UNITED STATES
NUCLEAR REGULATORY COMMISSION
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MA[illegible] 1986

Mr. J. A. Blaisdell, Chairman
UGRA Executive Committee
Northeast Utilities Service Co.
P. O. Box 270
Hartford, CT 06141-0270

Dear Mr. Blaisdell:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT, EPRI NP-2511-CCM, "VIPRE-01: A THERMAL-HYDRAULIC ANALYSIS CODE FOR REACTOR CORES", VOLUMES 1, 2, 3 AND 4

We have completed our review of the subject topical report submitted by the Utility Group for Regulatory Applications (UGRA) by letter dated December 1[illegible], 1984. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that UGRA publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, UGRA and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Charles E. Rossi

Charles E. Rossi, Assistant Director
for PWR Licensing-A
Division of PWR Licensing-A

Enclosure:
As stated

ENCLOSURE

Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01

1. INTRODUCTION

By letter dated December 17, 1984 (Ref. 1), the Utility Group for Regulatory Applications (UGRA) submitted a topical report on the VIPRE-01 code (Ref. 2) for staff review. VIPRE-01 is a subchannel thermal-hydraulic computer code developed by Battelle Pacific Northwest Laboratories under the sponsorship of the Electric Power Research Institute (EPRI). It was developed for the evaluation of nuclear reactor safety margin. Given the geometry of the reactor core and coolant channel, and the boundary conditions or forcing functions, VIPRE-01 calculates core flow distributions, coolant conditions, fuel rod temperature and the minimum departure from nucleate boiling ratio (MDNBR) for steady state, and certain operational transients and abnormal events. The boundary conditions or forcing functions are generally obtained from the calculations of reactor system codes, such as RETRAN (Ref. 3).

VIPRE-01 was developed on the strengths of the COBRA series of codes. Many features have been adopted from the COBRA codes to make VIPRE-01 reliable, efficient, and easy to apply to utilities' needs. The basic computational philosophy of VIPRE-01 comes from COBRA-IIIC (Ref. 4). Both use the subchannel analysis concept where the reactor core or fuel bundle is divided into a number of quasi-one-dimensional channels that communicate laterally by diversion crossflow and turbulent mixing. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow per unit length, and momentum pressure drop. The flow field is assumed to be incompressible and homogeneous, although models are added to reflect subcooled boiling and liquid/vapor slip. Fluid properties are functions of the local enthalpy and a uniform but time-varying system pressure with an option to add the effects of local pressure. Several features of the

COBRA-IIIC/MIT (Ref. 5) code were also incorporated into VIPRE-01, including the ability to split a given total flow for a uniform pressure drop and the ability to specify several different axial power profiles in the same case. COBRA-IV-I (Ref. 6) contributed problem dump/restart capability and a redimensioning preprocessor. VIPRE-01 also uses the spatially implicit energy equation from COBRA-IV-I. The COBRA-WC (Ref. 7) "RECIRC" scheme was included as an option to allow reverse and recirculating flow in both the steady state and transient calculations. Though the above discussed features and models exist in these other codes, VIPRE-01 brings them together into one package.

User convenience was a stated objective during the development of VIPRE-01, resulting in many user options and defaults. As a result, the user has the option of several heat transfer and CHF correlations. Also, a selection of correlations is available for calculating subcooled void, two-phase quality/ void and two-phase friction multipliers. In many cases, if a selection is not made by the user, the code will default to a programmed selection. In addition, several input data also have default values.

In 1984, a Design Review Committee of thermal hydraulic experts was formed by EPRI to perform a design review of the VIPRE code. A VIPRE-01 Design Review Executive Summary Report (Ref. 8) was submitted by UGRA for staff information. The objectives of the design review are to ensure that VIPRE-01 conforms to the EPRI criteria for the release of VIPRE-01 including adequacy of documentation, correctness of solution techniques and coding, and validation of the code against relevant test data and similar calculation techniques, and to assure that VIPRE-01 is adequate to perform analysis of operation transients and certain abnormal events for PWRs and BWRs.

The COBRA IIIC and COBRA-IV codes were developed under the sponsorship of the Energy Research and Development Administration and the Nuclear Regulatory Commission and have been used by the nuclear industry for subchannel thermal-hydraulic calculations and by the NRC staff for audit calculations. The COBRA-IIIC/MIT code was an offspring version of COBRA-IIIC with modifications made by MIT and has been reviewed and approved by NRC for licensing applications. Therefore, the staff review of the VIPRE-01 code is particularly

concentrated on the differences between the VIPRE-01 code and the COBRA codes, and on the data comparisons for verification and benchmark of the VIPRE-01 code.

Although the VIPRE-01 code was developed for the safety analyses of both PWR and BWR cores, this review was limited to application to PWR plants in steady state and transient analyses up to critical heat flux. Post CHF analyses and aspects of the code dealing with post-CHF phenomena were excluded from this review. In addition, the staff has expressed its position during the review that approval of the VIPRE-01 document does not permit the use of VIPRE-01 for licensing application. Each VIPRE-01 user should first submit a document describing its proposed use, computer codes with which it will interact, the source of each input variable, the selected correlations and the justifications of using the selected correlations. UGRA in its response (Ref. 9) agreed with the staff position and stated that it is the responsibility of each organization using VIPRE-01 for licensing applications to provide documentation describing how they intend to use VIPRE-01 and provide justification for their specific modeling assumptions and analysis techniques. Therefore, the staff review of VIPRE-01 is not intended to get into details of each option or correlation described in the VIPRE-01 document to determine its acceptability. Rather, the review is confined to the acceptability of the VIPRE-01 general thermal hydraulic solution for PWR licensing calculations.

2, STAFF REVIEW

The VIPRE-01 report consists of four volumes. Volume 1, Mathematical Modeling, addresses the theoretical development of general conservation equations, constitutive relationships, correlations and numerical solution methods. Volume 2 is User's Manual which provides the input descriptions and sample problems. Volume 3 is Programmer's Manual which describes general coding philosophy, routines, data files and auxilary programs. Volume 4 is Application which describes the VIPRE-01 validation efforts and results for various applications. In the original submittal (Ref. 1), only the first three volumes of VIPRE-01 were submitted. Revisions 1 and 2 were subsequently submitted on February 28, 1985 (letter, G. S. Srikantiah of EPRI to Y. Hsii of NRC) and

August 1, 1985 (letter, R. Sterner of Northeast Utilities to A. Gill of NRC), respectively. Volume 4 has not been formally published by EPRI, but a draft version was submitted to NRC. Our review of Volume 4 is based on the draft version with an understanding that the published version will be essentially the same as the draft version with all the deviations made known to the NRC staff.

2.1 VIPRE Thermal-Hydraulic Model

The basic computational philosophy of VIPRE-01 comes from COBRA-IIIC (Ref. 4). Both use the subchannel analysis concept where the reactor core or fuel bundle is divided into a number of quasi-one-dimensional channels that communicate laterally by diversion cross-flow and turbulent mixing.

The VIPRE-01 subchannel modeling allows a region of fluid flow to be described by a number of channels of various sizes and shapes. Channel size and shape may be small and regular or relatively large and irregular simply by inputting flow area and wetted perimeters. Hence, the hottest location in the core can be modeled in detail (such as a subchannel in a bundle array) and cooler locations in the core, which may include several bundles, can be lumped together into a single channel. In any analysis where channel sizes differ, it is desirable to input cross flow variables (which are model dependent) consisting of the gaps between the channels and the distance between connecting channel centroids.

For each axial segment, conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow per unit length, and momentum pressure drop. The flow field is assumed to be incompressible and homogeneous, although models are added to reflect subcooled boiling and co-current liquid/vapor slip. Fluid properties are functions of the local enthalpy and a uniform but time-varying system pressure with an option to add the effects of local pressure.

