Attachment A

Non-Proprietary Version

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DUKE POWER COMPANY

OCONEE NUCLEAR STATION

CORE THERMAL-HYDRAULIC METHODOLOGY

USING VIPRE-01

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K. B. Jones
D. R. Koontz

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Duke Power Company
Design Engineering Department
Nuclear Engineering Section
Charlotte, North Carolina

ABSTRACT

This report presents Duke Power Company's methodology for using VIPRE-01 for performing thermal-hydraulic analyses in support of Oconee Nuclear Station licensing activities. The VIPRE-01 thermal-hydraulic methodology and models are presented along with the results of sensitivity studies used in determining the acceptability of the various input criteria. This report meets the licensing requirements addressed in the Safety Evaluation Report for EPRI NP-2511-CCM, VIPRE-01, ref. 3.

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Revision 1 updates Revision 0 to incorporate subsequent approved methodologies. Revision 1 includes no unapproved technical changes.

DPC-NE-2003P-A describes Duke Power Company's methodology for using VIPRE-01 to perform steady-state core thermal-hydraulic analyses for the Oconee Nuclear Station. The original report describes the Mark-BZ fuel design, but the report also states that the VIPRE-01 models will be used to predict and evaluate the thermal-hydraulic effects of future fuel assembly designs. The Mark-B11 design (Reference 12) and VIPRE-01 input required to model the Mark-B11 design have been reviewed and approved as documented in DPC-NE-2005P-A, Revision 2.

The original report also discusses how design and modeling uncertainties are conservatively applied and are all assumed to occur simultaneously. Subsequently, the Statistical Core Design (SCD) methodology was developed and approved as documented in DPC-NE-2005P-A. Steady-state core thermal-hydraulic analyses are still performed as discussed in Section 6 of this report, but using the SCD methodology, nominal input parameters are used and the uncertainties are accounted for using the SCD limit. The impact on DNB of the following uncertainties are statistically combined using the SCD methodology given in DPC-NE-2005P-A:

1. Core power
2. RCS pressure
3. RCS temperature
4. RCS flow
5. Radial power ($F\Delta h$)
6. Axial power (F_z)
7. Axial peak location (z)
8. Hot Channel Flow Area

9. Rod power hot channel factor (Fq)

10. CHF correlation

11. VIPRE-01 model

For non-SCD analyses, uncertainties are still applied directly as discussed in Revision 0. Section 5.11 was updated to clarify the hot channel factors that were used for Oconee SCD and non-SCD analyses, reviewed and approved in DPC-NE-2005P-A, Revision 2.

Also, included in this revision is a rewritten Section 6.6, which discusses the DNB transient which is the basis for the Operational Maximum Allowable Peaking (MAP) limits. The approved topical report, DPC-NE-3005P-A, provides a detailed discussion of the transient analyses methodology. The two pump coastdown transient is analyzed using the clad surface heat flux versus time calculated by RETRAN-02. Thus, the VIPRE-01 conduction model as described in Revision 0 of this report is no longer used when calculating the Operational MAP limit statepoint. Pages 29-30 of the original report, which described the conduction model input used to calculate the transient DNB response, have been deleted.

This revision modified Section 6.4.3 to reflect that the generic analyses flow value is now listed in the Core Operating Limits Report (COLR) for each cycle.

The reference axial peaking value and the BWC correlation DNBR limit with VIPRE-01 were revised to be consistent with the current values used in the MAP methodology as discussed in the letter dated June 19,1989 from H. B. Tucker (Duke Power Company) to the USNRC, enclosed in Appendix B of this topical.

This revision of the topical is an update to reflect subsequent licensing approvals only. Revisions are denoted with a revision bar and a 1 in the right hand margin. The revisions with the bar and a 0 in the right hand margin were included in the original approved report, SER dated July 1989.

1.0 INTRODUCTION

Duke Power Company's Oconee Nuclear Station reactor core thermal-hydraulic design and licensing analyses have traditionally used very conservative methods to establish the maximum permissible core power and power distribution for various combinations of core outlet pressure and reactor outlet temperature to ensure that DNBR criteria are met. Conservative "closed-channel" computer codes have been used for Oconee Nuclear Station thermal-hydraulic analyses using the methodology described in reference 1. Crossflow computer codes which can predict flow redistribution effects within an open lattice reactor core, can more realistically predict the local fluid properties and thus, the departure from nucleate boiling ratio (DNBR) in the hot channels of the core.

This report presents the procedure used to apply the VIPRE-01 computer code for thermal-hydraulic analyses of Oconee reactor cores and fulfills the requirements addressed in the SER for using VIPRE-01 for licensing analyses, ref. 3. The geometric representation of the core is illustrated and discussed along with the models and empirical correlations used to determine friction pressure losses, coolant mixing and subcooled voids. Descriptions of the methodology used to determine the thermal-hydraulic limits which define the regions of safe operation in terms of power level, reactor coolant temperature and pressure, and power distribution are included in this report. The Oconee thermal-hydraulic analyses will continue to treat uncertainties, tolerances, and measurement errors conservatively. The methodology used to perform generic Oconee thermal-hydraulic analyses is discussed in the report. The need to perform the thermal-hydraulic analyses in conjunction with a reload arises

when there is a change in the fuel assembly design, a change in input assumptions of the generic analysis, or a change in the regulatory criteria.

2.0 CODE DESCRIPTION

VIPRE-01, ref. 2, is an open channel, homogeneous equilibrium, thermal-hydraulic code which features diversion crossflow and turbulent mixing to calculate the departure from nucleate boiling ratios (DNBRs). The code accepts input data which defines the geometric, hydraulic and thermal characteristics of the core, and permits the user to select correlations and solution methodologies.

Generally, core representation is made by inputting parameters defining and describing the number of channels and subchannels within the model and their individual channel and subchannel characteristics, such as flow area, wetted and heated perimeters, adjacent channel data, and centroidal distances between adjacent channels. Hydraulics of the code are defined by crossflow resistances determined from gap dimensions through which the channels communicate, spacer grid locations and form loss coefficients, mixing coefficients, two-phase flow correlations, friction pressure losses, and inlet flow distributions. Thermal modeling of the reactor core is a function of the core radial and axial power distribution, core power, operating conditions, hot channel factors, heat transfer correlations and correlation limits. VIPRE-01 was designed to perform steady-state and transient thermal-hydraulic analyses of nuclear reactor cores for normal operating conditions and several accident conditions. The VIPRE-01 code has been reviewed by the NRC and was found to be acceptable for referencing in licensing applications with the limitations addressed in ref. 3.

3.0 STATION DESCRIPTION

Oconee Nuclear Station consists of three, Babcock & Wilcox (B&W) designed, pressurized water reactors with each reactor rated at 2568 Mwt. Each reactor core consists of 177 fuel assemblies with each assembly having 208 fuel rods, 16 control rod guide tubes, and an instrument tube arranged into a 15 x 15 array. For Mark-BZ fuel, eight non-mixing vane spacer grids provide lateral stiffness and fuel rod positioning. For Mark-B11 fuel, 2 non-mixing vane spacer grids (upper and lower), 1 intermediate non-mixing vane spacer grid, and 5 mixing vane spacer grids provide lateral stiffness and fuel rod positioning. Typical dimensions and characteristics of the Mark-BZ and Mark-B11 (Reference 12) fuel designs are given in Table 3-1 and Table D-1 in Reference 11, respectively. 1

4.0 CORE MODELING

Traditionally, core thermal-hydraulic analyses have been performed using multi-pass analyses. In a mulit-pass analysis, fuel assemblies and the subchannels of the hot assembly are modeled in separate simulations and sometimes in different computer codes. A more direct approach involves only a single-pass. In a single-pass analysis, the hot subchannel and adjacent subchannels are modeled individually with larger and larger channels modeled toward the periphery of the core; the result is that all thermal-hydraulic DNB calculations can be performed using one code. VIPRE-01 has the capability to perform single-pass analyses.

An Oconee Nuclear Station reactor core is geometrically modeled using eighth-core symmetry with the center assembly modeled as the "hot" assembly, Figure 4-1. The hot assembly is the assembly in which the minimum DNBR (MDNBR) can be expected to occur. The hot assembly is divided into subchannels with boundaries formed by fuel rods and guide tubes within the assembly, Figure 4-2.

The hot assembly contains the "hot" subchannel (i.e., the subchannel which yields the MDNBR of the core). To conservatively determine the MDNBR for the core, the models use a high, relatively flat radial pin power distribution along with the application of hot subchannel factors and reduced hot subchannel flow area. The derivation and application of these factors will be discussed in more detail in Sections 5.11 and 5.12.

Selection of single-pass models for performing thermal-hydraulic analyses requires the development of different size models and comparisons of the different models at various operating conditions. Three different size models were developed and compared for modeling Oconee Nuclear Station Fuel:

64 Channel Model

9 Channel Model

8 Channel Model

All three models were developed assuming eighth-core symmetry. The 64 channel model consists of 36 subchannels making up the hot assembly with the remaining 28 channels individually modeling the rest of the assemblies in the eighth-core segment. The 64 channel model is depicted in Figures 4-1 and 4-2. The 8 and 9 channel models were formed by including two rows of subchannels around the hot subchannel (Channel 1) and lumping the rest of the hot assembly into one channel (Channel 7), Figure 4-3. The remaining 28 assemblies were either lumped into one large channel, Channel 8, in the case of the 8 channel model,

Figure 4-4., or into two large channels, Channels 8 and 9, in the case of the 9 channel model, Figure 4-5. The 64 and 8 channel models were compared to confirm the accuracy of the 8 channel model which will be used for steady-state and two-pump coastdown transient analyses. The 9 channel model will be used to evaluate potential transition core effects of differing fuel assembly types. As Table 3-1 shows, the different fuel assembly designs only incorporate minor changes in the basic Mark-BZ fuel assembly designs listed in Table 3-1; moreover, the 64, 9 and 8 channel models will be used to predict and evaluate the thermal-hydraulic effects for future fuel assembly designs.

For illustrative purposes, the number of assemblies lumped together to form Channels 8 and 9 of the 9 channel model, Figure 4-5, was based on the Oconee Unit 1, Cycle 11 core. The number of assemblies lumped together may vary with the cycle specific core configurations being evaluated. The assemblies are arranged in a manner which will give conservative DNBR results.

To determine the modeling detail required to accurately evaluate the hot channel local coolant conditions and the minimum departure from nucleate boiling ratio (MDNBR), the 64, 9 and 8 channel models were run using the operating conditions stated in Table 4-1. The RECIRC numerical solution option was chosen to calculate the results. The VIPRE-01 SER, ref. 3, pg. 17 states that the RECIRC numerical solution is acceptable for licensing calculations. The first two operating conditions, Case 1 and 2, correspond to the high temperature and the low pressure safety limits associated with the Reactor Protection System, ref. 1. The case 4 operating conditions correspond

to the initial conditions for the two pump coastdown transient. The case 3 operating conditions correspond to the operating conditions occuring at the limiting MDNBR during the two pump coastdown transient (i.e., the limiting statepoint in Figure 6-6). The Case 3 operating conditions are used to develop the maximum allowable pin peaks discussed in Section 6.5. Additional details of the Reactor Protection System will be discussed in Section 6.

4.1 STEADY-STATE SINGLE PASS MODEL COMPARISONS

The 64 and 8 channel model results are compared in Table 4-2. Results for the Case 1 operating conditions show that the 8 channel model conservatively predicted the MDNBR by 1.2% when compared to the 64 channel model MDNBR. Results for the Case 2 operating conditions showed the 8 channel model exhibited a 0.44% conservative difference in MDNBR when compared to the 64 channel model. Likewise, the 8 channel model exhibited a 2.2% conservative change in MDNBR for Case 3.

4.2 TRANSIENT MODEL COMPARISONS

The two pump coastdown transient is the most limiting DNB transient; therefore, the two pump coastdown transient was chosen to make a comparative study between the 64 and 8 channel models. The development of the transient modeling details are presented in Section 6.6. The transients were performed using the initial operating conditions from Table 4-1, Case 4. The 64 and 8 channel model transient results are presented in Table 4-3. Throughout the transient, the 8 channel model produced conservative MDNBRS in comparison to the 64 channel

model. The limiting MDNBR observed for the 8 channel model occurred at 4.1 seconds where the MDNBR = 1.216 (i.e., which is conservative in comparison to the 64 channel model MDNBR of 1.234 also occurring at 4.1 seconds).

4.3 TRANSITION CORE MODEL COMPARISONS

As mentioned earlier in Section 1.0, a thermal-hydraulic analysis must be performed whenever there is a change in the fuel assembly design, a change in input assumptions of the generic analysis, or a change in the regulatory criteria. Combinations of different fuel assembly designs in a reactor core constitutes a mixed (transition) core which must be evaluated to determine its effect on thermal-hydraulic performance. Transition core effects are determined by comparing results of a thermal-hydraulic (T-H) analysis explicitly modeling the mixed core with that of a T-H analysis for a non-mixed core. If the comparison shows the MDNBR is adversely affected, then a penalty must be assigned to that particular operating cycle.

The 64 channel and 9 channel transition core models were compared on a steady-state and transient basis to ascertain the accuracy of the 9 channel model. Table 4-4 presents a comparison of the 64 and 9 channel models steady-state results. In all cases, the 9 channel model produced conservative results. The steady-state runs using the Case 3 operating conditions produced the lowest MDNBRs, with the 9 channel model predicting MDNBRs 1.9% more conservative than the 64 channel model. A comparison of the 64 channel and 9 channel model two pump coastdown transient results using the Case 4 initial operating

conditions revealed that the 9 channel model again produced conservative MNDBRs (see Table 4-5); therefore, the 9 channel model will be used to assess any future Oconee Nuclear Station reloads involving transition cores.

4.4 RESULTS SUMMARY

In all of the studies and comparisons performed between the 64, 9, and and 8 channel models, the 9 and 8 channel models consistently produced conservative results. Duke Power Company will use the smaller channel models to perform thermal-hydraulic analyses since the 64 channel model requires an extensive amount of computer processing time. The 64 channel model will be used if the situation arises which requires the 1-2% conservatism currently available with the 8 an 9 channel models. The larger model would only be used for cycle specific evaluations requiring the additional margin.

5.0 VIPRE-01 DATA

The fuel assembly data used to develop each of the input parameters, such as flow area, wetted and heated perimeters, centroid distances, and gap widths are given in Table 3-1 for Mark-BZ fuel. Other important VIPRE-01 input is discussed in detail in the subsections, which follow. These discussions are based on non-SCD methodology per Revision 0 of this report and the Mark-BZ fuel design unless otherwise noted.

For non-Statistical Core Design (non-SCD) analyses, uncertainties are still applied directly as discussed in Revision 0 of this report. For SCD analyses, nominal input parameters are used and the uncertainties are accounted for using the Statistical Core Design (SCD) limit per Reference 11.

The following uncertainties are statistically combined using the SCD methodology given in Reference 11:

1. Core power
2. RCS pressure
3. RCS temperature
4. RCS flow
5. Radial power ($F\Delta h$)
6. Axial power (Fz)
7. Axial peak location (z)
8. Hot Channel Flow Area
9. Rod power hot channel factor (Fq)
10. CHF correlation
11. VIPRE-01 model

The SCD methodology is applied to both the Mark-BZ and Mark-B11 fuel designs.

The Mark-B11 fuel assembly design input parameters, VIPRE-01 correlations, and the BWU-Z CHF correlation with multiplier used in thermal-hydraulic analyses are discussed in References 11 and 12.

5.1 AXIAL NODING

Given the axial power shape and a specified heated rod length, VIPRE-01 determines the axial power factor for each axial node, ref. 2. The node length determines how well the code approximates the axial power shape, the shorter

the node length, the better the approximation of the curve. Volume 4 of the VIPRE-01 manual states as a general rule that nodes on the order of 2 or 3 inches long are recommended in the region where MDNBR is likely to occur, ref. 2. Calculations involving node sizes smaller than 2 or 3 inches require more computer processing time without gaining significant increases in the accuracy of results.

