A. C. (NRC) to Furia, R.(GPU), “Acceptance for Referencing Topical of Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004,” October 19, 1988.

19. Letter from Thadani, A. C. (NRC) to Boatwright, J. (TUEC), “Acceptance for Reference of RETRAN02/MOD005.0,” November 1, 1991.

20. Letter from Thomas, C. O. (NRC) to Rahe, E. P., Jr. (Westinghouse), “Acceptance for Referencing of Licensing Topical Reports: WCAP-7907 ‘LOFTRAN Code Description,” July 29, 1983.

21. “Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01,” U.S. Nuclear Regulatory Commission, July 18, 1985.

22. Wheeler, C. L., et al., “COBRA-IV-I: An Interim Version of COBRA for Thermal/Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores,” BMWL-1962, Richland, Washington, Pacific Northwest Laboratory, March 1973.

23. Chelemer, H., et al., “THINC-IV - An Improved Program for Thermal/Hydraulic Analysis of Rod Bundle Cores,” WCAP-7956-A, Westinghouse Electric Corporation, February 1989.

24. Hargrove, H. G., “FACTRAN-A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod,” WCAP-7908-A, Westinghouse Electric Corporation, December 1989.

25. “Safety Evaluation of Topical Report WCAP-11837, ‘Extension of Methodology for Calculating Transition Core DNBR Penalties',” U.S. Nuclear Regulatory Commission, October 12, 1989.

Principal Contributors: Tony Attard
Walt Jensen

Date: August 1, 2005