The governing equations for the conservation of mass, energy and momentum, and the two phase flow models of friction factors, two-phase friction multipliers, subcooled and bulk quality/void models are described in Volume 1 of the

VIPRE-01 report. These equations and models are reviewed by comparison with similar equations in the other widely used thermal hydraulic codes such as COBRA-IIIC and COBRA-IV. These comparisons showed the equations, calculational models and correlations to be similar in most aspects. VIPRE-01 contains flow-field models not found in COBRA-IIIC. For example, VIPRE-01 has subcooled quality, quality/void (EPRI void) and two-phase friction (EPRI two-phase friction) models not found in COBRA. In addition, VIPRE-01 contains the RECIRC solution option and a fuel rod heat conduction model with a dynamic gap model that are not found in the COBRA-IIIC and COBRA-IV codes. These models were evaluated on the basis of the qualification calculations, which are discussed in Sections 2.2 and 2.5.

There is a difference between VIPRE-01 and COBRA in the formulation of the cross-flow equations when solving the lateral momentum equations. The VIPRE-01 UPFLOW solution technique eliminates w (cross flow) rather than DP/DX (lateral pressure gradient) as done in COBRA. It is stated in the VIPRE-01 report that the COBRA technique results in a larger (usually 30 to 50%) coefficient matrix [A] and is less diagonally dominant and that the new VIPRE-01 formulation can be solved faster and is less prone to convergence difficulties when solved iteratively.

Usually strong diagonal dominance of a matrix implies that the matrix is well conditioned and the more well conditioned a matrix is, the more stable the numerical solution to the matrix problem. We agree that in steady state cases where the time step has no effect, the coefficient matrix [A] in VIPRE-01 is always more diagonally dominant than that in COBRA. As stated in the COBRA-IIIC documentation, however, reducing Δx or Δt in the COBRA equations adds more diagonal dominance to the matrix [A] for transient cases. Since the VIPRE-01 UPFLOW solution technique solves for pressure rather than crossflow, the Δx and Δt terms have been eliminated in the UPFLOW formulation. By simply reducing the Δx and Δt in COBRA for transient cases, it is always possible to make the COBRA-IIIC coefficient matrix [A] more diagonally dominant than the VIPRE-01 UPFLOW model. Although the UPFLOW model in VIPRE-01 may be more efficient than the corresponding COBRA model, UPFLOW may be more sensitive and less numerically stable than the corresponding COBRA-IIIC model in transient

analyses. UGRA, in response to a staff question (Ref. 9) asserts that the diagonal dominance of the coefficient matrix should not be a critical issue because these inner equations in both VIPRE and COBRA are solved by direct elimination rather than the iterative method. Based on the comparison of calculated results from the VIPRE-01 UPFLOW technique with COBRA-IIIC calculations (Section 9, Volume 4 of the VIPRE-01 documentation), the UPFLOW formulation gives results comparable to COBRA-IIIC. Consequently, the UPFLOW solution option is judged to be properly implemented, and is acceptable.

In the crossflow calculations, the lateral form loss coefficient K_G is used for crossflow resistance. K_G is input as a uniform value in COBRA-IIIC, but is a variable dependent on geometry in VIPRE-01. It is stated in the VIPRE-01 report that when K_G is made proportional to the centroid distance divided by the rod pitch, an improvement is observed in the stability of the UPFLOW solution for some calculations. However, the uniform value used in COBRA-IIIC does not appear to present any stability difficulties for the same calculation. Hence we questioned whether the UPFLOW convergence is sufficiently sensitive to K_G to require the user to input data that accurately reflects the geometry. In response, UGRA states that K_G has no direct effect on stability but is a physical drag coefficient for lateral flow through one gap. The use of the non-uniform K_G's increases the crossflow resistance for lumped channels in proportion to the number of rod rows between the channel centroids. We agree with UGRA that K_G has no direct effect on numerical stability. Indeed the use of a non-uniform K_G represents an improvement over a uniform constant when lumped channel models are used in the core thermal hydraulic analysis. Using K_G proportional to the centroid distance divided by the gap width provides a correct physical representation in cases where various sizes of lumped subchannels can be modeled to represent a reactor core.

The effects of the damping factors on convergence and numerical stability are also studied. A damping factor is an input weighting factor used in the numerical iterative process to obtain a tentative value of an updated parameter from the values of the previous iterations. VIPRE-01 contains a default value for each input damping factor used in the iterative process. For example, a default value of 0.8 is used for the damping factor Dp for lateral pressure

difference update in the UPFLOW solution, and the damping factor (D_R) for axial and cross flow updates in the RECIRC solution is also defaulted to 0.8.

The use of a damping factor when solving nonlinear problems iteratively is consistent with good numerical practice. Nonetheless, the use of a damping factor other than the default value (0.8) may signify inherent numerical instability. It suggests that when the computations are unstable with D_R = 0.8, the user may have to reduce D_R to achieve convergence since use of smaller value of D_R tends to enhance stability. In response to a staff question, UGRA stated that a damping factor of 0.8 is appropriate for most of the problems encountered and that convergence problems are usually the result of modeling and not an inappropriate choice of the damping factor. This suggests in cases where D_R = 0.8 doesn't work, the user probably should modify the core modeling instead of adjusting D_R. This, however, is a convergence problem and does not necessarily mean bad results (or false convergence). Indeed if the convergence criteria are properly selected and a convergence problem occurs, the calculation would stop and sufficient information will be printed out by VIPRE to allow the user to determine the state of convergence of each simulation. Therefore, the use of damping factors and their effect on numerical stability is not a concern.

2.2 VIPRE Verification and Qualification

Qualification or benchmark calculations are provided in Volume 4 of the VIPRE-01 report. These calculations were reviewed for agreement with existing data. These qualification calculations cover a large range of data from comparisons of VIPRE-01 calculations to simple heat-conduction problems having analytical solutions, to complex experiments involving flow blockage, two-phase pressure drop, void fraction measurements, fuel temperatures and heat transfer.

The review is discussed in the following sections.

2.2.1 Flow Distribution, Pressure Drop and Void Fraction

Flow distribution and two-phase pressure drop calculations using VIPRE-01 are compared to test data. The flow distribution calculations were made simulating experiments designed to measure flow redistribution resulting from partial flow blockages and large power gradients. These calculations were preformed with the RECIRC solution option. In general, the VIPRE-01 benchmarking calculations for flow redistribution were in excellent agreement with the test data indicating that both the axial and lateral flow equations are properly implemented. One of these calculations simulated a "complete" flow blockage test. The results of this calculation showed flow reversal occurring in the blocked channels. Thus, these flow redistribution calculations also indicated the RECIRC option is valid, properly implemented and capable of calculating reversed flow.

Section 2 also contains two-phase pressure drop calculations where two-phase friction multipliers were compared with values determined from experiments. The calculated multipliers were based on various combinations of the quality, bulk-void and two-phase friction multiplier correlations available in VIPRE-01. Based on these calculations and their comparison with test data, the multipliers obtained from the Columbia/EPRI correlation in conjunction with the EPRI void model were consistently (11 tests out of 12) in better agreement with the data. The VIPRE-01 documentation recommends these correlations and the code will default to these correlations if not otherwise specified by the user.

Section 3 contains void fraction calculations and their comparisons with test data. These VIPRE-01 calculations used the Columbia/EPRI two-phase friction multiplier correlation. Three sets of calculations were made with each of the three quality relationships (homogeneous, Levy subcooled and EPRI) available in the code. Two sets of these calculations considered four quality/void correlations: homogeneous, Armand, Smith slip and the Zuber-Findlay correlations. The third set, which used the EPRI quality model, considered just the EPRI quality/void correlation. This latter combination is the only quality/void relationship available when selecting the EPRI quality model. This combination is referred to as the EPRI void model and is the default model in VIPRE-01.