Results of an axial node length sensitivity study performed with the 8 channel steady-state model are presented in Table 5-1. A comparison was made between a three-inch node length, uniformly applied to the axial length of the rod from 4.125 to 142.125 inches, and two ranges of two-inch axial node lengths applied to the rod at elevations ranging from 32.125 to 94.125 inches and 81.125 to 143.125 inches. As Table 5-1 shows, the three inch node lengths produced slightly conservative MDNBRS; therefore, the three-inch node length will be used for all Oconee Nuclear Station thermal-hydraulic analyses.

5.2 ACTIVE FUEL LENGTH

Uranium fuel both densifies and swells when irradiated. Densification effects are predominant at low burnup and swelling effects are predominant at higher burnup. Fuel densification decreases the active fuel length while fuel swelling tends to increase the active length. [

]

5.3 CENTROID DISTANCE

The location of each subchannel or channel is defined by numbering all the channels, inputting connecting channel numbers, and defining the distance between centroids of adjacent channels. The centroidal distance in a normal square array, is the subchannel pitch. The centroidal distance determines the length over which the crossflow exists and defines the lateral pressure gradient in the crossflow momentum equation. The centroidal distance for a channel cut by a line of symmetry is the same as the centroidal distance for the complete channel, ref. 2. For the lumped subchannels, the centroidal distance is increased from its individual subchannel value in proportion to the number of rod rows between channel centroids. Likewise the centroidal distances between lumped assemblies is increased in proportion to the rows of assemblies between the lumped channel centroids.

5.4 EFFECTIVE CROSSFLOW GAPS

Crossflow resistances are calculated by inputting connecting channel information and crossflow gap widths. The product of the gap width and the axial node length defines the lateral flow area between channels. The gap widths are easily calculated given the rod pitches and diameters. The gap width for a fuel assembly or any lumped channel is the sum of the subchannel gaps through which the two assemblies communicate.

5.5 SPACER GRID FORM LOSS COEFFICIENTS

Form loss coefficients are used to account for the unrecoverable pressure losses caused by the abrupt variation in flow area and turbulence at a spacer grid. The Mark-BZ fuel assemblies have six intermediate zircaloy spacer grids and two inconel end grids. Form loss coefficients determined for the different types of subchannels (i.e. unit, thimble tube, peripheral, instrument guide tube, and corner channels) and for the overall grid are used in the thermal-hydraulic analyses. Spacer grid form loss coefficients are developed from full size fuel assembly flow tests performed by the vendor. Individual subchannel form loss coefficients are determined analytically by the vendor from the overall grid form loss coefficients.

5.6 CORE BYPASS FLOW

Core flow is equal to the total reactor coolant system flow less the bypass flow, which is defined as that part of the flow which does not contact the effective heat transfer surface area. The bypass flow paths are the 1) core shroud, 2) core barrel annulus, 3) control rod guide tubes and instrument tubes, and 4) all interfaces separating the inlet and outlet regions of the reactor vessel. A typical value of the design bypass flow is 9.0%; however, the bypass flow rate is dependent on the number of control rod and burnable poison rod assemblies in the core since they act as guide tube plugging devices. The actual core bypass flow must be verified each cycle to assure that it is less than that used in the generic analysis.

5.7 INLET FLOW DISTRIBUTION

VIPRE-01 allows the user to specify the core inlet flow maldistribution. The Oconee core thermal-hydraulic analyses include a 5% reduction in inlet flow to the hot assembly to conservatively represent the results obtained in B&W's 1/6-scale Vessel Model Flow Test, ref 1. More restrictive flow maldistribution factors are used for operation with less than four reactor coolant pumps. Table 5-2 shows that the use of a 5% inlet flow maldistribution produces slightly conservative results compared with a uniform inlet flow distribution.

5.8 VIPRE-01 CORRELATIONS

Empirical correlations are used in the VIPRE-01 code to model turbulent mixing and the effects of two-phase flow on friction pressure losses, non-equilibrium subcooled boiling, and the relationship between the quality and void fraction. The correlations which have been selected for use in the Oconee thermal-hydraulic analyses are discussed in the subsections which follow.

5.8.1 FRICTION PRESSURE LOSS

Pressure losses due to frictional drag are calculated for flow in both the axial and lateral directions. In the axial direction the friction pressure loss is calculated by

$$\frac{dP}{dZ} = f \frac{G^2 v'}{2g_c D_h}$$

where f = friction factor determined from an empirical correlation defined by user input

G = Mass flux, lbm/sec-ft^2

v' = specific volume for momentum, ft.^{3}/lbm

g_c = force-to-mass units conversion factor, 32.2 lbm-ft/lbf-sec^2

D_h = hydraulic diameter based on wetted perimeter, ft.

Based on the recommendation in ref. 2, vol. 4 of the VIPRE-01 manual, the default Blasius smooth tube friction factor expression

$$f = 0.32\ Re^{-0.25} + 0.0$$

will be used to calculate the friction pressure loss for turbulent flow. Based on sensitivity study results given in Table 5-3, the friction pressure loss for two-phase flow will be calculated using the EPRI two-phase friction multiplier.

In the lateral direction the pressure loss is treated as a form drag loss that is calculated by

$$\Delta P = K_G \frac{|w| w\ v'}{2S^2 g_c}$$

where K_G = loss coefficient in the gap between adjacent channels

w = crossflow through a gap, lbm/sec-ft

v' = specific volume for momentum, ft ^{3}/lbm

S = gap width, ft

g_c = force-to-mass units conversion factor, 32.2 $\frac{\text{lbm-ft}}{\text{lbf/sec}^2}$

When rod arrays are modeled as lumped channels the effective crossflow resistance is the sum of the resistance of the rod rows between the lumped channel centroids. The lateral loss coefficient becomes

$$K_{ij} = N\ K_G$$

where N is the number of rod rows between lumped channels and K_G is the nominal drag coefficient for a single gap. Crossflow resistance coefficients are not precisely known, but sensitivity study results discussed in Volume 4 of ref. 2 show that for applications where the axial flow is predominant relative to crossflow, crossflow resistance has an insignificant effect on mass flux and DNBR. A subchannel drag coefficient, K_G, of 0.5 will be used with the coefficient for lumped channels calculated internally by the code based on the input centroid distances between lumped channels and the standard subchannel fuel rod pitch.

5.8.2 TURBULENT MIXING

The VIPRE-01 transverse momentum equation includes terms to calculate the exchange of momentum between adjacent channels due to turbulent mixing. Two parameters must be input to include turbulent mixing: a turbulent momentum factor (FTM) and a turbulent mixing coefficient (β).

The turbulent momentum factor (FTM) defines how efficiently the turbulent crossflow mixes momentum. FTM can be input on a scale from 0.0 to 1.0, where 0.0 indicates that the crossflow mixes enthalpy only and 1.0 indicates that it

mixes enthalpy and momentum with the same strength. In actuality, some proportion of enthalpy and momentum mixing does take place; therefore, turbulent momentum factors of 0.8 and 1.0 are probably more representative of actual crossflow effects. Sensitivity studies discussed in Vol. 4 of ref. 2 show that changes in the fraction of momentum mixing have negligible impact on the flow field; therefore, FTM = 0.8 is recommended, ref. 2. Sensitivity studies using the 8 channel model were performed for the Case 1 and 2 operating conditions given in Table 4-1. The runs were made using FTM = 0.0, 0.8 and 1.0. The results of the analyses are presented in Table 5-4. Since the results show that MDNBRs for an FTM = 0.8 lie between the MDNBRs for FTM = 0.0 and 1.0, and since FTM = 0.8 more realistically assumes some momentum mixing, an FTM = 0.8 will be used in all future Oconee thermal-hydraulic analyses.

Turbulent crossflow between adjacent channels is calculated by

$$w' = \beta S \bar{G}$$

where w' is the turbulent flow per axial length, β is the turbulent mixing coefficient, S is the gap width, and $\bar{G}$ is the average mass flux of the adjacent channels. Based upon vendor predictions of mixing test results, a mixing coefficient of [ ] will be used for all Oconee Nuclear Station core thermal-hydraulic analyses.

5.8.3 TWO-PHASE FLOW CORRELATIONS

Two correlations are used in VIPRE-01 to make two-phase flow predictions. The first correlation is referred to as the subcooled void correlation. It uses a

quality model to calculate the flowing vapor mass fraction including the effects of subcooled boiling. Once the flowing vapor mass fraction is calculated, the bulk void correlation is applied to calculate the void fraction including any effects due to slip, ref. 2, Vol. 1.

Sensitivity studies were performed using three different combinations of subcooled void and bulk void correlations to evaluate their effects on the hot channel local coolant conditions and MDNBR.

Subcooled Void	Bulk Void
Levy	Zuber-Findlay
Levy	Smith
EPRI	EPRI

The hot channel local coolant conditions and MDNBRs are given in Table 5-3 for the Case 1 and 2 operating conditions. As Table 5-3 shows, the combination of the Levy subcooled void correlation and the Zuber-Findlay bulk void correlation yields slightly conservative results. Section 3.3 of Vol. 4 of the VIPRE-01 manual, ref. 2, presents the results of VIPRE-01 predictions of the Martin void fraction tests at high pressure (1565 and 1991 psia). Of the two-phase flow correlations evaluated, the Levy/Zuber-Findlay combination compared most favorably with the test results. The Levy subcooled void correlation and the Zuber-Findlay correlation will be used for Oconee thermal-hydraulic analyses.

5.9 REFERENCE DESIGN POWER DISTRIBUTION

The reference design power distributions are shown in Figures 4-1 through 4-5. The power distributions were designed to be conservatively high and relatively flat in the vicinity of the hot subchannel. The pin power peaking gradient within the area of the hot subchannel is approximately 1%. The pin power distribution was verified to be conservative by comparison with predicted physics power distributions. The reference design power distribution was developed using a radial-local hot pin peak, $F_{\Delta}H^{N}$, of 1.714 and an assembly power of 1.6147. The $F_{\Delta}H^{N}$ = 1.714 is the same reference pin peak used in the methodology discussed in reference 1. The two pump coastdown transient is analyzed as discussed in Section 6.6 using the reference design power distribution. A different design power distribution may be used to add or delete margin in the transient analysis. As discussed in Section 6.5 and 6.6 maximum allowable peaking (MAP) limits are calculated to define combinations of radial and axial peaking that provide equivalent DNB protection. 0 1

5.10 AXIAL POWER DISTRIBUTION

The axial power shape used to develop the results presented in this report was a [ ] chopped cosine axial power shape. Predicted and actual axial power shapes vary for cycle specific reloads and transients since they are functions of control rod positions, xenon transients, etc. The effect on DNB of different axial flux shapes is taken into account as discussed in Section 6.5. 1

A routine has been added to the VIPRE-01 code to generate axial power shapes with inlet, symmetric, or outlet peaks. The routine is based on the following constraints on an axial power shape.

F(x) is continuous from (B,E)

F'(x) is continuous from (B,E)

$$\frac{1}{E-B}\int_B^E F(x)dx = 1.0$$

where F(x) = axial power shape as a function of the axial location, x

B,E = beginning and ending normalized location of the active fuel length

The reference 1.50 axial flux is generated using the new axial shape routine.

5.11 HOT CHANNEL FACTOR

The local heat flux factor, F_q'', and the power factor, F_q, are conservatively applied to the hot subchannel (i.e., the instrument guide tube subchannel) of the hot assembly to compensate for possible deviations of several parameters from their design values.

For non-SCD analyses, Fq" is only used in the computation of the surface heat flux of the hot pin when calculating the DNBR in the hot subchannel, ref. 1.

Previously, Fq" included factors/penalties accounting for the (1) effects of local variations in the pellet enrichment and weight on local (hot spot) power, (2) power spikes occurring as a result of flux depressions at spacer grids, and (3) axial nuclear uncertainty. An Fq" factor accounting for all these effects was applied when calculating Maximum Allowable Peaking Limits (MAP) limits. Since References 7 and 8 show that local heat flux spikes have no effect on the critical heat flux results, the first two penalties are not required. Therefore, Fq" is only used to account for axial nuclear uncertainty. For non-SCD analyses, the F_q" for Mark-BZ and Mark-B11 fuel is [ ], ref. 6. For SCD analyses, the axial nuclear uncertainty is accounted for by using the SCD limit.

The power factor, Fq = [ ] ref. 1 (Mark-BZ) and [ ] (Mark-B11) ref. 11, accounts for variations in average pin power caused by differences in the absolute number of grams of U-235 per rod. The loading tolerance on U-235 per fuel stack and variation on the powder hot mean enrichment are considered in determining the factor, ref. 1. F_q is applied to the heat generation rate of the hot pin of the hot subchannel.

5.12 FLOW AREA REDUCTION FACTOR

The hot subchannel flow area is reduced by [ ] to account for variations in as-built subchannel coolant flow areas in non-SCD analyses. The hot channel flow area uncertainty is statistically combined in the SCD methodology per Reference 11.

5.13 BWC CRITICAL HEAT FLUX CORRELATION

The BWC critical heat flux (CHF) correlation, ref. 4, will be used for Oconee

thermal-hydraulic analyses for Mark-B10 and earlier fuel. The BWC correlation was originally developed for B&W 17x17 Mark-C fuel. Subsequently, as discussed in ref. 4, B&W showed that the BWC correlation can be used for 15x15 Zircaloy grid Mark-BZ fuel.

The BWC correlation was developed by B&W using the LYNTX2 crossflow computer code, ref. 5. To justify use of the BWC correlation with the VIPRE-01 code the Zircaloy grid CHF test results given in ref. 4 were predicted using VIPRE-01 and compared with B&W's LYNX2 results. The VIPRE-01/BWC results for all 211 data points were used to determine a DNBR limit which provides a 95% probability of precluding DNB at a 95% confidence level.

Figures 5-1, 2, and 3 show the B&W LYNX2 versus VIPRE-01 calculations for the BWC Measured-to-Predicted (M/P) CHF, mass velocity, and quality at the CHF location, respectively. These figures show that the VIPRE-01 coolant conditions and BWC CHF predictions are essentially the same as B&W's LYNX2 predictions. Figure 5-4 shows the measured CHF versus the VIPRE-01 predicted CHF for all 211 points demonstrating that the overall prediction of the correlation is correct. The ratio of measured-to-predicted CHF is plotted versus local quality, mass velocity, and pressure in Figures 5-5 through 5-7, respectively. These figures show that there is no bias in the correlation relative to important fluid parameters. Calculation of the design DNBR limit is based on the assumption that the M/P CHF values are normally distributed. This was verified statistically using the D-prime test.

A DNBR limit is calculated so that cores can be designed to operate below the CHF. The DNBR limit is the lowest DNBR that can be calculated (for any core condition) for the limiting pins in the core and ensure with 95% confidence that 95% of the limiting pins are not in film boiling. The design DNBR limit was calculated using the following expression developed in ref. 4:

$$\text{DNBR Limit} = \frac{1.0}{M/P - K_{N,\gamma,\rho}\ \sigma}$$

where M/P = mean measured-to-predicted CHF ratio

$K_{N,\gamma,\rho}$ = on-sided tolerance factor based on degrees of freedom (N), confidence level (γ), and portion of population protected (ρ).

σ = standard deviation of measured-to-predicted CHF values

For the VIPRE-01/BWC combination the design DNBR limit is 1.161. The DNBR limit is calculated as shown in the following.

$$N = 211$$

$$M/P = 1.0076$$

$$K_{211,\ 0.95,\ 0.95} = 1.832$$

$$\sigma = 0.0797$$

$$\text{DNBR Limit} = \frac{1.0}{1.0076 - [\ 1.832\ \ (0.0797)\ ]} = 1.161$$

For all Oconee thermal-hydraulic analyses using VIPRE-01 and the BWC correlation, a design DNBR limit of 1.18 + margin will be used.

The applicable range of variables for the BWC correlation are:

Pressure	$1600 < P < 2600$ psia
Mass Velocity	$0.43 < G < 3.8$ Mlbm/hr-ft^2
Quality	$-0.20 < X < +0.26$

6.0 OCONEE THERMAL-HYDRAULIC ANLAYSES

6.1 SUMMARY

A thermal-hydraulic analysis of the Oconee reactor cores is necessary to define the core thermal margin and acceptable operating limits. The crossflow code thermal-hydraulic analysis methods used to derive the core safety and operating limits are the same as the previously approved methods in ref. 1. The safety and operating limits are used to ensure core protection against anticipated transients and steady-state operation. Some of the Reactor Protection System (RPS) trip functions is given in Table 6-1. The safety limits are derived from thermal-hydraulic analyses based upon various combinations of power, pressure, temperature and flux-to-flow limits. A new analysis is performed for a reload core whenever there is a significant change in the fuel design, a change in the input assumptions of the generic analysis, or a change in the regulatory criteria.

6.2 THERMAL-HYDRAULIC DESIGN CRITERION

The thermal-hydraulic design criterion is that no core damage due to DNB occur during steady-state operation or anticipated transients. DNB is defined as the point where bubble generation on the clad heat transfer surface forms an insulating blanket over the surface heating area, thus, causing a large clad surface temperature rise. The departure from nucleate boiling ratio (DNBR) is defined as the ratio of the critical heat flux at a point on the rod to the

actual heat flux at the same point. DNBR is calculated using Babcock and Wilcox's BWC Correlation. The minimum DNBR (MDNBR) is limited to 1.18 + margin as previously explained in Section 5.13. 1

6.3 CORE SAFETY LIMITS

Core safety limits are determined to protect the core during steady-state operation and anticipated transients. The core safety limits prevent overheating and possible rupture of the cladding which would release fission products to the coolant. Fuel clad overheating is prevented by restricting operation to within the nuclear boiling regime where clad temperature is only slightly above the coolant temperature. Two core safety limits directly provide DNB protection:

1.	Pressure - Temperature Envelope	Figure 6-2
2.	Power - Power Imbalance Limits	Figure 6-1

6.4 PRESSURE-TEMPERATURE

The Pressure-Temperature (PT) envelope defines a region of allowable operation in terms of reactor coolant system (RCS) pressure and vessel outlet temperature. The PT envelope provides DNB protection as well as protection for the RCS. The three reactor trips that define the region of allowable operation as shown in Fig. 6-2 are: 0

1. High temperature trip
2. Low pressure trip
3. Variable low pressure trip

To ensure that the PT envelope provides DNB protection, PT curves are determined for [ ] reactor coolant (RC) pump operation. The PT curves are the combinations of RCS pressure and vessel outlet temperature that yield the design DNBR limit (CHF correlation limit plus margin) or the CHF correlation quality limit. The PT envelope must be more restrictive than the most limiting PT curve in Fig. 6-2. 1 0

The PT curves are calculated using the 8 channel model discussed in Section 4.0. The VIPRE-01 input that is used to calculate the generic PT curves is discussed in subsections 6.4.1 through 6.4.4 which follow.

6.4.1 REFERENCE POWER DISTRIBUTION

The reference power distribution discussed in Section 5.8 and shown in Fig. 4-3 and 4-4 is used to calculate the PT curves. The reference axial power profile used to calculate the PT curves is symmetric chopped cosine with a peak to average value of [ ] A different reference axial power shape may be used as necessary to ensure that the MDNBR during a two pump coastdown transient is greater than the design DNBR limit. The axial power shape can change as a result of rod motion, power change, or due to a xenon transient. Power - power imbalance limits, ref. 1, provide protection for the core from the effects of skewed axial power distributions. To determine the power - power imbalance imbalance limits maximum allowable (MAP) limits are calculated as discussed in Section 6.5. 0

6.4.2 CORE POWER

Using non-SCD methods, the maximum power level for 4 pump operation, 112% FP, is set by the high flux trip setpoint with adjustment made for uncertainties 1

and margin. The maximum power level [

] The PT curves are calculated for the maximum power levels for

[ ] pump operation.

6.4.3 RCS Flow

The generic Oconee thermal-hydraulic analyses will be based on the RCS flow value listed in the Core Operating Limit Report (COLR) which is lower than the measured flow for any of the three Oconee units. The value is listed in terms of percent design flow. For example, a value of 107.5% design flow is equal to 378,400 pgm (107.5% of 88,000 gpm/pump). This value could be increased for a cycle specific analysis to take credit for the flow margin at a particular unit.

6.4.4 CORE INLET TEMPERATURE

For a given core power, flow (number of operating RC pumps), and pressure the vessel outlet temperature at which the MDNBR equals the design DNBR limit defines a point along a PT curve. VIPRE-01 is run at several pressures to determine the core inlet temperatures that yield the design DNBR limit.

6.5 GENERIC MAXIMUM ALLOWABLE PEAKING LIMIT CURVES

In order to provide DNB protection for axially asymmetric and symmetric power distributions, a series of maximum allowable pin peaks are calculated such that the MDNBR limit is obtained. Maximum allowable peaking (MAP) limits are calculated in the form of lines of constant MDNBR for a range of axial peaks with the location of the peak varied from the bottom to the top of the core. This is performed for axial peaks of [

] The axial peaks were generated using the new axial shape routine discussed in Section 5.10. The maximum allowable peaks are multiplied by their respective axial peaks to obtain Total Maximum Allowable Peaking Limits (i.e., MAP limits). The MAP Limits are plotted for each axial peak and X/L to form a set of MAP limit curves. The MAP limits provide a basis for equating the symmetric and asymmetric power distributions. MAP limits are compared in a maneuvering analysis with peaks resulting from design power transients as discussed in ref. 1. Two sets of generic MAP Limit curves are determined. One set is used to determine the DNB operational offset limits, and the other set is used to determine the Reactor Protection System (RPS) DNB limits. [

]

Operational MAP limit curves are developed in the same manner as the RPS MAP limits based on the two pump coastdown transient as explained in the following section. A typical set of Operational MAP limit curves generated with the 8 channel model is shown in Figure 6-5.

If any negative peaking margins (predicted peaking greater than the appropriate MAP limit) are determined during a maneuvering analysis, ref. 1, the MDNBR will be calculated for the limiting predicted power distribution. The predicted radial power distribution and axial flux shape is input directly into the VIPRE-01 code. 0

6.6 OPERATIONAL MAP LIMIT GENERATION

Duke Power topical DPC-NE-3005P-A, UFSAR Chapter 15 Transient Analysis Methodology (ref. 9) provides a detailed discussion of the transient analyses methodology. The DNB transients are analyzed using the RETRAN-02, Ref. 10, and VIPRE-01 computer codes. The transient boundary conditions (power, pressure, temperature, and flow) versus time used in VIPRE-01 to calculate DNBR are determined in RETRAN-02.

The reference power distribution is used in VIPRE-01 to determine the mininmum DNBR statepoint. [

]

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7.0 REFERENCES

1. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002P-A, Rev. 2, Duke Power Company, Charlotte, NC, October 1985.

2. J. M. Cuta, et. Al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," EPRI-NP-2511-CCM, Rev. 1, Vol. 1-5, Battelle Pacific Northwest Laboratories, September 1989.

3. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA), "Acceptance for Referencing of Licensing Topical Report, VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores," EPRI-NP-2511-CCM, Vol. 1-5, May 1, 1986.

4. BWC Correlation of Critical Heat Flux, BAW-10143-A, Babcock and Wilcox, Lynchburg, Va. April 1985.

5. LYNX2: Subchannel Thermal-Hydraulic Analysis Program, BAW-10130-A, Babcock and Wilcox, Lynchburg, VA, July 1985.

6. Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004P-A, November 1992.

7. Westinghouse Topical Report WCAP-8202, Effect of Local Heat Flux Spikes on DNB in Non-Uniformly Heated Bundles, K. W. Hill, F. E. Motley, and F. F. Cadek, August 1973.

8. Combustion Engineering Topical Report CENPD-207, CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies With Standard Spacer Grids - Part 2, Non-Uniform Axial Power Distribution, June 1976.

9. Duke Power Company Oconee Nuclear Station UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005P-A, Rev. 1, August 1999.

10. RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Rev. 4, EPRI, November 1988.

11. Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005P-A, Rev. 2, June 1999.

12. BAW-10229P-A, Mark-B11 Fuel Assembly Design Topical Report, Framatome Cogema Fuels, Lynchburg, Va., October 26, 1999.

TABLE 3-1. MARK-BZ FUEL ASSEMLBY DATA
(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, in. (Nom.)	0.430
Thimble tube diameter, in. (Nom.)	0.530
Instrument tube diameter, in. (Nom.)	0.554
Fuel rod pitch, in. (Nom.)	0.5663
Fuel assembly pitch, in. (Nom.)	8.587
Fuel Rod Length, in. (Nom.)	153.7

0

GENERAL FUEL CHARACTERISTICS

	Material	Quantity	Location	Type
Grids:	Inconel	2	Upper and Lower	Non-mixing Vane
	Zircaloy	6	Intermediate	Non-mixing Vane

	Material	Quantity
Fuel rods:	Zircaloy-4	208

Fuel Cycle Design Assembly Features

Fuel Assy. Designation:	Mark B4Z	Mark B5Z	Mark B6	Mark B7
Features:				

TABLE 4-1. OPERATING CONDITIONS

CASE*	Power %	Flow %	Pressure PSIA	Inlet Temperature °F
1	[			]
2				
3				
4				

*All cases were performed using a [ ] axial peak unless otherwise noted.

TABLE 4-2. COMPARISON OF 64 CHANNEL AND 8 CHANNEL MODEL STEADY-STATE RESULTS (TYPICAL)

	MDNBR			MASS VELOCITY (MLBM/HR-FT^2)			EXIT QUALITY		
CASE 1[a]	Ch. 1	Ch. 2	Ch. 3	Ch.1	Ch.2	Ch.3	Ch. 1	Ch. 2	Ch.3
64 Channel Model	1.412	1.490	1.569	1.86	2.06	2.03	0.121	0.114	0.102
8 Channel Model	1.395	1.476	1.558	1.80	2.00	1.99	0.124	0.115	0.102
CASE 2[a]									
64 Channel Model	1.601	1.674	1.750	1.93	2.15	2.19	0.087	0.081	0.070
8 Channel Model	1.594	1.670	1.749	1.90	2.13	2.17	0.088	0.081	0.069
CASE 3[a]									
64 Channel Model	1.243	1.338	1.454	1.37	1.54	1.53	0.163	0.154	0.138
8 Channel Model	1.216	1.315	1.433	1.32	1.48	1.47	0.166	0.156	0.138

a) Cases 1, 2, 3 are in reference to the operating conditions given in Table 4-1.

TABLE 4-3. COMPARISON OF 64 CHANNEL AND 8 CHANNEL MODEL TRANSIENT RESULTS (TYPICAL)

Time (sec)	64 Channel Model Channel 1 MDNBR	8 Channel Model Channel 1 MDNBR
0.0	1.833	1.830
0.5	1.813	1.807
1.0	1.774	1.767
1.5	1.715	1.708
2.0	1.645	1.636
2.5	1.569	1.558
2.7	1.528	1.517
2.9	1.501	1.489
3.1	1.471	1.454
3.3	1.436	1.420
3.4	1.396	1.376
3.5	1.356	1.333
3.6	1.331	1.308
3.7	1.306	1.285
3.8	1.280	1.260
3.9	1.261	1.238
4.0	1.241	1.221
4.1	1.234	1.216
4.2	1.245	1.221
4.3	1.281	1.250

TABLE 4-4. COMPARISON OF 64 CHANNEL AND 9 CHANNEL TRANSITION CORE MODEL STEADY-STATE RESULTS (TYPICAL)

	MDNBR			MASS VELOCITY (MLBM/HR-FT^2)			EXIT QUALITY		
CASE 1[a]	Ch. 1	Ch. 2	Ch. 3	Ch.1	Ch.2	Ch.3	Ch. 1	Ch. 2	Ch.3
64 Channel Model	1.412	1.490	1.568	1.86	2.06	2.03	0.121	0.114	0.102
9 Channel Model	1.398	1.479	1.560	1.80	2.00	1.99	0.123	0.115	0.102
CASE 2[a]									
64 Channel Model	1.600	1.674	1.750	1.93	2.15	2.19	0.087	0.081	0.070
9 Channel Model	1.597	1.673	1.751	1.90	2.13	2.18	0.087	0.081	0.069
CASE 3[a]									
64 Channel Model	1.243	1.338	1.454	1.37	1.54	1.53	0.163	0.154	0.138
9 Channel Model	1.220	1.318	1.436	1.32	1.48	1.47	0.165	0.155	0.138

a) Denotes Cases 1, 2 and 3 are operating conditions from Table 4-1.

TABLE 4-5. COMPARISONS OF 64 CHANNEL AND 9 CHANNEL TRANSITION CORE MODEL TRANSIENT RESULTS (TYPICAL)

Time (sec)	64 Channel Model Channel 1 MDNBR	9 Channel Model Channel 1 MDNBR
0.0	1.834	1.831
0.5	1.814	1.808
1.0	1.775	1.769
1.5	1.715	1.710
2.0	1.646	1.639
2.5	1.571	1.562
2.7	1.529	1.519
2.9	1.502	1.491
3.1	1.472	1.456
3.3	1.438	1.422
3.4	1.396	1.379
3.5	1.359	1.337
3.6	1.333	1.301
3.7	1.309	1.283
3.8	1.283	1.262
3.9	1.263	1.243
4.0	1.244	1.229
4.1	1.238	1.224
4.2	1.249	1.230
4.3	-----	1.256
4.4	-----	1.302

TABLE 5-1. 8 CHANNEL MODEL AXIAL NODE LENGTH SENSITIVITY STUDY (TYPICAL)

Operating Conditions[a]	Axial Peak	$\frac{X}{L}$	Node Size (in.)	Node Study Elevation (in.)	Channel 1 MDNBR	MDNBR @ Elevation (in.)
(CASE 1)	1.65	0.5	3	4.125-142.125	[ ]	97.1-100.1
	1.65	0.5	2	81.125-143.125[b]		99.1-101.1
(CASE 2)	1.65	0.5	3	4.125-142.125		94.1-97.1
	1.65	0.5	2	81.125-143.125[b]		95.1-97.1
(CASE 1)	1.70	0.1	3	4.125-142.125		64.1-67.1
	1.70	0.1	2	32.125-94.125[c]		66.1-68.1
(CASE 2)	1.70	0.1	3	4.125-142.125		58.1-61.1
	1.70	0.1	2	32.125-94.125[c]		58.1-60.1

Notes

a) Operating conditions from Table 4-1.
b) 4.125-81.125 in. range modeled with three-inch nodes.
c) 4.125-32.125 and 94.125-143.125 in. ranges modeled with three-inch nodes.

TABLE 5-2. 8 CHANNEL MODEL INLET FLOW DISTRIBUTION SENSITIVITY STUDY (TYPICAL)

Operating Condition

Case 1 from Table 4-1.

Percent Flow to Hot Assy.	MDNBR Channel 1	Channel 2	Channel 3
[			]
	MASS VELOCITY (MLBM/HR-FT^2)		
[			]
	EXIT QUALITY		
[			]

TABLE 5-3. 8 CHANNEL MODEL TWO-PHASE FLOW CORRELATION AND FRICTION MULTIPLIER SENSITIVITY STUDY (TYPICAL)

Operating Conditions[a]	Sub-Cooled Void	Bulk Void	Two-Phase Friction Multiplier	MDNBR			Mass-Velocity (MLBM/HR-FT²)			Exit Void Fraction			Exit Quality		
				Ch. 1	Ch. 2	Ch. 3	Ch. 1	Ch. 2	Ch. 3	Ch. 1	Ch. 2	Ch. 3	Ch. 1	Ch. 2	Ch. 3
CASE 1:															
	LEVY	ZUBR	EPRI												
	LEVY	ZUBR	HOMO												
	LEVY	SMIT	HOMO												
	EPRI	EPRI	EPRI												
	EPRI	EPRI	HOMO												
CASE 2:															
	LEVY	ZUBR	EPRI												
	LEVY	ZUBR	HOMO												
	LEVY	SMIT	HOMO												
	EPRI	EPRI	EPRI												
	EPRI	EPRI	HOMO												

a) Denotes operating conditions from Table 4-1.