The results of these calculated void fractions and comparisons with test data were mixed. Some combinations of quality/void correlations agreed better with one test series, whereas, a different combination would provide better agreement for another series of tests. In general, the EPRI void model performed as well as any other combination for the test data considered. Part of the mixed results may be attributed to uncertainty in the test data. In some cases the data demonstrated considerable scatter and no attempt was made to quantify the uncertainty.

Most of the two-phase pressure drop and void fraction data available are in the range of 1000 psia or below, which limits the benchmark calculations for these variables since the code is to be applied to PWR conditions in which the pressure is typically greater than 2000 psia. Therefore, neither the code calculations nor the data base used to qualify the two-phase pressure drop and void fraction calculations covered PWR operating conditions. However, one test at 1990 psia was simulated in a VIPRE-01 calculation with excellent agreement. In addition, the liquid-to-vapor density ratio moves closer to 1.0 as the pressure is increased resulting in smaller void fractions, a more homogeneous flow and smaller two-phase friction multipliers. Therefore, no significant changes would be expected in results at PWR conditions and the benchmark calculations with low pressure data provide sufficient verification for the VIPRE-01 two-phase pressure drop and void fraction calculations.

Based on the evaluation of the benchmark calculations discussed above, we conclude that the EPRI void model and the EPRI correlation for two-phase friction are acceptable for licensing calculations. In some applications, other correlations or combination of correlations may provide better results. It may be advisable, therefore, for the user to review these benchmark calculations to determine which correlations would give the best results for a particular application. It is noted that the slip model quality/void correlation was excluded from the benchmark calculations. This slip model requires a slip ratio be input by the user. When this option is considered for licensing analyses, the user should justify their input for the slip ratio.

2.2.2 Heat Transfer

The VIPRE-01 qualification analyses in Section 4 included benchmark calculations to qualify the fuel rod heat conduction model. These benchmark calculations include comparisons to analytical heat conduction problems and to fuel centerline temperature data. The comparison to the analytical problem showed excellent agreement and was done for both steady-state and transient conditions. The transient calculation was completed with two time step sizes. Results from the smaller time step were in slightly better agreement with the analytical solution. Calculations for comparison with measured fuel centerline temperature data used the VIPRE-01 dynamic gap conductance model. This gap conductance model is based on the Ross-Stoute model and is used in the fuel rod heat transfer codes GAPCON (Ref. 10, 11) and FRAP (Ref. 12, 13) which have been extensively benchmarked to experimental data. Thus, the VIPRE-01 fuel rod conduction model is comparable to the model in the GAPCON and FRAP codes. Although the benchmark calculations and their comparison to fuel centerline temperature data were limited in scope and considered only steady-state conditions, the results verify that the heat conduction model is valid and properly implemented. Based on the use and qualification of the model in GAPCON and FRAP, we conclude that the fuel rod heat-conduction model is acceptable for licensing analyses.

Section 5 contains VIPRE-01 heat transfer calculations compared with saturated heat transfer data. These calculations were completed using the five saturated boiling heat transfer correlations available in VIPRE-01 for each of the experiments simulated. However, an adequate data base for the evaluation of boiling heat transfer correlations is not provided. Although all correlations yielded reasonable results, the results are varied and no single correlation is recommended in the VIPRE-01 documentation. The results of the comparisons, however, can be used to evaluate the behavior of the correlations and can serve as a basis for selecting an appropriate one for a particular application. Therefore, we recommend that VIPRE-01 users review the boiling heat transfer calculations in Volume 4 prior to selecting a saturated heat transfer correlation.

2.3 CHF Analyses

The qualifying calculations also contained a large number of minimum DNBR calculations (Section 6, Volume 4), which used the various CHF correlations available in VIPRE-01. One correlation, the EPRI-1 correlation was developed for both BWR and PWR applications and is the default correlation. The DNBR limit as well as its derivation are not provided.

The intent of this review is concentrated on the validity of the VIPRE-01 thermal hydraulic analysis. Any new CHF correlation which has not been previously approved for licensing application should be submitted in a separate topical report for NRC review and approval. In addition, the use of VIPRE-01 with an approved CHF correlation and its safety limit should be justified by showing that, given the correlation data base, VIPRE-01 gives the same or a conservative safety limit. UGRA in response to a staff question stated that the justification for the use of a particular CHF correlation and its safety limit will be provided by each organization using VIPRE-01 for a particular licensing application. Our review effort, therefore, has not concentrated on whether a particular CHF correlation is acceptable for licensing calculations. Rather the review concentrated on a broader question of whether a CHF correlation developed with steady state CHF test data can be used with the VIPRE-01 transient solution in the calculations of DNBR for reactor transients.

Most CHF correlations and their respective DNBR limits are developed with steady state CHF test data and steady state thermal hydraulic analyses. The use of a steady state CHF correlation in the design DNBR calculations of the reactor transients has been accepted by NRC when used with a quasi-steady state or "snap shot" approach. In the quasi-steady state approach, the DNBR calculation at any instant during a transient is performed with a steady state core thermal hydraulic analysis where the boundary conditions provided by the system transient analyses are held constant and therefore the calculated local fluid conditions in the core may deviate somewhat from the actual fluid condition for the transient. Since VIPRE-01 is a transient core thermal hydraulic code, the use of a steady state CHF correlation coupled with a transient solution required evaluation.

Section 6.7 of Volume 4 of VIPRE provides a study to determine the applicability of the steady-state CHF correlations to the transient analysis. The study utilized the transient CHF data obtained from the tests performed at General Electric's ATLAS facility (Ref. 14). Though VIPRE-01 predictions are conservative in predicting the time to CHF earlier than the data for most of the cases analyzed, the results also show that the transient DNBR response is dependent upon the CHF correlation used and that the use of the steady state CHF correlation becomes more questionable as the transients become more rapid and complex.

The staff, in addition to requesting the UGRA to provide additional justifications for the acceptability of the steady state CHF correlations for the VIPRE-01 transient analysis, also did an extensive review of the available literature regarding transient CHF. A majority of the transient CHF data and studies involve reactor coolant flow and power transients. Other data come from depressurization transients simulating LOCA blowdown. A summary of findings is given below.

Tong (Ref. 15) reviewed the transient CHF test data of Tachibana, et. al. (Ref 16), Martenson (Ref 17), and Schrock, et. al. (Ref 18) for the power ramp transients and found that the transient CHF values are higher than those under steady state and that the transient CHF increases as the power impulse time decreases. His review of other transient CHF data with power ramp and flow coastdown by Tong, et. al. (Ref 19, 20), Moxon and Edwards (Ref 21), and Cermak, et. al. (Ref 22) also concluded that transient CHF can be predicted by using steady state CHF data from rod bundles having the same geometry and tested under the same local fluid conditions. LeTourneau and Green (Ref 23) used a rod bundle test section to experimentally simulate constant-power flow coastdown transients, and Zielke and Wilson (Ref 24) studied the CHF data from power ramp and flow reduction tests simulating the control rod ejection and four pump coastdown transients, respectively. They found that the use of CHF correlations developed based on steady state rod bundle CHF data can correctly or conservatively predict transient critical heat flux when the instantaneous local flow conditions are used. Yuelys-Miksis and Shier (Ref. 25) analyzed some Columbia University CHF data from flow decay transients using the Columbia

CHF correaltion and the transient solution of COBRA-IIIC thermal hydraulic code. The results of their analysis indicate that a steady state CHF correlation coupled with a transient thermal hydraulic code which provides the instantaneous local fluid conditions accurately predicts transient CHF. A study by Babcock and Wilcox Company (Ref. 26) using the LYNXT thermal hydraulic code and the B&W-2 CHF correlation to predict CHF under power and flow transients shows that both quasi-steady state and transient analyses produce essentially the same transient minimum DNBR predictions and that the LYNXT/B&W-2 analysis predicts CHF occurrences earlier than the experimental indication in more than 95 percent of the cases analyzed. Ginoux (Ref. 27) also studied the transient CHF data and concluded that it is conservative to use steady state CHF correlations for power excursion burnout and that the loss of flow transients due to a pump failure are much slower than power excursion and therefore almost no change with respect to steady state conditions is expected as proved by measurements. Leung (Ref. 28) made an extensive literature survey and concluded that the use of steady state CHF correlations and local parameters can predict CHF for power and flow transients.