Table 5-4. 8 CHANNEL MODEL TURBULENT MOMENTUM FACTOR SENSITIVITY STUDY (TYPICAL)

Operating Conditions	FTM	MDNBR			MASS VELOCITY @ MDNBR LOCATION (MLBM/HR-FT^2)			EXIT QUALITY		
		Ch. 1	Ch. 2	Ch. 3	Ch. 1	Ch. 2	Ch. 3	Ch. 1	Ch. 2	Ch. 3
CASE 1:										
	0.0									
	0.8									
	1.0									
CASE 2:										
	0.0									
	0.8									
	1.0									

TABLE 6-1. RPS TRIP FUNCTIONS

Reactor Trip	Monitored Parameter	Trip Setpoint During 4-Pump Operation [1]	Purpose of Trip	
1. Overpower Trip	Neutron flux	105.5% FP	To provide core protection during transients involving uncontrolled power increase.	
2. Power-flow-imbalance trip	Neutron flux, RC flow and power imbalance	Flux/flow = 1.094	To provide core protection during transients involving a flow reduction and during core conditions involving excessive power peaking	1
3. RCS pressure-temperature trip	RCS pressure and RC outlet temperature	Function of RC outlet temperature	To provide core protection during transients involving a reduction in pressure or a reduction in core heat removal and to ensure reactor shut down during a small break LOCA	1
4. Low RCS pressure trip	RCS pressure	1800 psig	To provide core protection during transients involving a pressure reduction	
5. RC Pump Monitor trip	Neutron flux and pump contact monitor voltage	$\geq$ 2% RTP with $\leq$ 2 pumps operating	To provide core protection during loss of RC pumps	1
6. High RCS pressure trip	RCS pressure	2355 psig	To provide protection of RCS pressure boundary from excessive pressures	
7. High RCS temperature trip	RC outlet temp.	618°F	To prevent excessive temperature in the RCS	
8. High RB pressure trip	RB pressure	4 psig	To ensure reactor shutdown during LOCA and SLB inside containment	1

Note

1. **The RPS trip functions listed above are for information only. The actual RPS trip functions are specified in the Oconee Nuclear Station Improved Technical Specifications or the Core Operating Limits Report.** 1

FIGURE 4-1. 64 CHANNEL MODEL EIGHTH-CORE REPRESENTATION

FIGURE 4-2. 64 CHANNEL MODEL HOT ASSEMBLY DETAIL

FIGURE 4-3. 8 AND 9 CHANNEL MODEL HOT ASSEMBLY DETAIL

FIGURE 4-4. 8 CHANNEL MODEL EIGHTH CORE REPRESENTATION

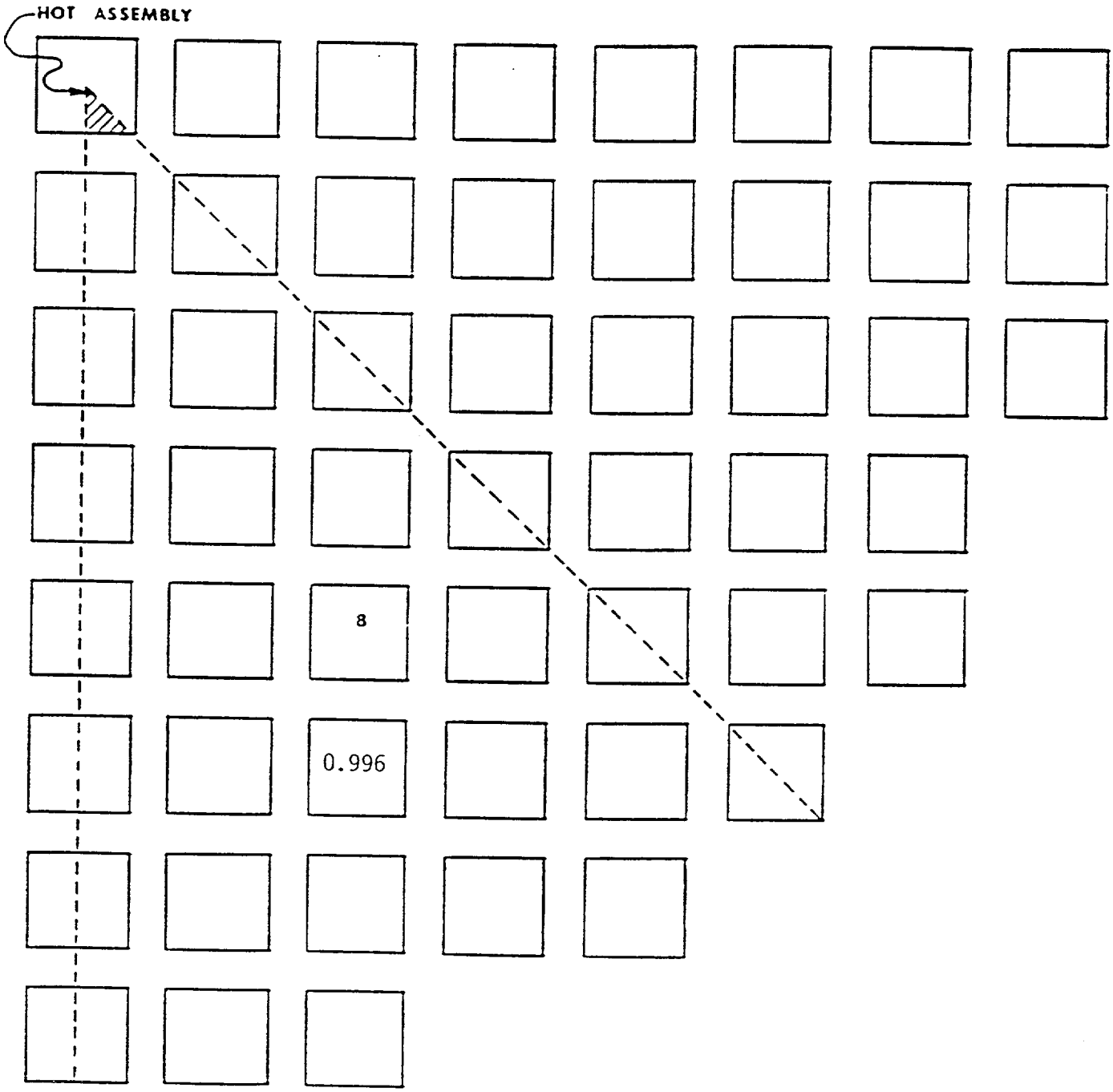


FIGURE 4-5. 9 CHANNEL MODEL EIGHTH CORE REPRESENTATION

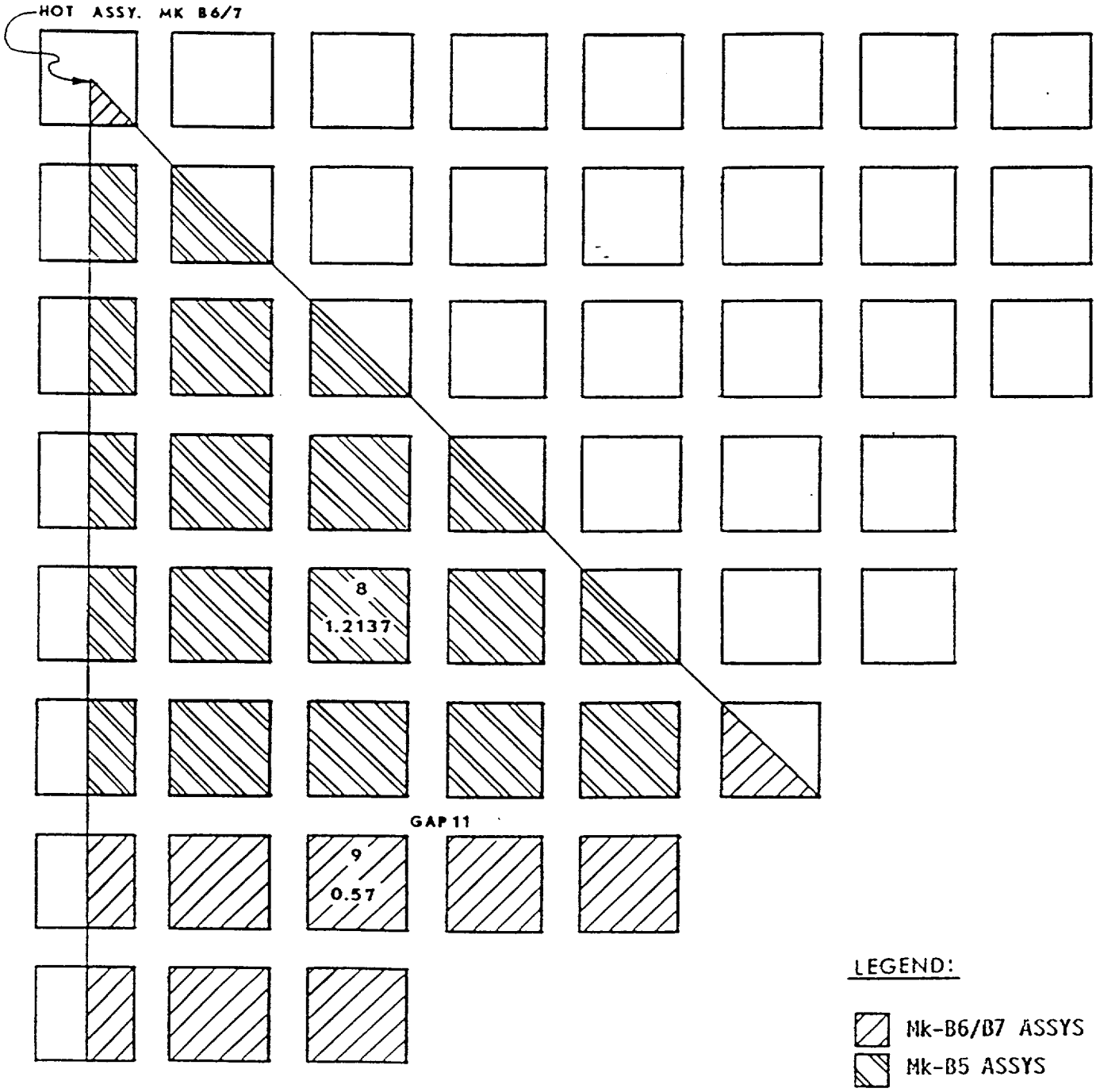


FIGURE 5-1

VIPRE-01 vs. LYNX2 M/P CHF
BWC CHF CORRELATION

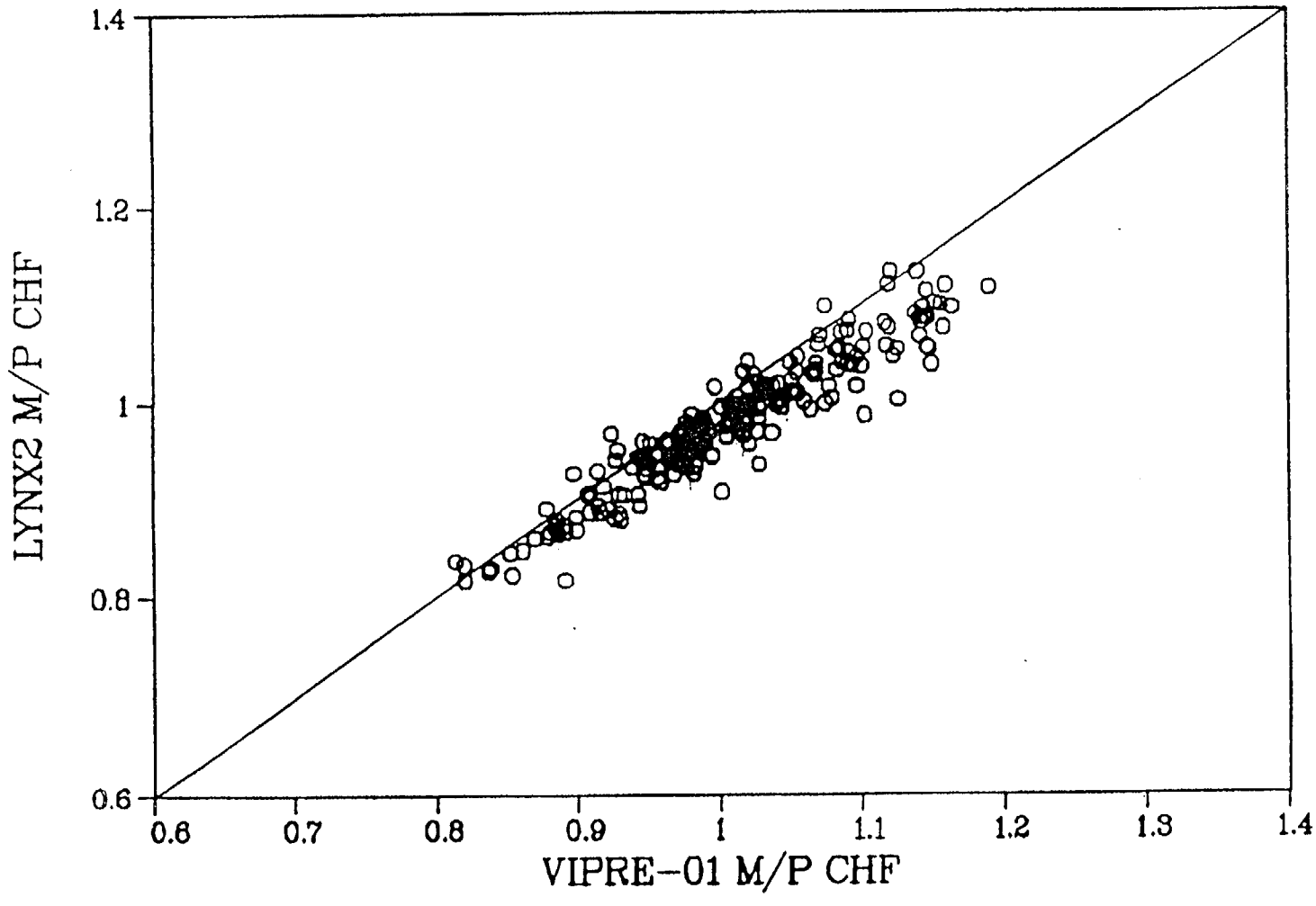


FIGURE 5-2

VIPRE-01 vs. LYNX2 MASS VELOCITY AT CHF
BWC CHF CORRELATION

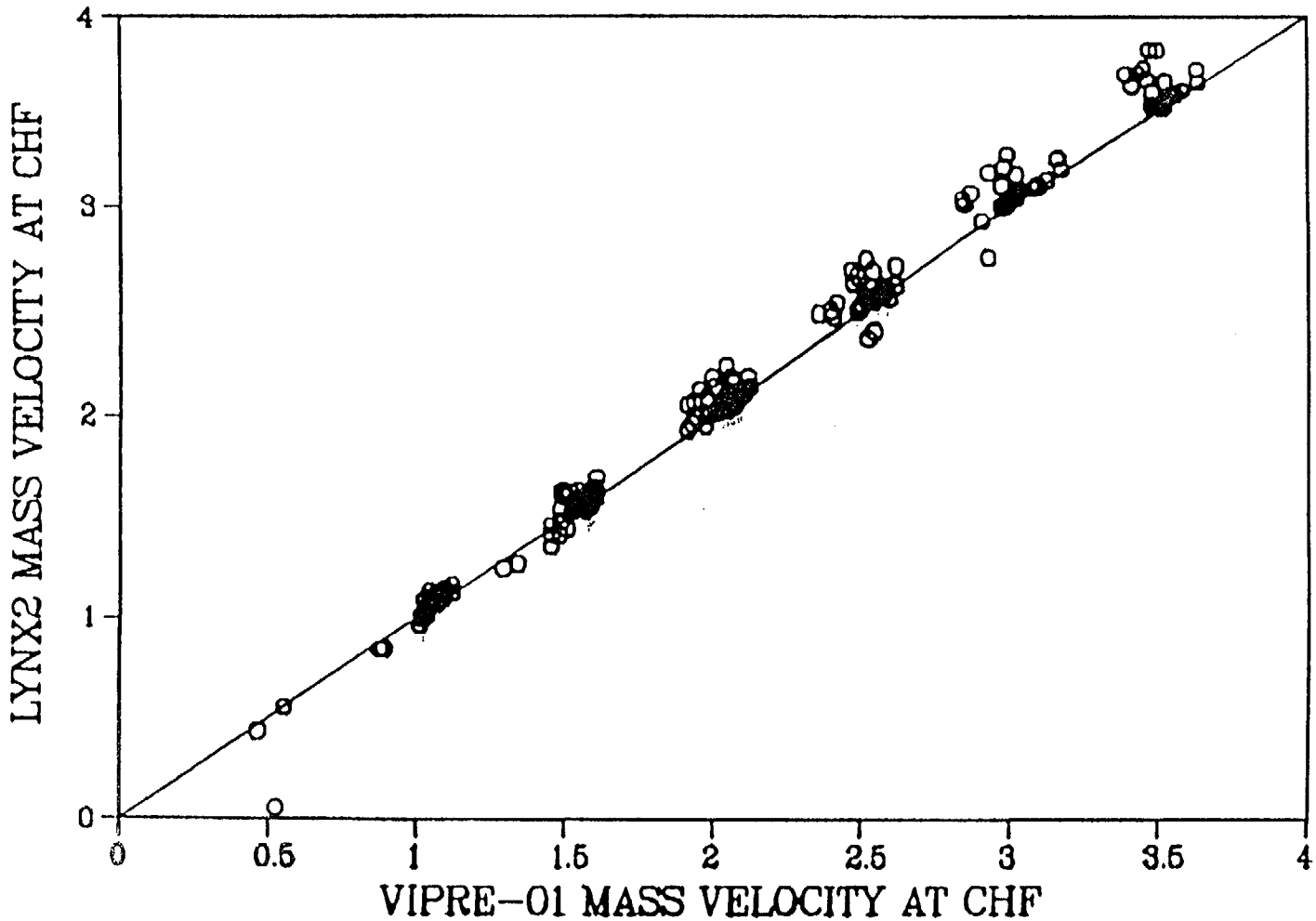


FIGURE 5-3

VIPRE-01 vs. LYNX2 QUALITY AT CHF
BWC CHF CORRELATION

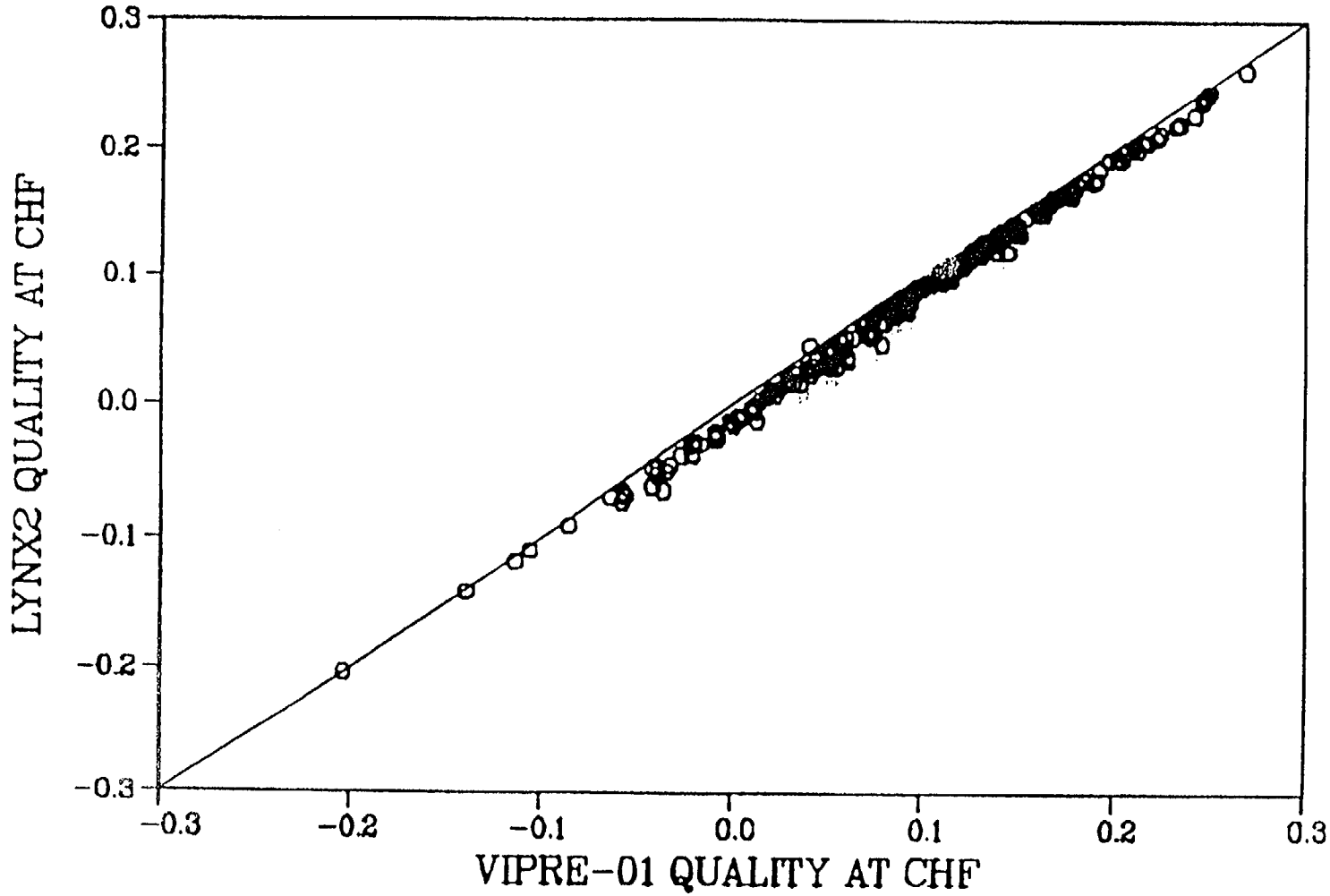


FIGURE 5-4

MEASURED vs. PREDICTED CHF
VIPRE-01, BWC CORRELATION

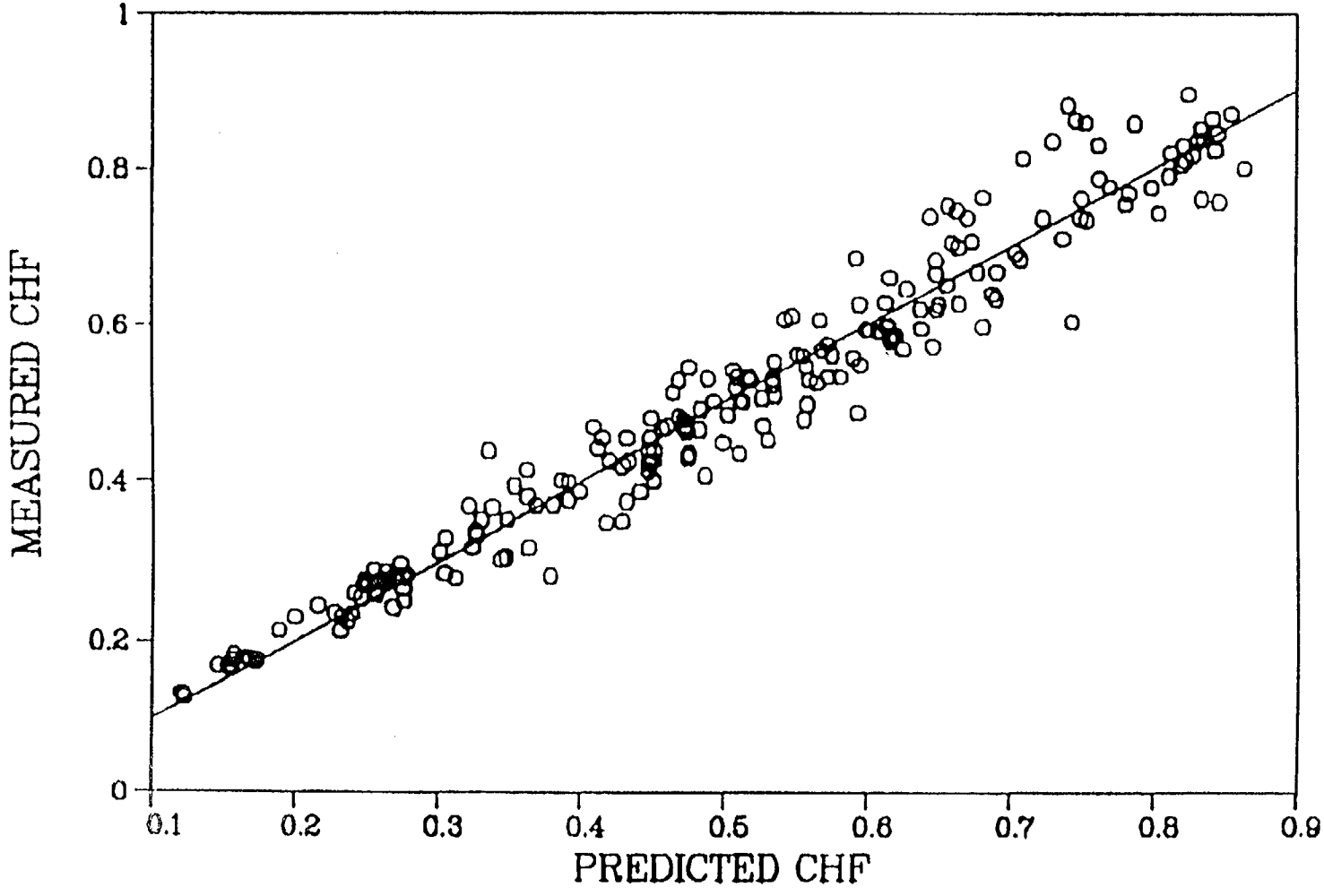


FIGURE 5-5

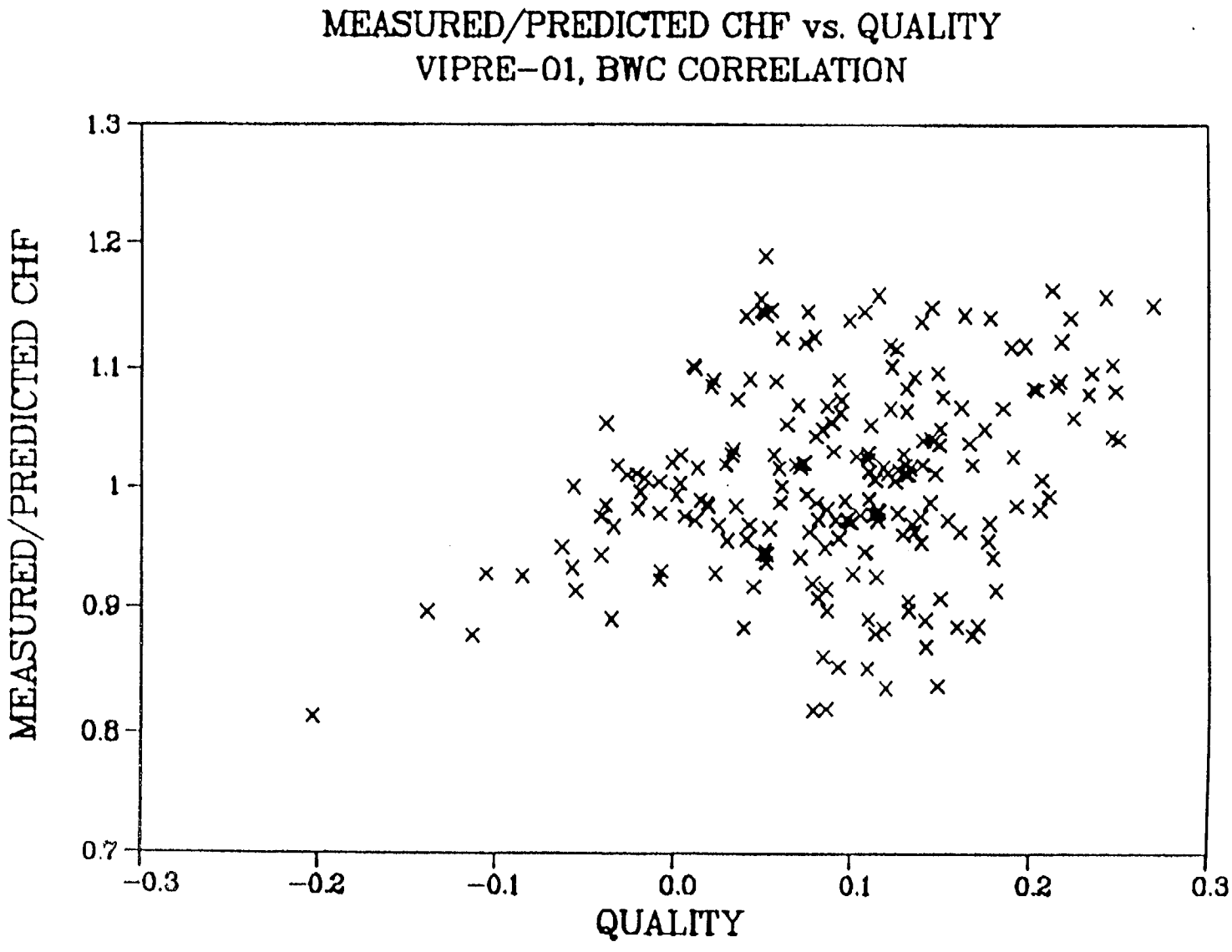

MEASURED/PREDICTED CHF vs. QUALITY
VIPRE-01, BWC CORRELATION
MEASURED/PREDICTED CHF
1.3
1.2
1.1
1
0.9
0.8
0.7
-0.3
-0.2
-0.1
0.0
0.1
0.2
0.3
QUALITY

FIGURE 5-6

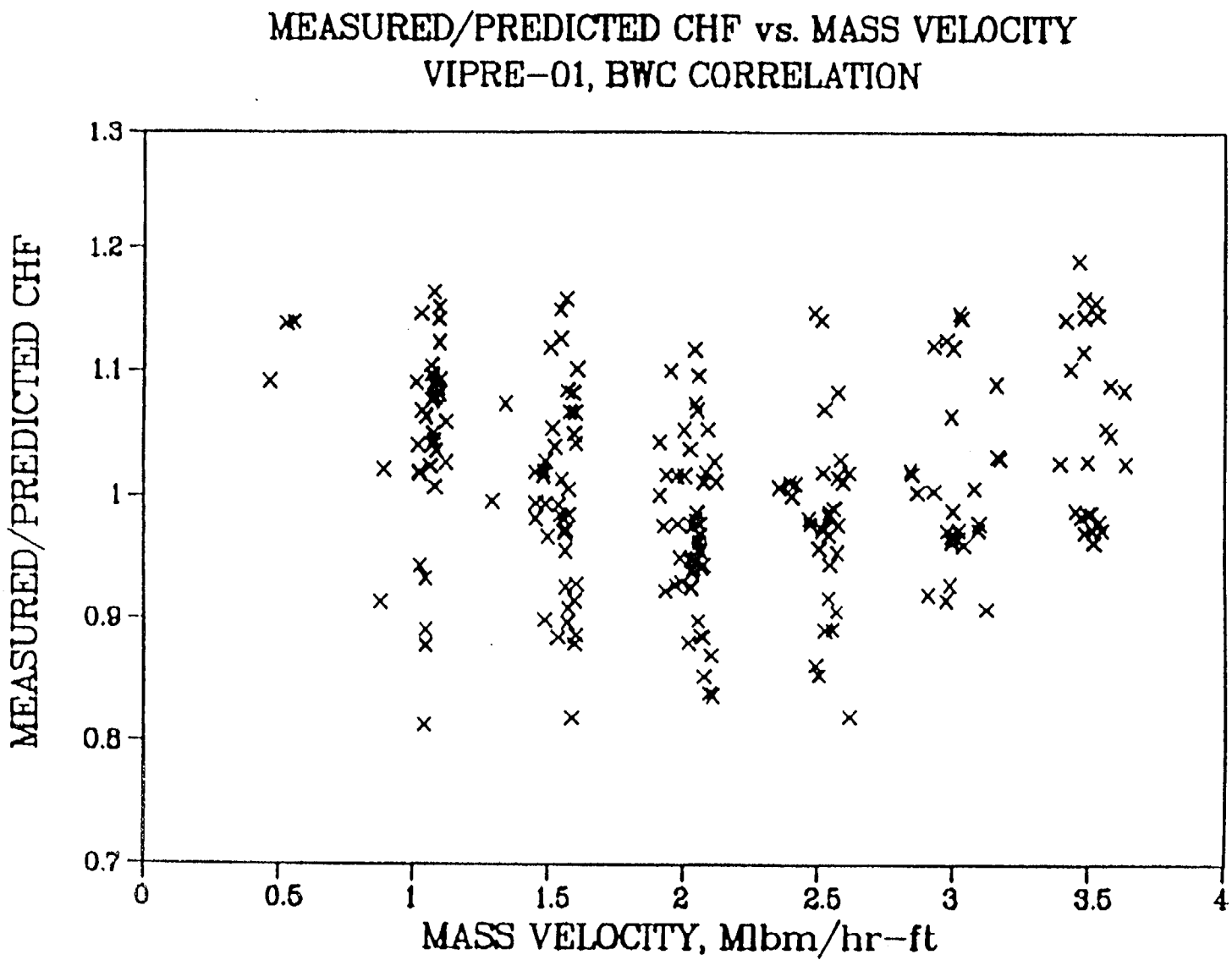

MEASURED/PREDICTED CHF vs. MASS VELOCITY
VIPRE-01, BWC CORRELATION
MEASURED/PREDICTED CHF
1.3
1.2
1.1
1
0.9
0.8
0.7
0
0.5
1
1.5
2
2.5
3
3.5
4
MASS VELOCITY, Mlbm/hr-ft