Observations regarding the CHF data for the depressurization transients are somewhat inconsistent. Cermak et. al. (Ref. 22) studied the test data from experiments with a 21-rod bundle with pressure blowdown of originally subcooled boiling flow and found that the transient CHF can be predicted from the steady state CHF data of the same local flow conditions. Babcock and Wilcox Company (Ref. 29) analyzed the transient CHF data with pressure reduction at a rate of 20 psi per second and found that the use of instantaneous system pressure and local flow conditions in a steady state CHF correlation provided acceptable or conservative results. Leung and Gallivan (Ref. 30) analyzed Combustion Engineering's single tube blowdown experimental data simulating a double ended break and found that the three round tube CHF correlations (Bowring, Biasi and CISE) used in the analysis correlate the transient CHF data reasonably well. On the contrary, a Westinghouse study (Red. 31) on the transient CHF of a 5x5 square rod array under conditions typical of the blowdown phase of a large break LOCA (with a depressurization rate of 100 to 350 psi per second) showed mixed results. Some CHF correlations such as B&W-2 predict the blowdown CHF time reasonably well, whereas others such as Biasi and Bowring WSC-2

correlations either failed to predict CHF or predicted CHF to occur later than the tests indicated. The fact that the Biasi correlation, which was developed with round tube steady-state CHF data, predicts the single tube transient CHF data well but fails to predict rod bundle transient CHF data indicates that the transient CHF prediction of a correlation is dependent upon its ability to predict the steady state CHF data. Since the Biasi correlation was developed with round tube CHF data, it may not be capable of predicting the steady state CHF of a rod bundle, let alone the transient CHF for a rod bundle. The contradicting results also suggest that the use of a steady state CHF correlation in a more rapid depressurization transient such as LOCA may need further examination. However, since Appendix K to 10 CFR 50 has ruled some steady state CHF correlations to be acceptable for LOCA analysis and since VIPRE-01 is not intended to be used for LOCA analysis, the application of a steady-state CHF correlations for the rapid depressurization in the blowdown phase is not a subject of this review.

In response to a staff question, UGRA provided additional analyses (Ref. 32) using the transient CHF test data (Ref. 33, 34, 35) obtained at the Oak Ridge National Laboratory's Thermal Hydraulic Test Facility (THTF). The THTF tests simulate a power ramp transient and two transients with power ramp-depressurization-flow variations. The analyses were performed with the VIPRE-01 transient solutions and the EPRI-1 CHF correlation. The results show that the VIPRE-01/EPRI-1 calculations predict CHF to occur (CHF is assumed to occur when the calculated DNBR is equal to or less than 1) earlier than the tests indicated, i.e. the VIPRE-01/EPRI-1 prediction is conservative. The UGRA also provided the comparisons between the DNBRs calculated with the transient and quasi-steady state approaches. The comparisons were done by Texas Utility for Comanche Peak Station and Northeast Utilities for Haddam Neck on a complete loss of flow transient, and by TVA for Sequoyah Station for an uncontrolled rod withdrawal transient. The comparisons were done with the W-3 CHF correlation and VIPRE-01 in which both transient solution and "snap shot" approach were used. The results show good agreement in the DNBRs calculated by both transient and quasi-state approaches.

To summarize, studies have shown that the transient CHF for power ramp and flow coastdown transients are higher than the steady state CHF, and that, except for very rapid depressurization, the use of CHF correlations developed with steady state CHF data can correctly or conservatively predict the transient CHF when the instantaneous local fluid conditions are used. Since VIPRE-01 is not intended to be used for LOCA analysis with rapid depressurization, we conclude that using a steady state CHF correlation in VIPRE-01 transient calculation is acceptable provided that (1) the CHF correlation with VIPRE-01 calculations and its DNBR limit have been reviewed and approved by NRC, and (2) the application is within the range of applicability of the correlation including fuel assembly geometry, spacer grid design, pressure, coolant mass velocity, quality, etc.

2.4 Sensitivity Analysis

Section 7 of Volume 4 of the VIPRE-01 report provides the results of a sensitivity study of the effects of many input parameters on the VIPRE-01 thermal hydraulic solution and DNBR prediction. The input parameters studied are axial and radial noding, cross flow resistance, turbulent mixing, inlet flow distribution, specification of fluid properties, the problem solution options and time step size. In general, these sensitivity calculations provide valuable information to the user in setting up calculational models and selecting proper options. The results of the study of the effects on DNBR predictions show that:

1. The VIPRE-01 predictions are sensitive to axial noding in that enough nodes must be provided to resolve the detail in the axial power profile and the flow field such as flow redistribution around area changes, local pressure drop and local boiling. Once a sufficient accuracy is obtained, the VIPRE-01 prediction becomes insensitive to further axial node refinement.

2. As long as at least one full row of subchannels are placed to completely surround the hot channel to adequately resolve the details of the flow field in the vicinity of the hot channel, the hot channel flow conditions

are very insensitive to the core radial layout or how the rest of the hot bundle and core are modeled.

3. For most of normal VIPRE-01 applications when the axial flow is predominant relative to the cross flow, the gap-to-length ratio (s/l) and crossflow resistance (kij) have insignificant effects on the mass flux and DNBR.

4. The inlet flow distribution has a very small effect on the VIPRE-01 results in the upper half of the core since the flow redistribution recovers the inlet flow maldistribution within the first few feet.

5. Aside from a possible local pressure effect, VIPRE-01 is insensitive to the way the fluid properties are computed by either the interpolation of the table entries or direct evaluation of internal properties formulations.

6. The hot channel flow conditions and DNBR are sensitive to the turbulent mixing. Use of a higher turbulent mixing efficient will result in smoothing the hot channel flow conditions, i.e. a decrease in enthalphy and increase in flow rate and DNBR. Therefore, unless the value of the turbulent mixing coefficient can be verified by experimental data, either no turbulent mixing or a conservatively small mixing coefficient should be used in licensing calculations. The VIPRE-01 user should justify the value of the turbulent mixing coefficient in the application.

7. Since VIPRE-01 used the implicit solution technique, there is no time step limitation relative to numerical stability. However, the time step size used in the calculation does have an effect on the calculated results. The study shows that without subcooled boiling, the VIPRE-01 results are invariant with decreasing time step after a point. However, with the consideration of subcooled boiling, the resulting DNBR is greatly affected by the time step sizes. The behavior is attributed to the subcooled boiling models used in VIPRE-01. Both Levy and EPRI subcooled void models were profile fits developed solely from steady state data which depend on the

local flow rate with no inertia term. The sensitivity study results show that a profile fit subcooled void model based on steady state data is not suitable in boiling transients in which the Courant number ($N_c = u\Delta t/\Delta x$, where u is flow velocity, Δt is time step and Δx is axial node size) is less than 1.

8. A study is made of the calculation results using the three numerical solutions options in VIPRE, i.e. the UPFLOW direct solution, UPFLOW iterative solution and the RECIRC solution. A comparison of the test channel crossflow profiles which are the most sensitive to change in solution shows that the three options have essentially the same results except that the RECIRC calculated cross flow differs somewhat in the upper portion of the core due to the difference in the finite difference formulation for crossflow. However, this difference would result in only slight differences in axial flow and DNBR since crossflow velocity represents only a small fraction of the axial flow. The study shows that the UPFLOW and RECIRC options are properly implemented and these solution techniques are acceptable for licensing calculations.

2.5 Audit Calculation

As a part of the VIPRE-01 review effort, we have performed an audit calculation using the COBRA-IV computer code to provide a comparison with the VIPRE-01 results. The event selected was a locked rotor in a PWR reactor coolant pump at full power which results in a rapid decline in coolant flow and results in departure from nucleate boiling. The input to and result of the VIPRE-01 calculations were provided by Northern State Power Company for the Prairie Island Nuclear Station Unit 1.

2.5.1 Computer Code Input

The basic approach to reactor fuel bundle or core modeling in VIPRE-01 and COBRA-IV is to divide the bundle or core into computational cells or channels. Use of the computational cell concept allows subchannel, bundle or core and general flow field analysis to be considered in a unified approach. The

computational cell concept also allows the user to discretely describe the hot fuel rod(s) and the associated subchannels while allowing the remainder of the bundle or core to be considered by lumping the descriptive input variables. The calculational models for the Prairie Island Unit 1 core, which were input to VIPRE-01 and COBRA-IV are represented by Figures 1 and 2 respectively. Although both models assumed 1/8 core symmetry, the VIPRE-01 model contained 79 channels, whereas, the COBRA-IV model contained 29. The VIPRE-01 model discretely represented 49 fuel rods and 36 associated subchannels. On the other hand, the COBRA-IV model represented 7 fuel rods and 4 subchannels with an increasing degree of lumping as the model moves away from the hot fuel rod.

Other than the difference in the number of channels to model the reactor core, other inputs including user selected options and correlations, grid loss coefficients, lateral flow resistance factors and reactor operations conditions are essentially the same for both VIPRE-01 and COBRA-IV. Operating conditions consist of reactor power level and distribution, the system operating pressure and the inlet coolant mass flow and enthalpy. Input transient boundary conditions or forcing functions consist of time histories of the reactor operating variables shown in Figure 3. The reactor power history is input as the core average heat flux. The axial distribution is shown in Figure 4 and the radial distribution provided in Table 1. The radial distribution corresponds to the fuel rods identified in Figures 1 and 2. Weighting factors, which correspond to the number of fuel rods associated with each rod in the model, are included in Table 1. Other input data considered important to the calculations are contained in Table 2. The RECIRC solution technique was used for the VIPRE-01 calculations, whereas the explicit solution scheme was used for the COBRA-IV calculation.

Since the reactor power is input as an average heat flux, the fuel rod conduction models were not used during these calculations. Hence, neither a comparison of calculated fuel rod temperatures nor an evaluation of the VIPRE-01 fuel rod heat conduction model was obtained.

2.5.2 Comparison of Calculated Results

A comparison of several steady state variables calculated by VIPRE-01 and COBRA-IV is provided in Table 3. The minimum DNBR calculated by COBRA-IV is about 3.3% greater than that calculated by VIPRE-01. However, the VIPRE-01 value includes a "cold-wall" factor that the COBRA-IV value does not and when this is considered, the difference in the steady state MDBR is reduced to about 1.7%. The greatest difference in the other calculated data represented in Table 3 is ~4.5% which occurred in the coolant mass flux at the location of minimum DNBR. The steady state data calculated by the two codes are in excellent agreement.

Minimum DNBRs calculated by the two codes during the transient are compared in Figure 5. The initial DNBRs agree within about 3.3% as discussed above. At approximately 0.5 seconds, however, the location of the Minimum DNBR calculated by both codes changed to another subchannel (VIPRE-01 subchannel 49 and COBRA-IV subchannel 4) which does not contain a control-guide tube. Consequently, the cold-wall factor applied to the minimum DNBR by VIPRE-01 is 1.0 and the minimum DNBRs are in better agreement. This good agreement continues until about 1.5 seconds when the data begin to diverge with the VIPRE-01 data being more conservative. The divergence continues to about 2.5 seconds when a slow convergence is observed. The maximum difference in these minimum DNBR values, occurring at about 2.5 seconds, is 0.123 for VIPRE-01 compared to 0.278 for COBRA-IV.

The greatest difference in minimum DNBR, which occurred at about 2.5 seconds, is well after the DNBR of 1.0 was calculated and consequently is of no interest in either design or safety analyses. Based on the good comparison between these calculations for minimum DNBR ≧1.0, it is concluded that the flow models and the RECIRC solution option used in the VIPRE-01 calculation are valid and properly implemented. It is concluded therefore that the RECIRC solution option is acceptable for licensing analyses.