FIGURE 5-7

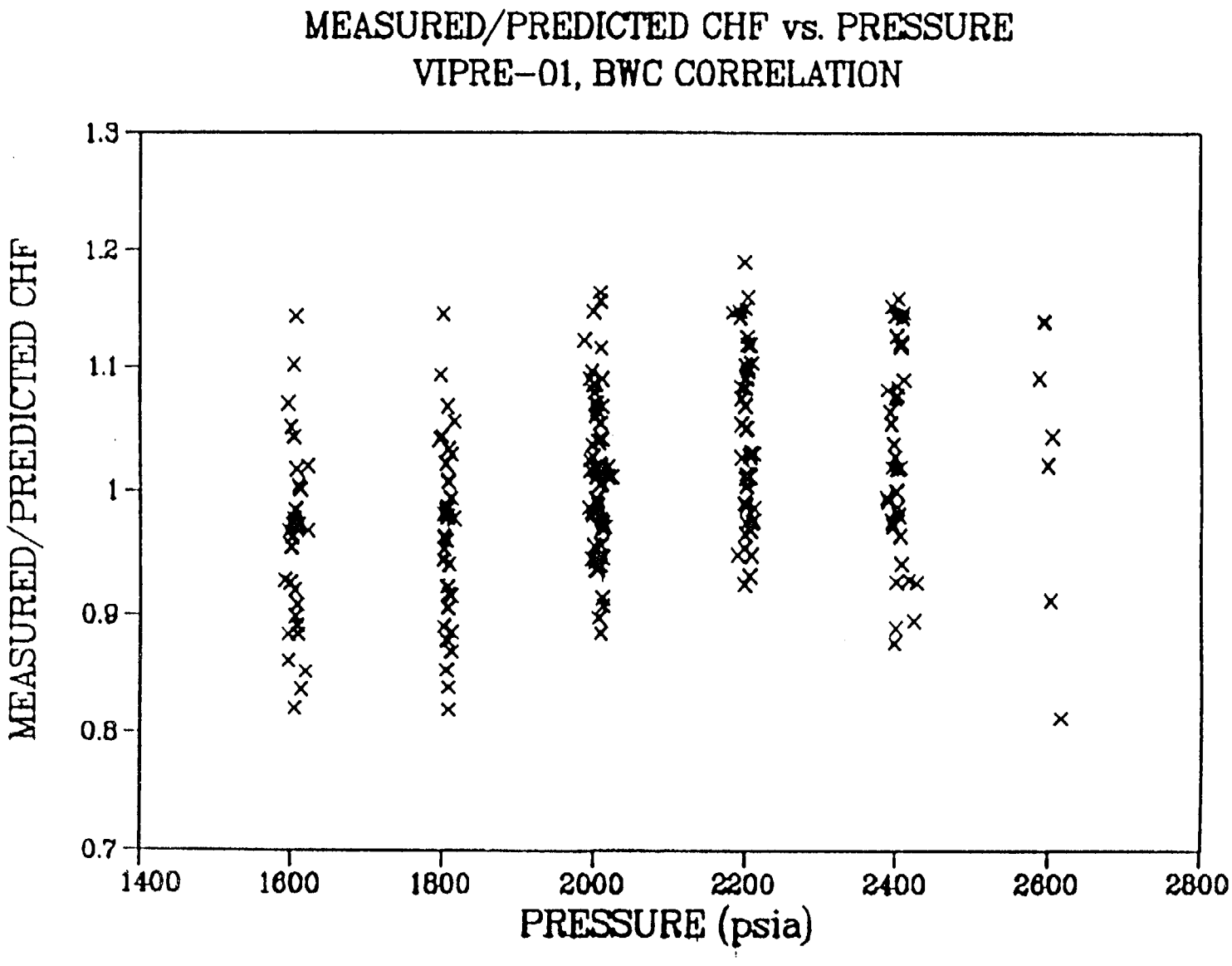
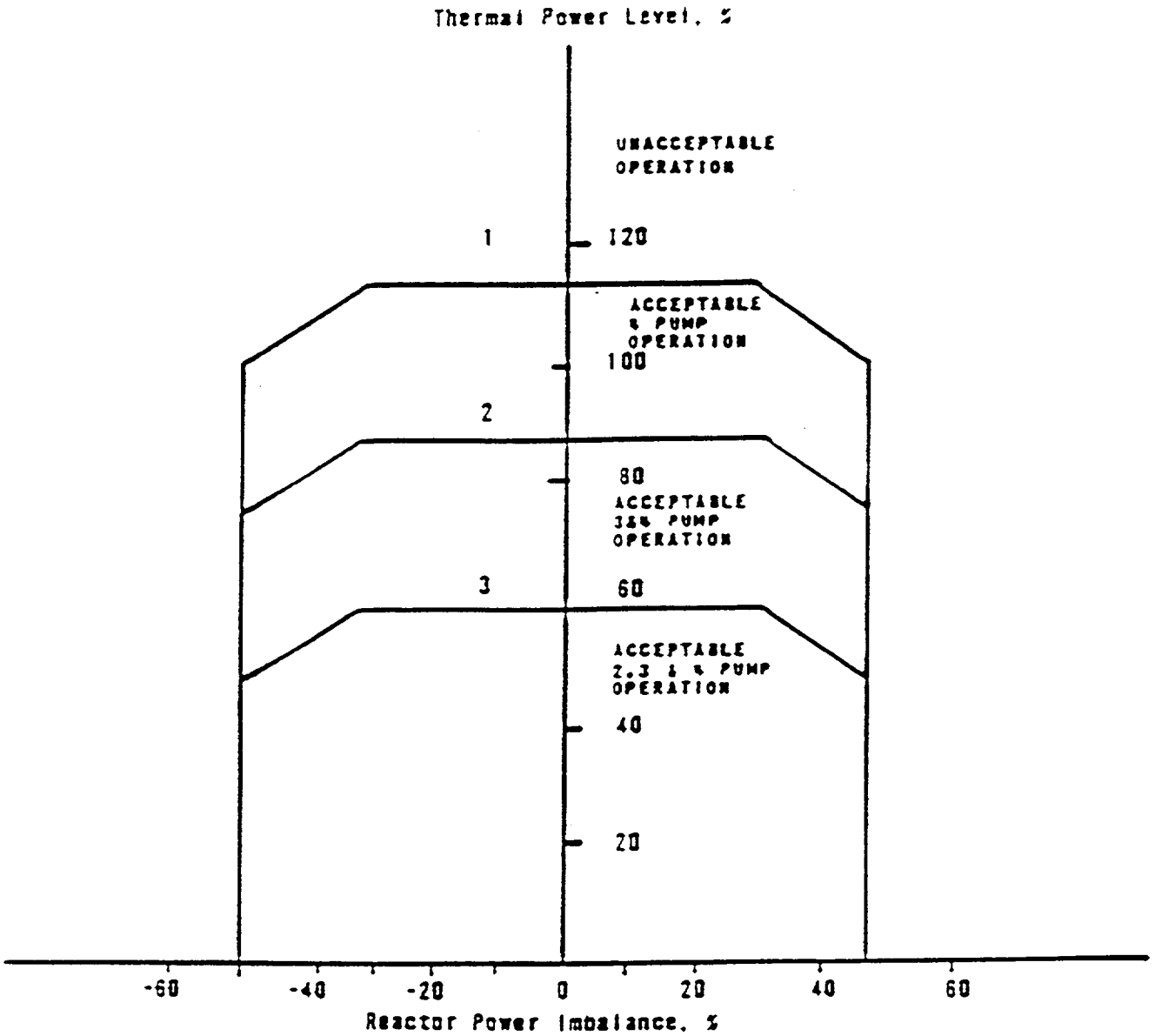

MEASURED/PREDICTED CHF vs. PRESSURE
VIPRE-01, BWC CORRELATION
MEASURED/PREDICTED CHF
1.3
1.2
1.1
1
0.9
0.8
0.7
1400
1600
1800
2000
2200
2400
2600
2800
PRESSURE (psia)

FIGURE 6-1 RPS CORE PROTECTION SAFETY LIMITS



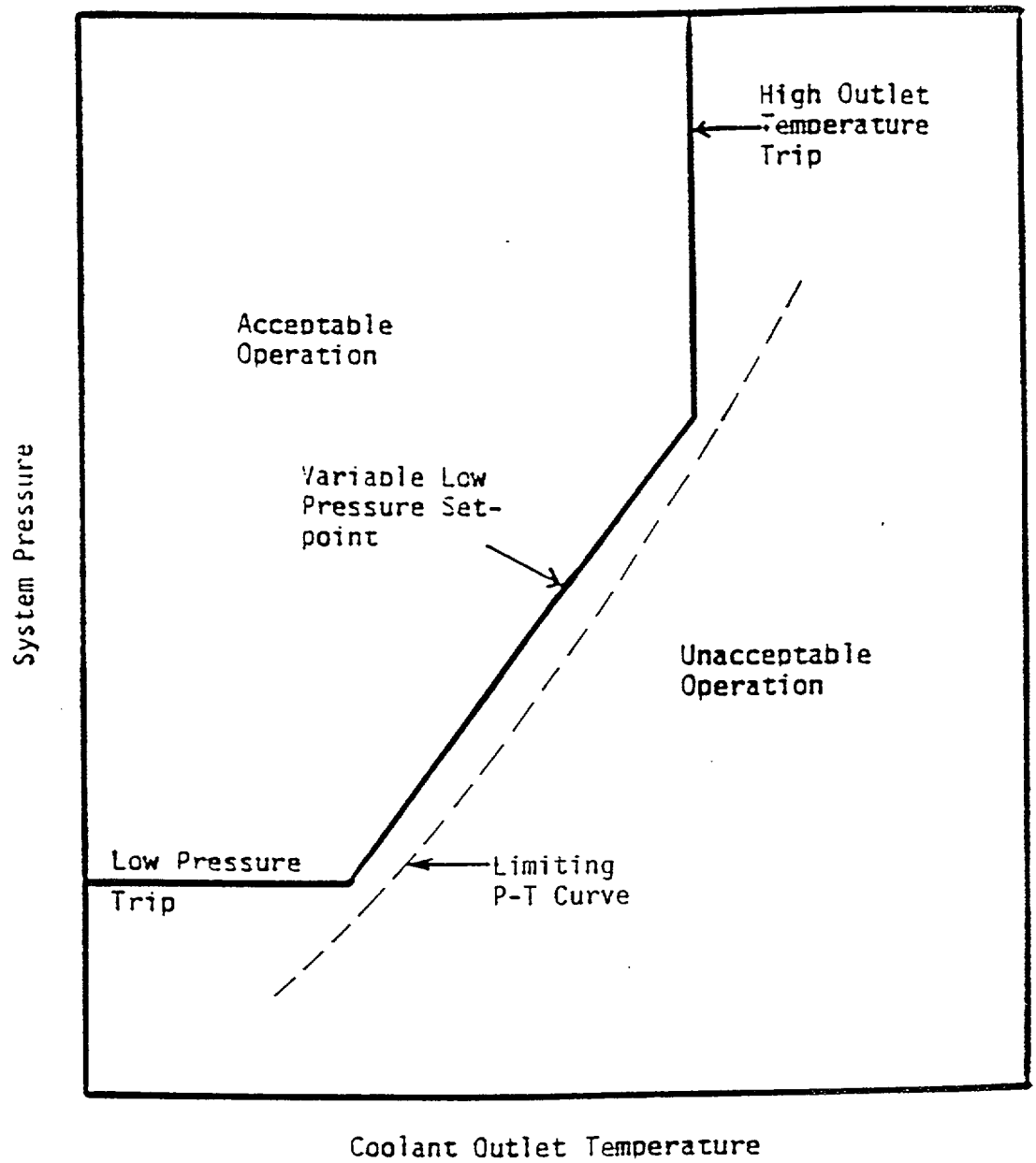


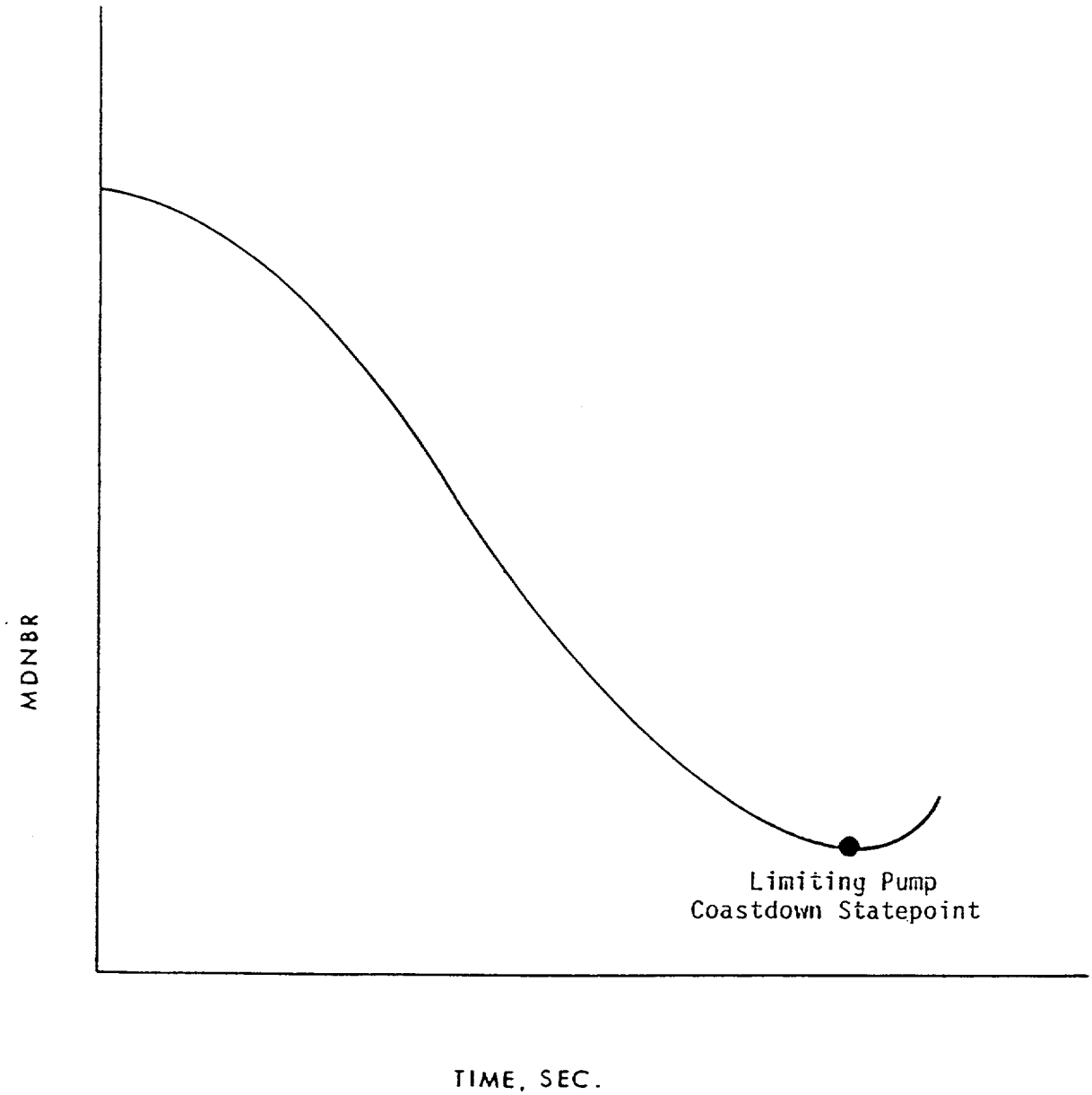
FIGURE 6-2. RPS P-T CORE PROTECTION ENVELOPE

FIGURE 6-3. HIGH TEMPERATURE TRIP MAPS

FIGURE 6-4. LOW PRESSURE TRIP MAPS

FIGURE 6-5. FLUX-TO-FLOW MAPS

FIGURE 6-6 TYPICAL 2 PUMP COASTDOWN TRANSIENT RESULTS



Appendix A

Safety Evaluation Report

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

July 19, 1989

Docket Nos: 50-269
50-270
50-287

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: SAFETY EVALUATION REPORT ON DPC-NE-2003, "CORE THERMAL-HYDRAULIC METHODOLOGY USING VIPRE-01" (TACS 69377/69378/69379)

The staff and its consultant, International Technical Services, have reviewed your Topical Report DPC-NE-2003, "Core Thermal-Hydraulic Methodology Using VIPRE-01" submitted for application to the Oconee Nuclear Station, Units 1, 2 and 3. We have found the topical report to be acceptable for referencing in the core thermal-hydraulic analyses for the Oconee units with the following limitations:

(1) The acceptable DNBR limit is 1.18. Acceptance of a DNBR limit less than 1.18 will require analysis of a broader CHF data base and detailed staff review.