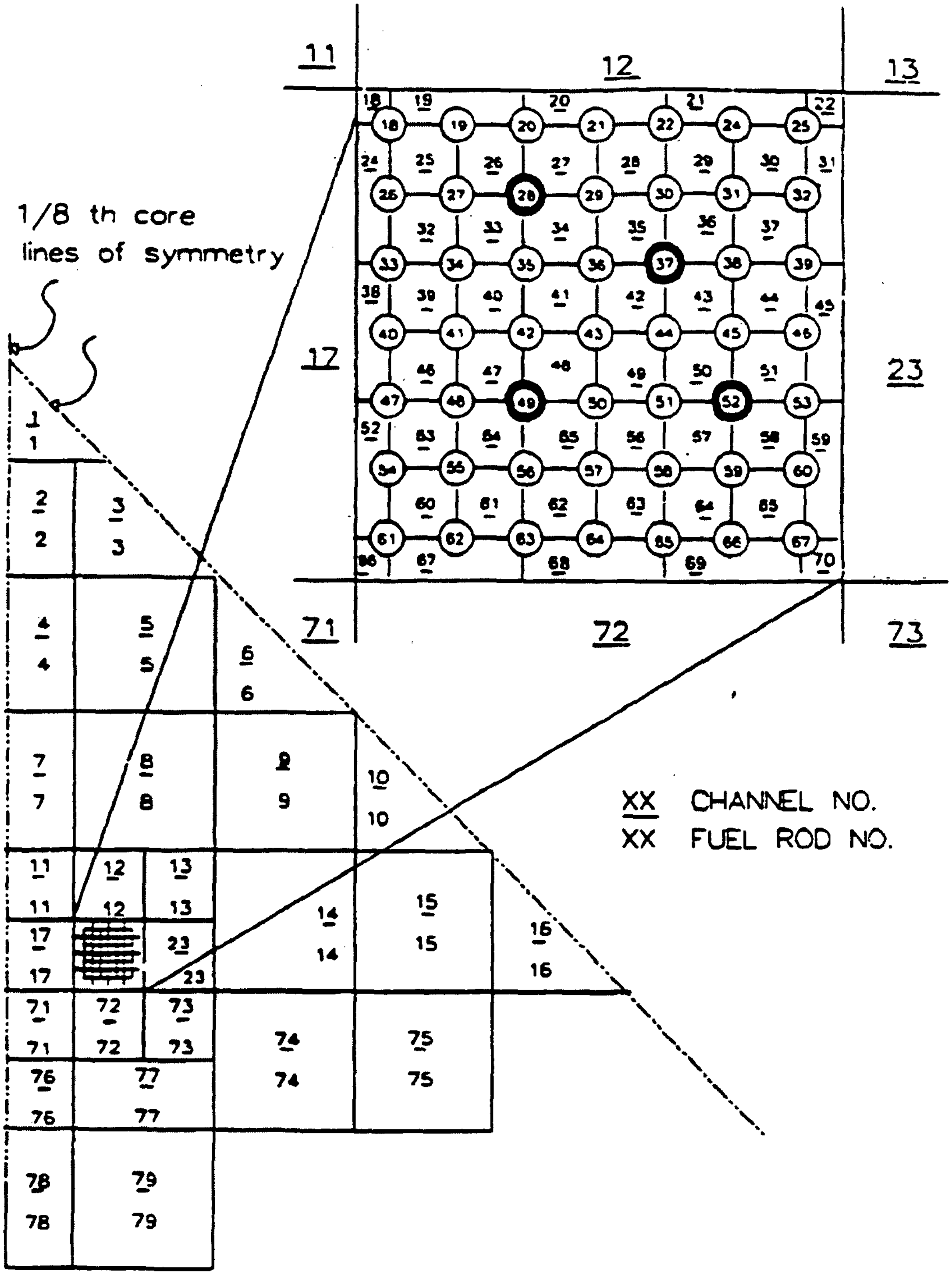

Figure 1. VIPRE-01 input model for Prairie Island assuming 1/8th core symmetry.

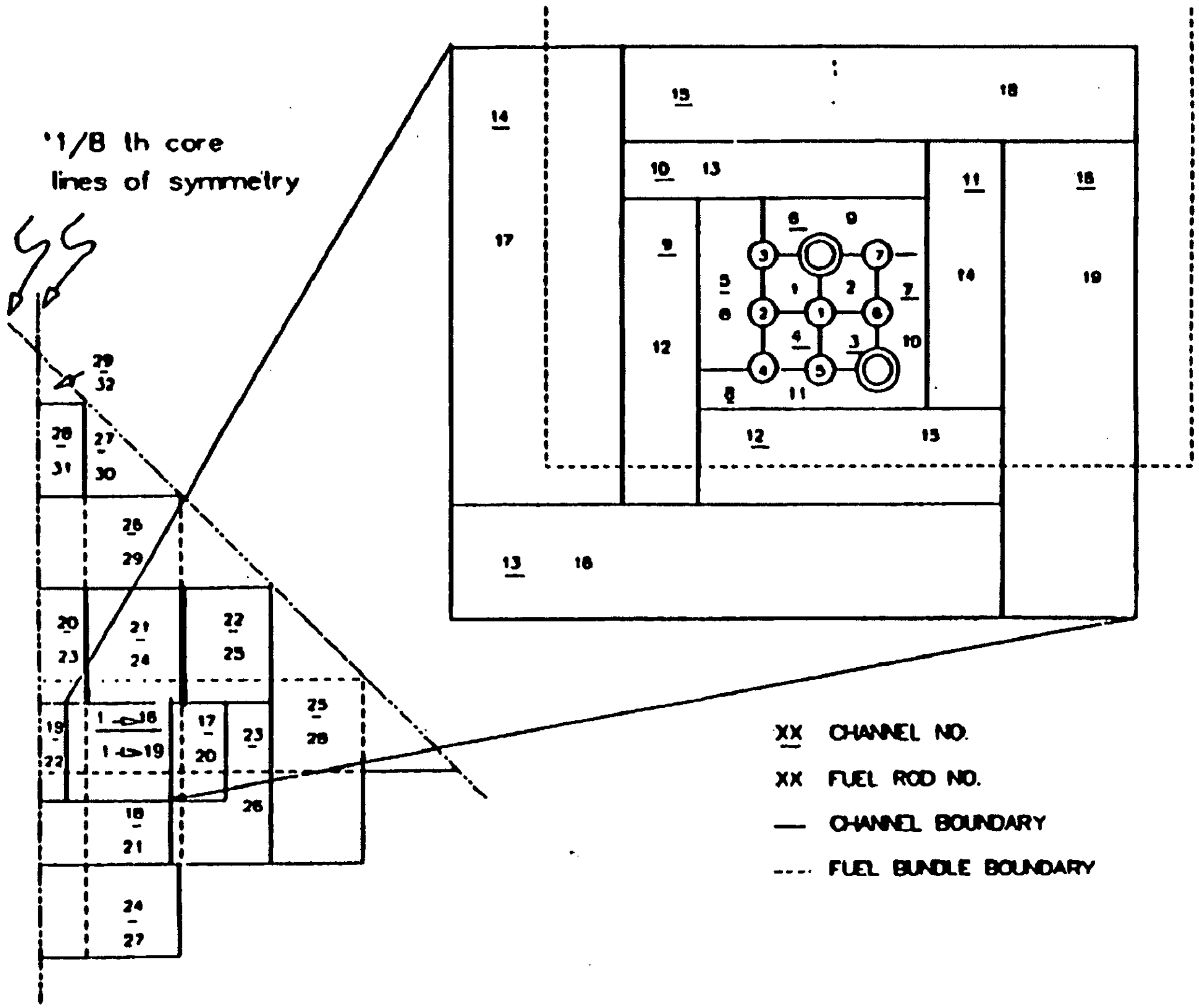

Figure 2. COBRA-IV input model for Prairie Island assuming $1/8$th core symmetry.

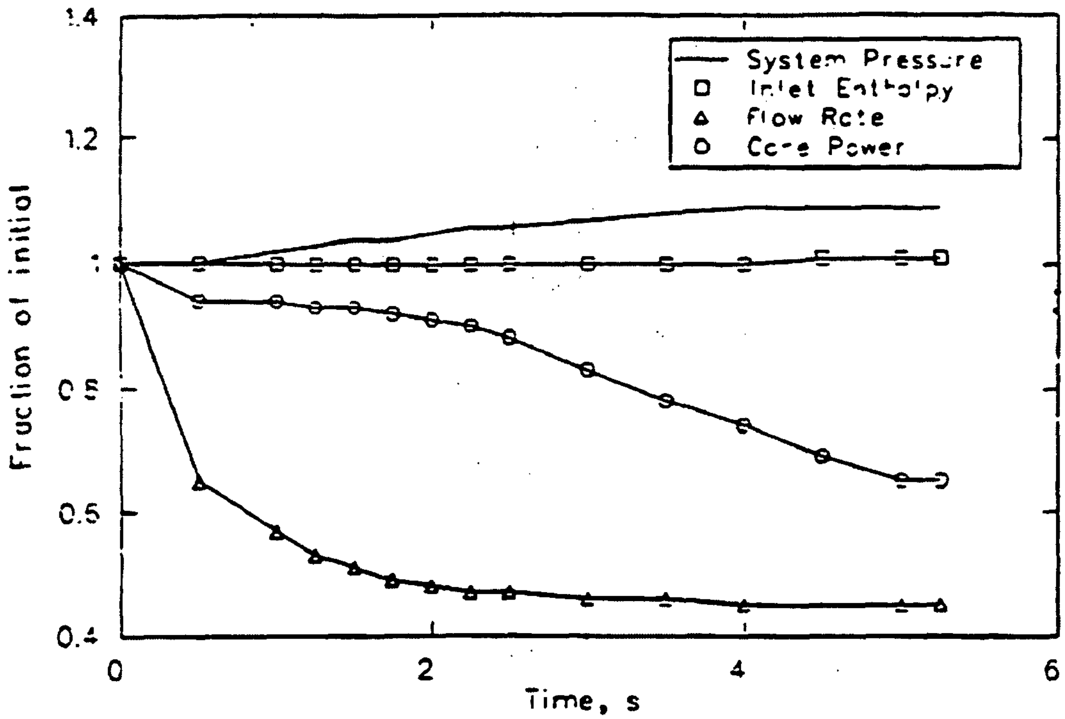

Figure 3. Forcing functions input to the Prairie Island locked rotor calculations.

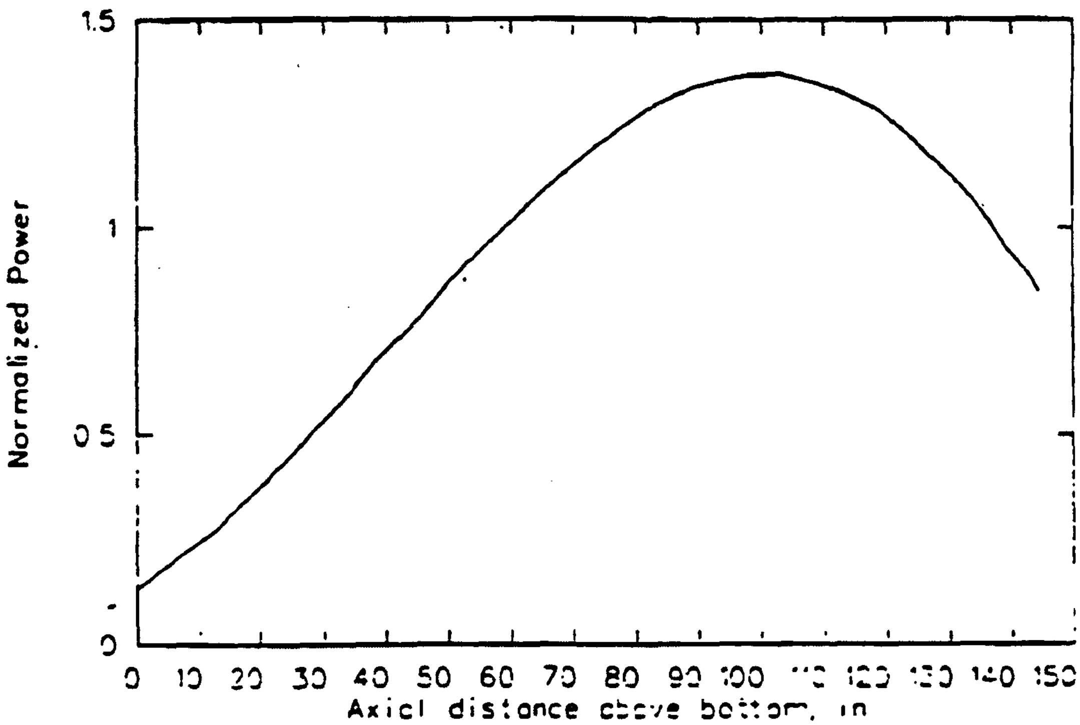

Figure 4. Axial power distribution input to the Prairie Island locked rotor calculations.

TABLE 1. FUEL ROD POWER AND WEIGHTING FACTORS INPUT TO VIPRE-01 AND COBRA-IV

VIPRE-01			VIPRE-01 (continued)			COBRA-IV		
Rod	Power	Weight	Rod	Power	Weight	Rod	Power	Weight
1	0.847	22.38	39	1.477	1.00	1	1.597[a]	1.00
2	1.100	89.50	40	1.510	1.00	2	1.532	1.00
3	1.004	89.50	41	1.495	1.00	3	1.548	1.00
4	0.946	89.50	42	1.520	1.00	4	1.520	1.00
5	1.139	179.00	43	1.532	1.00	5	1.528	1.00
6	0.951	89.50	44	1.597[a]	1.00	6	1.545	1.00
7	1.240	89.50	45	1.545	1.00	7	1.523	1.00
8	0.979	179.00	46	1.499	1.00	8	1.526	1.25
9	1.164	179.00	47	1.512	1.00	9	1.504	1.75
10	1.239	89.50	48	1.506	1.00	10	1.501	1.75
11	1.000	44.75	50	1.520	1.00	11	1.500	1.50
12	1.294	44.75	51	1.528	1.00	12	1.475	11.25
13	1.291	44.75	53	1.513	1.00	13	1.433	11.50
14	0.941	179.00	54	1.509	1.00	14	1.326	11.50
15	0.973	179.00	55	1.492	1.00	15	1.387	12.00
16	0.628	89.50	56	1.504	1.00	16	1.248	44.00
17	1.015	44.75	57	1.493	1.00	17	1.074	44.50
18	1.509	1.00	58	1.497	1.00	18	1.287	42.00
19	1.490	1.00	59	1.513	1.00	19	1.254	42.50
20	1.505	1.00	60	1.493	1.00	20	1.029	126.50
21	1.492	1.00	61	1.524	1.00	21	1.154	168.70
22	1.472	1.00	62	1.506	1.00	22	1.081	51.00
23	1.287	44.75	63	1.507	1.00	23	1.197	106.50
24	1.479	1.00	64	1.505	1.00	24	1.022	213.00
25	1.519	1.00	65	1.506	1.00	25	1.123	213.00
26	1.613	1.00	66	1.505	1.00	26	1.032	235.75
27	1.511	1.00	67	1.506	1.00	27	0.607	270.00
29	1.539	1.00	71	1.289	44.75	29	0.873	540.00
30	1.520	1.00	72	1.241	44.75	29	1.038	360.00
31	1.496	1.00	73	1.218	44.75	30	0.999	90.00
32	1.482	1.00	74	1.079	179.00	31	1.094	90.00
33	1.511	1.00	75	0.728	179.00	32	0.842	22.38
34	1.498	1.00	76	1.222	44.75			
35	1.526	1.00	77	1.074	89.50			
36	1.548	1.00	78	0.874	89.50			
38	1.523	1.00	79	0.478	179.00			

a. Hot rod.

NOTE: VIPRE-01 rod numbers representing rods with zero power, eq, guide tubes, were not include.

TABLE 2. VIPRE-01 AND COBRA-IV COMPUTER CODE INPUT, USERS' OPTIONS AND SELECTED CORRELATIONS

Input	
Hot channel geometry	
Flow area	0.1713 in.^{2}
Heated perimeter	1.316 in.
Wetted perimeter	1.316 in.
Total axial length	152.0 in.
Fuel rod diameter	0.417 in.
No. of axial nodes	40
No. of axial locations for spacer grid	7
Reactor operating conditions	
Ave. heat flux	0.2019 X 10^6 Btu/hr-ft^2
System pressure	2220.0 psia
Inlet coolant enthalpy	534.2 Btu/lb
Inlet coolant flow	2.377 X 10^6 lb/hr-ft^2
Power deposited directly in coolant	2.6%
Gap resistance factor	0.15
Turbulent momentum factor	1.0
Turbulent friction coefficients[a]	AA = 0.84, BB = -0.2, CC = 0.0
User Options	
Subcooled void model[b]	Levy
Fuel rod heat conduction	None
Turbulent mixing	None
Two-phase friction multiplier	Homogeneous
Solution scheme	
VIPRE-1	RECIRC
COBRA-IV	Explicit
Selected Correlations	
Void fraction	Homogeneous
CHF	W-3

a. Friction factors determined by $f = AA\ Re^{BB} + CC$.

b. The Levy correlation is not available in the explicit COBRA-IV.