(2) The studies provided in the topical report were performed with the fuel assembly design currently used in the Oconee units. Although the approach is acceptable for future fuel assembly designs, you should ensure that the selected correlations are used within their applicability ranges.

A copy of our Safety Evaluation Report is enclosed. This completes our action under TAC Nos. 69377, 69378 and 69379.

Sincerely,

Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/encl:
See next page

cc:

Mr. A. V. Carr, Esq.
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Cook, Purcell & Reynolds
1400 L Street, N.W.
Washington, D.C. 20005

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Division
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 34623-1693

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. Paul Guill
Duke Power Company
Post Office Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Mr. Alan R. Herdt, Chief
Project Branch #3
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Ms. Karen E. Long
Assistant Attorney General
N. C. Department of Justice
P.O. Box 629
Raleigh, North Carolina 27602

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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO TOPICAL REPORT DPC-NE-2003,
"CORE THERMAL-HYDRAULIC METHODOLOGY USING VIPRE-01"
DUKE POWER COMPANY
OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
DOCKET NOS. 50-269, 50-270 AND 50-287

1.0 INTRODUCTION

Duke Power Company (DPC) submitted Topical Report DPC-NE-2003, "Core Thermal-Hydraulic Methodology Using VIPRE-01," for Nuclear Regulatory Commission staff review in a letter dated August 31, 1988 (Ref. 1) and amended by a letter of May 3, 1989 (Ref. 2). This report documents DPC's use of the VIPRE-01 code (Ref. 3) in lieu of the currently used codes, CHATA and TEMP (Refs. 4 and 5), for Oconee Nuclear Station licensing core thermal-hydraulic methodology. The Oconee core thermal-hydraulic analyses are routinely performed for fuel reloads to ensure that the departure from nucleate boiling ratio (DNBR) limit will not be violated during steady state overpower condition and anticipated transients. These analyses consist of (1) a steady state thermal hydraulic analysis to determine the allowable pressure-temperature operating limits and the power distribution limits, and (2) an analysis of the limiting two pump coastdown transient to determine a flux/flow reactor trip setpoint. Since the methodology of determining these safety and operating limits has been reviewed and approved (Ref. 6) previously, the staff review of the topical report concentrated on the use of the VIPRE-01 code in the core thermal hydraulic calculations.

VIPRE-01 is an open-lattice subchannel core thermal-hydraulic code. In the open-lattice analysis, 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. This approach more realistically considers the

flow redistribution effects in the open-lattice core of a pressurized water reactor (PWR) and results in less severe hot channel thermal hydraulic conditions than that obtained from the closed-channel approach used in CHATA.

VIPRE-01 was developed by Battelle Pacific Northwest Laboratories under the sponsorship of the Electric Power Research Institute. In December 1984, the Utility Group for Regulatory Application submitted the VIPRE-01 code for NRC staff review (Ref. 7). In approving VIPRE-01 for PWR licensing applications (Ref. 8), the staff required each VIPRE-01 user to submit separate documentation describing its intended use of VIPRE-01 and providing justification for its specific modeling assumptions, choices of particular models and correlations, and input values of plant specific data.

In a letter of June 19, 1989 (Ref. 9), DPC indicated that the VIPRE-01 models and methodology described in DPC-NE-2003 are related to the reload thermal hydraulic analyses, that the methodology of using VIPRE-01 model for predicting the minimum DNBRs resulting from FSAR Chapter 15 transients, except for the two-pump coastdown, are described in Topical Report DPC-NE-3000, and that the VIPRE-01 methodology for transient analyses may be different from that used in DPC-NE-2003. Therefore, the scope of the staff review of DPC-NE-2003 was limited to the application of VIPRE-01 in the steady state and two-pump coastdown analyses.

2.0 STAFF EVALUATION

The staff review and evaluation of DPC-NE-2003 included: (1) the nodal sensitivity studies to determine the radial noding details and the axial node sizes, (2) the plant-specific core thermal-hydraulic parameters such as the crossflow parameters, grid loss coefficients, core bypass flow, inlet flow distribution and flow area reduction factor, power distributions, hot channel factors, (3) the selected two-phase flow, heat transfer models and correlations, (4) the validation of the BWC critical heat flux correlation (Ref. 10) and the DNBR limit in conjunction with VIPRE-01, and (5) the fuel pin heat conduction parameters.

The review was performed with technical assistance from International Technical Services (ITS), and its review findings are contained in the technical evaluation report (TER) which is attached. The staff has reviewed the ITS TER and concurred with its findings.

3.0 CONCLUSION

The staff has reviewed the Topical Report DPC-NE-2003 and finds it acceptable for referencing in the Oconee reload thermal-hydraulic analyses, subject to the following limitations:

(1) The validation analysis with limited CHF data has demonstrated that the approved DNBR limit of 1.18 for the BWC CHF correlation, which was derived with the LYNX2 thermal-hydraulic code, is conservative and acceptable for use with VIPRE-01. Acceptance of a DNBR limit less than 1.18 will require analysis of broader CHF data base and detailed staff review.

(2) The studies provided in the topical report were performed with the Mark BZ fuel assembly design currently used in Oconee units. Though the approach described is acceptable for future fuel assembly designs, DPC should ensure that the selected correlations be used within their applicability ranges.

4.0 REFERENCES

1. Letter from H. B. Tucker (DPC) to USNRC Document Control Desk, "Oconee Nuclear Station, Docket Nos. 50-269, -270, -287, Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003," August 31, 1988.

2. Letter from H. B. Tucker (DPC) to USNRC Document Control Desk, "Oconee Nuclear Station, Docket Nos. 50-269, -270, -287, Topical Report DPC-NE-2003, 'Core Thermal-Hydraulic Methodology Using VIPRE-01'; Response to Request For Additional Information," May 3, 1989.

3. EPRI-NP-2511-CCM-A, "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores," 4 Volumes, Electric Power Research Institute.

4. BAW-10110, Rev. 1, "CHATA - Core Hydraulic and Thermal Analysis," May 1977.

5. BAW-10021, "TEMP - Thermal Enthalpy Mixing Program," April 1970.

6. Letter from P. C. Wagner (USNRC) to W. O. Parker, Jr. (DPC), Attachment: Safety Evaluation Report on NFS-1001, "Oconee Nuclear Station Reload Design Methodology," July 29, 1981.

7. Letter from J. A. Blaisdell (Northeast Utilities Service Co.) to H. R. Denton (USNRC), Subject related to UGRA submittal of the VIPRE-01 code, December 17, 1984.

8. Letter from C. E. Rossi (NRC) to J. A. Blaisdell, Chairman, UGRA Executive Committee, "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," May 1, 1986.

9. Letter from H. B. Tucker (DPC) to USNRC Document Control Desk, "Oconee Nuclear Station, Docket Nos. 50-269, -270, -287, Response to Questions Regarding Differences Between Duke Topical Reports DPC-NE-2003 and DPC-NE-3000," June 19, 1989.

10. BAW-10143-A, "BWC Correlation of Critical Heat Flux," April 1985.

ATTACHMENT

ITS/NRC/89-2

TECHNICAL EVALUATION OF THE CORE THERMAL-HYDRAULIC METHODOLOGY USING VIPRE-01 TECHNICAL REPORT DPC-NE-2003 FOR THE DUKE POWER COMPANY OCONEE NUCLEAR STATION

1.0 INTRODUCTION

In Duke Power Company (DPC) topical report DPC-NE-2003, dated August 1988 (Ref. 1), DPC presented a description and qualification of their core thermal-hydraulic methodology using VIPRE-01 (Ref. 2) for steady-state and for two reactor coolant pump coastdown analyses of the Oconee Nuclear Station reload. VIPRE-01 has been previously reviewed and approved for application to pressurized water reactor (PWR) plants in steady-state and transient analyses with heat transfer regimes up to critical heat flux. The NRC safety evaluation report (SER) on VIPRE-01 (Ref. 3) includes conditions requiring each user to document and submit to the NRC for approval its procedure for using VIPRE-01 and provide justification for its 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 and grid loss coefficient including defaults. This topical report was prepared to address these issues.

The purpose of this review was to assure conformity of the DPC topical report and supplemental information (Ref. 4, 5) to the VIPRE SER requirements, and to evaluate acceptability of DPC's intended use of the code as described in the report.

In the past DPC used (Ref. 6) CHATA, a closed-channel (no energy or mass interchange among assemblies) computer code for core-wide analysis, and TEMP

to determine the maximum permissible core power and distribution under various operating conditions for Oconee core thermal-hydraulic design and licensing analyses. Although this approach was conservative, these codes were unable to realistically predict flow redistribution effects in an open lattice reactor core.

The VIPRE-01 computer code (Ref. 2) is an open-channel (permitting lateral communication among channels by diversion crossflow and turbulent mixing) thermal-hydraulic computer code developed to evaluate nuclear reactor core safety limits. The code assumes the flow to be incompressible and homogeneous and incorporates models to reflect subcooled boiling and liquid/vapor slip. The input data to the VIPRE-01 code are the geometry of the reactor core and coolant channels with thermal-hydraulic characteristics, and boundary conditions. In addition, the user must select among certain correlations in the code for use in the particular analysis being performed. The code calculates the core flow distributions, coolant conditions, fuel rod temperatures and the minimum departure from nucleate boiling ratio (MDNBR).

The DPC submittal, in fulfillment of VIPRE SER (Ref. 3) conditions, contains DPC's geometric representation of the core, its selection of thermal-hydraulic models and correlations, and a description of the methodology used for steady-state core reload design analysis and for a two-pump coastdown transient. These analyses are performed to determine the core thermal margin and acceptable safety and operating limits and to analyze a two-pump coastdown transient. It is not DPC's intent to use this methodology for FSAR Chapter 15 type licensing transient analysis.

2.0 EVALUATION

Acceptability of DPC's application of the VIPRE-01 computer code for thermal-hydraulic calculation of DNB for Oconee was evaluated with respect to the sensitivity of the computed steady-state operating conditions to input selection, nodalization, thermal-hydraulic modeling, and correlations, by examination of the overall conservatism in the results.

2.1 CORE NODALIZATION

2.1.1 Radial Noding Sensitivity

Since the VIPRE-01 code performs the thermal-hydraulic calculations simultaneously for all subchannels (a single-pass approach) and permits flexibility in selection of channel sizes and shapes, a sensitivity study was performed to determine the sensitivity of predicted DNBR to the subchannel model sizes. The modeling of the reactor core uses the 1/8-core symmetry in which the hot assembly is located at the center of the core. The hot assembly includes the hot subchannel in which the minimum DNBR is expected to occur.

The thermal-hydraulic calculations were performed for three different core subchannel models; a 64 channel model, a 9 channel model, and an 8 channel model. The 64 channel model consists of 36 subchannels representing the hot assembly and 28 subchannels individually modeling each of the remainder of assemblies in the 1/8-core segment. In the 8 channel model, 6 subchannels around the hot subchannel in the hot assembly are modeled individually. The rest of the subchannels in the hot assembly and the remaining 28 assemblies in the core are lumped into 2 individual subchannels (Channels 7 and 8). The 9 channel model, developed for evaluation of transitional mixed core effects, includes an additional subchannel to account for the different fuel assembly designs in the transition core.

The nodalization sensitivity studies used the same thermal-hydraulic correlations and models which DPC intends to use in future reload licensing analysis. Review of the particular correlations and thermal-hydraulic models selected is provided in Section 2.2.

Steady-state and transient calculations using the previously approved RECIRC numerical solution option were performed using these three different core models at four different operating conditions: the high temperature and the low pressure safety limits, and two different sets of initial conditions for pump coastdown transients including one representing the limiting MDNBR case.

Sensitivity to the core model size was studied by comparing the results of using the 64 and 8 channel models. The 8 channel model was found to yield MDNBRs ranging from 0.44% to 2.2% lower than the 64 channel model. We therefore find DPC's use of the 8 channel model acceptable for Oconee steady-state and 2-pump coastdown reload thermal-hydraulic analysis.

Sensitivity of the core models to transitional mixed core effects was examined using the 9 and 64 channel models in both steady-state and 2-pump coastdown transient conditions. For steady-state conditions, the 9 channel transition core model predicted 1.9% lower MDNBR than the 64 channel model. For the transient analysis the MDNBR predicted by the 9 channel model was 1.6% lower than the 64 channel model. Based upon these sensitivity studies, DPC intends to used the 9 channel model for steady-state and pump coastdown reload analyses involving transition cores of the Oconee Nuclear Station.

2.1.2 Axial Noding Sensitivity

A steady-state sensitivity analysis for axial node length was performed with the 8 channel model using two sizes: a 3-inch node length applied uniformly and a 2-inch node length applied where DNB is expected to occur. The results indicated that the 3-inch axial nodes produced slightly more conservative MDNBR than did the 2-inch nodes. We, therefore, find that use of 3-inch uniformly spaced axial nodes is acceptable for Oconee reload steady-state and pump coastdown thermal-hydraulic analyses.

2.2 VIPRE-01 Input Data

DPC's approach to generation of input to the VIPRE-01 code was reviewed for acceptability. No review was conducted of the input data in comparison to the actual physical geometry.

2.2.1 Active Fuel Length

Since power is distributed over the length of the active fuel, a shorter length yields higher power density, causing greater heat flux and is

therefore conservative. DPC's choice for the active fuel length as described in Section 5.2 of Ref. 1 is conservative when compared to hot conditions. When a different assumption is used, DPC should justify its conservatism.

2.2.2 Centroid Distance and Effective Crossflow Gaps

The centroidal distance is used in the crossflow momentum equation to determine the lateral pressure gradient. The gap width is used in determination of the crossflow area. DPC calculates these parameters from channel geometry following the code's prescription.

2.2.3 Spacer Grid Form Coefficients

Pressure losses across the spacer grids impact the axial pressure distribution and therefore the axial location of DNB. The spacer grid form loss coefficients were obtained from a full size fuel assembly test conducted by the vendor (B&W). To determine the individual subchannel form loss coefficient, DPC stated (Ref. 4) in response to our question that the vendor used its computer code, GRIL. The input data to the GRIL code are the individual subchannel geometry, drag areas and coefficients, and the coolant information. From this input, the code calculates individual subchannel loss coefficients, an overall grid loss coefficient and subchannel velocities based on single-phase flow input data by a iterative process. The calculated overall grid loss coefficient is matched with the measured value by adjusting the velocity field in the subchannel until consistency between the measured and predicted values is achieved. DPC has stated that the calculated velocity profiles were compared by the vendor with the experimental data and showed good agreement (Ref. 4).

2.2.5 Core Bypass Flow

DNB is influenced by the aggregate flow rate past the location being examined, and therefore by the core bypass flow. Since the bypass flow depends on the number of control rod and burnable poison rod assemblies in the core, this is a cycle dependent parameter. Therefore, the core bypass

flow data used in the analysis should be based on a bounding value or on a cycle specific data.