TABLE 3. COMPARISON OF STEADY STATE DATA CALCULATED BY VIPRE-01 AND COBRA-IV FOR THE PRAIRIE ISLAND PLANT

Variable, Units	VIPRE-01	COBRA-IV
Hot channel inlet mass flux[a], Mlb/hr-ft²	2.430	2.444
Hot channel max enthalphy, Btu/lb	673.4	678.8
Hot channel pressure drop, psi	25.97	25.86
Peak heat flux, MBtu/hr-ft²	.4682	.4680
Axial node at peak heat flux	27	27
MDNBR	2.052	2.120
Equilibrium quality at the MDNBR location	-.112	-.111
Coolant mass flux at MDNBR, Mlb/hr-ft²	2.254	2.155
Subchannel at MDNBR[b]	42	1
Axial node at MDNBR	33	33

a. The hot channel is defined as that channel with the greatest coolant enthalpy. Thus, VIPRE-01 subchannel 49 and COBRA-IV subchannel 4 are the hot channels.

b. VIPRE-01 subchannel 42 and COBRA-IV Subchannel 1 actually represent the same core subchannel.

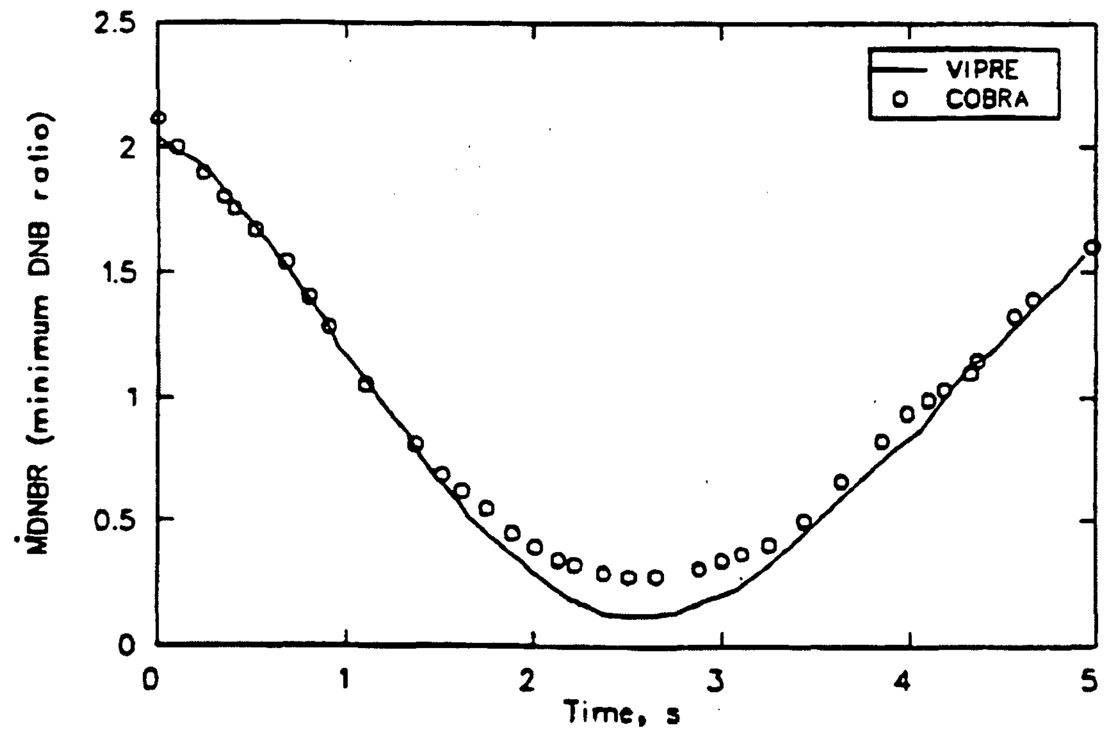

Figure 5. MDNBR values calculated by VIPRE-01 and COBRA-IV for the Prairie Island locked rotor transient.

2.6 Quality Control

Since VIPRE-01 is a new code, developed by one organization (Battelle Pacific Northwest Laboratories) and used by several others for various applications, it is anticipated that the code will undergo numerous changes and corrections especially during the initial phases as a production code. The staff raised the question on the quality assurance program to assure consistency in application of VIPRE-01 for safety analysis. Specifically, the questions are (1) what steps will be taken to assure that only the approved version of VIPRE-01 will be used by UGRA? and (2) how will future modification to VIPRE-01 be implemented? The UGRA's response to these questions is repeated below.

"VIPRE-01 MOD-01, which will be the approved version of VIPRE, has been transmitted to the utilities. VIPRE prints out its version identification on each page of output. If a user makes a modification to the code, they are required, by licensing agreement with EPRI, to change the version identification so that it is clear that their version differs from the approved one.

It is intended that future modifications to VIPRE-01 MOD-01, will be limited to error corrections. Based on the number and severity of the errors reported, new versions will be created by Battelle from time to time under a formal quality assurance procedure (FATE-84-100, Rev. 2, December 1984). The new version will be a new MOD to VIPRE-01 (the version now being reviewed is VIPRE-01 MOD-01, so the next version will be VIPRE-01 MOD-02, etc.). Users are notified of errors reported between release of new versions through a procedure established between Battelle and UCCEL.

Upon EPRI approval, Battelle delivers new VIPRE versions in the form of magnetic tapes to EPRI's code distribution contractor, UCCEL Corporation, who confirms the information on the tape delivered by Battelle through their own quality assurance procedure (UCCEL Corporation Supplemental Policy and Procedures for Distribution of EPRI Software, Rev. 4, March

1985). UCCEL then creates copies of the tapes, and distributes them to licensed users."

We find this quality assurance procedure to be acceptable.

3.0 Summary and Conclusion

The staff has reviewed the VIPRE-01 topical report submitted by UGRA. The review was limited to PWR applications with heat transfer regimes up to critical heat flux. The review consisted primarily of an evaluation of the internal program including the governing conservation equations and constitutive equations, the two-phase flow and heat transfer models, and the numerical solutions techniques. We have also reviewed the VIPRE-01 verification and qualification calculations, sensitivity studies on the user's input options and code defaults. In addition, an audit calculation was performed with a locked rotor transient using COBRA-IV to compare with the VIPRE-01 calculation using the RECIRC solution. Based on this review effort we conclude that the VIPRE-01 computer code is acceptable for PWR licensing calculations subject to the following conditions:

(1) The application of VIPRE-01 is limited to PWR licensing calculations with heat transfer regime up to CHF. Any use of VIPRE-01 in BWR calculations or post CHF calculations will require prior NRC review and approval.

(2) Use of a steady state CHF correlation with VIPRE-01 is acceptable for reactor transient analysis provided that the CHF correlation and its DNBR limit have been reviewed and approved by NRC and that the application is within the range of applicability of the correlation including fuel assembly geometry, spacer grid design, pressure, coolant mass velocity, quality, etc. Use of any CHF correlation which has not been approved will require the submittal of a separate topical report for staff review and approval. The use of a CHF correlation which has been previously approved for application in connection with another thermal hydraulic code other than VIPRE-01 will require an analysis showing that, given the correlation

data base, VIPRE-01 gives the same or a conservative safety limit, or a new higher DNBR limit must be used, based on the analysis results.

(3) Each organization using VIPRE-01 for licensing calculations should submit separate documentation describing how they intend to use VIPRE-01 and providing justification for their specific modeling assumptions, choice of particular two-phase flow models and correlations, heat transfer correlations, CHF correlation and DNBR limit, input values of plant specific data such as turbulent mixing coefficient, slip ratio, grid loss coefficient, etc., including defaults.

(4) If a profile fit subcooled boiling model (such as Levy and EPRI models) which was developed based on steady state data is used in boiling transients, care should be taken in the time step size used for transient analysis to avoid the Courant number less than 1.

(5) The VIPRE-01 user should abide by the quality assurance procedures described in Section 2.6 of this report.

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