2.2.6 Inlet Flow Distribution

CHF is decreased and the probability of DNB is enhanced if flowrate is reduced due to a flow maldistribution. The use of 5% inlet flow maldistribution to the hot assembly with all four reactor coolant pumps operating yielded slightly more conservative results than a uniform inlet flow distribution. This value is supported by a B&W 1/6-scale Vessel Model Flow Test and was previously approved for Oconee reload analysis (Ref. 5). For operation with less than four reactor coolant pumps operating, more restrictive flow reduction factors are applied.

2.2.7 Flow Area Reduction Factor

DPC reduced the hot subchannel flow area by a factor as stated in Section 5.12 of Ref. 1 to account for variations in as-built subchannel coolant flow area.

2.2.8 Reference Design Power Distribution

The reference design power distribution was developed using a radial-local hot pin peak of 1.714 which has been previously approved for Oconee reload analysis (Ref. 5, 6). The corresponding assembly power was 1.6147.

2.2.9 Axial Power Distribution

The axial power shape used in the analyses was a chopped cosine shape with a conservatively determined peaking factor. Although the axial power shape is cycle specific and transient dependent, the use of generic bounding axial power curves accounts for the effect on DNB of different axial shapes. This is discussed in Section 2.4.

DPC added an optional new routine to VIPRE-01 to generate the axial power

shapes using a generalized power function. The currently defined function can generate both symmetric and skewed power shapes but cannot generate certain power shapes (such as double peaked) because of limitations of the generalized function used. The axial power shapes calculated using this routine agreed with the symmetric axial shapes calculated using VIPRE-01 symmetric cosine routine for axial peaks of 1.2 and 1.5 (Ref. 4).

DPC intends to maintain two options for power shape generation: one is to use this routine and the other is to use a user specified table. The use of this routine is acceptable so long as the computed power shapes represent the true power shapes to be analyzed.

Although analyses in this report were performed using a higher axial peaking factor, DPC will continue to use the reference axial peaking factor consistent with the current FSAR Chapter 15 transient analysis in the reload licensing analysis (Ref. 5).

2.2.10 Hot Channel Factor

The power factor, F_q, used to account for variations in average pin power caused by differences in the fuel loading per rod was selected to be 1.0107 which has been previously approved for Oconee reload analysis (Ref. 6).

The local heat flux factor, F_q'', used to account for the uncertainty in the manufacturing tolerances was selected to be 1.0137. In the determination of the maximum allowable peaking limits, two additional factors were used to increase the limit to 1.0371. These factors were 1.007 to account for power spikes occurring as a result of the flux depressions at the spacer grids, and 1.016 to account for axial nuclear uncertainty (Ref. 6). All of these factors have been previously approved for Oconee reload analysis.

2.3 VIPRE-01 Correlations

VIPRE-01 requires empirical correlations for the following models:

a. turbulent mixing

b. friction pressure loss
c. two-phase flow correlations (subcooled and saturated void, and void-quality relation)
d. single-phase forced convection
e. nucleate boiling heat transfer
f. critical heat flux

2.3.1 Friction Pressure Loss, Subcooled Void, Single-Phase and Two-Phase Flow Correlations

For single-phase turbulent flow the Blasius smooth tube friction factor, a default option in VIPRE-01, will be used to calculate the friction pressure loss in the axial direction. Crossflow resistance has a minimal effect on MDNBR in transients where axial flow dominates. DPC's selection therefore has an inherent assumption of axial flow dominance. This choice is acceptable since we agree that in the analyses to be performed in the context of this topical report, the flows are expected to be axially dominant.

For two-phase flow, subcooled and bulk void correlations, a sensitivity study using six different combinations of three subcooled and bulk void correlations was performed for two operating conditions. The results indicated that the use of Levy subcooled void and Zuber-Findlay bulk void correlations, in conjunction with EPRI two-phase friction multiplier results in conservatively predicted DNBR relative to other combinations of correlations. DPC intends to use this combination in Oconee steady-state and pump coastdown reload analysis.

This is consistent with the VIPRE-01 SER findings.

2.3.2 Turbulent Mixing

The lateral momentum equation requires two parameters: a turbulent momentum factor and a turbulent mixing coefficient.

The turbulent momentum factor (FTM) describes the efficiency of the momentum

mixing: 0.0 indicating that crossflow mixes enthalpy only; 1.0 indicating that crossflow mixes enthalpy and momentum at the same strength. A sensitivity study using the 8 channel model was performed for two operating conditions and for three different values of FTM of 0.0, 0.8, and 1.0 and found little sensitivity in DNBR by different values of FTM. Conservative DNBR's were obtained with zero (Table 5-4 in Ref. 1). However, in reality there will be always some momentum mixing. An FTM of 0.8 has been recommended by the VIPRE-01 code developer.

Since the turbulent mixing coefficient determines the flow mixing rate, it is an important parameter. Based upon tests using a 5x5 heated bundle conducted by B&W, where the subchannel exit temperatures were measured, a mixing coefficient was conservatively determined for B&W Mark-B fuel (Ref. 4). This will be used in the Oconee core steady-state and pump coastdown reload thermal-hydraulic analysis (Ref. 1).

2.3.3 Single-Phased Forced Convection, Nucleate Boiling Heat Transfer

DPC will use (for its steady-state and pump coastdown analyses) the default EPRI single-phased forced convection correlation and Thom subcooled and saturated nucleate boiling correlations, both of which were found to result in conservative MDNBR for the two-pump coastdown transient.

2.3.4 BWC Critical Heat Flux Correlation

The BWC correlation (Ref. 7) was originally developed for 17x17 Mark-C fuel, and later used for 15x15 Zr grid Mark-BZ fuel. The use of BWC correlation with the LYNX2 code (Ref. 8) for 15x15 Zr grid Mark-BZ fuel was previously approved by NRC with a design limit of 1.18 (Ref. 8, 9).

All Oconee thermal-hydraulic analyses using VIPRE-01 and the BWC correlation will use a design limit of 1.18. Since the BWC correlation is now being used with VIPRE-01, it is necessary for DPC to demonstrate that the DNBR limit of 1.18 for BWC CHF correlation used in VIPRE-01 can predict its date base of DNB occurrence with at least a 95% probability and a 95% confidence level.

In Section 5.13 of the topical report, DPC performed validation using more than 200 data point. Results show a 95%/95% limit of 1.16. Therefore use with VIPRE-01 of the previously approved (with LYNX2) value of 1.18 is conservative and acceptable. DPC agreed that when a lower DNBR limit becomes desirable with use of BWC CHF correlation with VIPRE-01, it will submit a separate topical report documenting analysis based on a broader CHF database for detailed NRC review and approval.

2.4 Oconee Thermal-Hydraulic Analyses

Using the input, assumptions, and thermal-hydraulic correlations selected and justified in the subject topical report, DPC discussed its methodology to perform steady-state and generic two-pump coastdown analyses necessary to define the core thermal margin or safety limits and acceptable operating limits.

The core safety limits that provide DNB protection are pressure - temperature (P-T) envelope and power - power imbalance limits. The P-T envelope defines a region of allowable operation in terms of reactor coolant system pressure and coolant temperature (Ref. 6).

To ensure that the P-T envelope provides adequate DNB protection, P-T curves are determined for different numbers of RC pump operation. P-T curves are the combinations of RCS pressure and vessel outlet temperature that yield the design DNBR limit or the BWC correlation quality limit. The P-T envelope must be more restrictive than the most limiting P-T conditions. VIPRE-01 was used to generate the generic P-T curves using the 8 channel model.

The following are input to the code for generation of P-T curves:

1. a symmetric chopped cosine with a conservative axial peaking factor;
2. 112 % of full power for 4-pump operation, and the power level for other modes of pump operation are based on trip setpoint plus margin for uncertainties;
3. 104% of design RCS flow for 4 pumps; appropriately lower for less

than 4-pump operation;

4. minimum coolant temperature; and
5. generic maximum allowable peaking (MAP) limit curves.

Having developed the P-T curves, DPC, as part of its reload analysis, performs a two-pump coastdown transient to determine the flux/flow trip setpoint. This trip provides DNB protection during a loss of one or more reactor coolant pumps.

For this 2-pump coastdown analyses, the input to the fuel rod heat conduction model in VIPRE were determined by sensitivity studies evaluating impact of pellet/clad gap, gas composition and pellet radial power profile to the DNBR. Results led to a conservative set of eight fuel parameters for the conduction model input.

The methodology described in the report is acceptable.

3.0 CONCLUSIONS

We find that the subject topical report, together with DPC responses, contains sufficient information to satisfy the VIPRE-01 SER requirement that each VIPRE-01 user submit a document describing proposed use, sources of input variables, and selection and justification of correlations as it relates to use by DPC for reload stead-state and pump coastdown analyses.

We further find that the manner in which the code is to be used for such analyses, selection of nodalization, models, and correlations provides, except as limited below, adequate assurances of conservative results and is therefore acceptable.

The following items are limitations regarding application of DPC-NE-2003:

1. An MDNBR limit of less than 1.18 with the BWC CHF correlation, as described in Section 5.13 of DPC-NE-2003, requires further justification based on broader CHF database for detailed review.

2. Studies presented in this report are performed using design data for Mark-BZ fuel assemblies, which are currently used in Oconee. Although the approach described in this report is acceptable, for future analysis of reloads which incorporate other fuel, DPC should assure that the VIPRE-01 computer code be used within the range of applicability.

3. The scope of this review and the applicability of findings are limited to DPC's use of VIPRE-01 for core reload steady-state and a two-pump coastdown transient analyses.

4.0 REFERENCES

1. "Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003, August 1988.

2. "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM Revision 2, EPRI, July 1985.

3. Letter from C.E. Rossi (NRC) to J.A. Blaisdell (UGRA), (Transmittal of VIPRE-01 Safety Evaluation Report), May 1, 1986.

4. Letter from H.B. Tucker (DPC) to USNRC, "Response to the Request for Additional Information," May 3, 1989.

5. Letter from H.B. Tucker (DPC) to USNRC, "Response to Questions Regarding Differences Between Duke Topical Reports DPC-NE-2003 and DPC-NE-3000," June 19, 1989.

6. "Duke Power Company Oconee Nuclear Station Reload Design Methodology," DPC-NE-1001A, Rev. 4, April 1981.

7. "BWC Correlation of Critical Heat Flux," BAW-10143P-A, April 1985.

8. "LYNX2-Subchannel Thermal-Hydraulic Analysis Program," BAW-10130A, October, 1976.

9. "Duke Power Company Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002A, October 1985.

Appendix B

Responses to Requests for Additional Information

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

May 3, 1989

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555
Attention: Document Control Desk

Subject: Oconee Nuclear Station, Docket Numbers 50-269, -270, and -287
Topical Report DPC-NE-2003, "Core Thermal-Hydraulic Methodology Using VIPRE-01"; Response To Request For Additional Information

I submitted, by letter of August 31, 1988 the subject Topical Report for NRC review. By letter dated March 22, 1989, the NRC staff requested additional information. Attached are responses to the staff's questions. Also attached are errata sheets, which correct various typographical errors and/or provide additional clarifying information. Upon approval of the Topical Report the entire document will be reprinted with the corrected pages.

Please note that the original submittal was a proprietary document, and my August 31, 1988 letter contained an affidavit attesting to that fact. The responses to the questions and the errata sheets should be considered part of the Topical Report, and should be withheld from public disclosure.

If we may be of any further assistance, please call Scott Gewehr at (704) 373-7581.

Very truly yours,

Hal B. Tucker

H. B. Tucker

SAG163/lcs

xc: S. D. Ebneter, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

P. H. Skinner
Senior Resident Inspector
Oconee Nuclear Station

Darl Hood, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Question 1.

Section 4.0 of the topical report describes the nodalization sensitivity study performed to demonstrate that the simplified core models to be used for licensing calculations are conservative relative to the more detailed model. (a) Was the study performed with the same thermal-hydraulic models and/or correlations to be used for licensing calculations? If not, identify those models and correlations which are not the same. (b) Would the use of different correlations and/or models lead to different nodalization sensitivity study results? Demonstrate the conservatism of the simplified core model with the final T-H models for licensing application. (c) It is understood that only the BWC correlation will be used for critical heat flux calculation. What do you intend to do if the core conditions are outside the ranges of applicability of the BWC correlation?

Response

(a) The nodalization study was performed using the same models and correlations that will be used for licensing calculations.

(b) The use of different correlations and/or models would not lead to different nodalization sensitivity study results. Sensitivity studies (turbulent momentum factor, void models, etc.) have been performed for the McGuire and Catawba Nuclear Stations for Westinghouse optimized fuel using both a large (75 channel) model and a simplified (8 channel) model. Both models gave essentially identical sensitivity study results and the same conclusions were drawn from the 75 channel and 8 channel results.

The conservatism of the simplified core model that will be used for licensing calculations is discussed in Sections 4.1 and 4.2 of the report.

(c) Following the methodology discussed in Section 6.0 of the report, all of the core conditions analyzed as a part of the generic Oconee thermal-hydraulic analysis are within the ranges of applicability of the BWC correlation. If conditions must be analyzed that are outside the range of the BWC correlation the NRC will be informed of the CHF correlation that will be used.

Question 2.

Section 5.5 states that the spacer grid form loss coefficients for the individual subchannels are determined analytically by the vendor from the overall grid form loss coefficient. Provide sufficient detail of the analytical determination of the individual subchannel form loss coefficients. Are these values for single or two phase flow.

Response

Spacer grid subchannel form loss coefficients are calculated by B&W Fuel Company using the grid loss evaluation program GRIL. The GRIL code is able to determine subchannel form loss coefficients analytically based on individual subchannel geometries and experimentally determined overall grid loss coefficients. Subchannels geometries are defined in GRIL by inputting dimensions, drag areas, and drag coefficients for the different objects which obstruct flow in the individual subchannels. These objects include such things as hard stops, spring stops, and spacer grid webbing. GRIL calculates grid loss coefficients based on single-phase flow with coolant flow information being input in the form of average coolant density, average kinematic viscosity, and average Reynolds number. Flow velocity in the rod gap is calculated by boundary layer theory using a universal velocity profile which relates dimensionless velocity to wall distance parameters at different flow regimes. Actual calculation of the subchannel loss coefficients in GRIL is an iterative process. For the first iteration, the channel flow velocities are assumed to be equal to the average velocity in the channel. Using the individual subchannel geometry and drag information, GRIL calculates individual subchannel loss coefficients, an overall grid loss coefficient, and new subchannel velocities. The iterative process continues until the calculated overall grid loss coefficient matches the experimental value. Comparisons made to laser doppler velocimeter (LDV) test results have shown that the subchannel velocity profiles calculated by GRIL agree well with experimental data.

Question 3.

Section 5.8.2 discusses the determination of the value (proprietary) of the turbulent mixing coefficient to be used for all Oconee Nuclear Station core thermal-hydraulic analyses based on vendor prediction of the mixing test results. Explain the process of vendor prediction of mixing test results and mixing coefficient, and explain how this correlates to the Oconee computation.

Response

In subchannel crossflow codes such as VIPRE-01, the turbulent exchange between subchannels i and j is defined by

$$w'_{ij} = \beta \, s_{ij} \bar{G}$$

where $\bar{G}$ is the average mass flux of the adjacent subchannels, s_{ij} is the width of the gap between subchannels, and β is the turbulent mixing coefficient. The mixing coefficient is usually obtained by performing tests using a heated bundle. A test specifically designed for B&W Mark-B fuel was performed by Columbia University. Single-phase subchannel mixing data were obtained from a 5x5 rod array by measuring subchannel exit temperatures for 57 tests covering the range of test conditions shown below. A least-squares statistic based on exit temperature differences was calculated to determine, in conjunction with a subchannel crossflow code, an optimum value of the turbulent mixing coefficient. The optimum value of β was found to be [ ] with a standard deviation of [ ] As a result of this test, B&W uses a value for β of [ ] for all Mark-B fuel crossflow analyses. Duke Power will also use a value of [ ] for all Oconee Nuclear Station core thermal-hydraulic analyses of Mark-B fuel.

Range of Test Conditions

System Pressure	2200 psia
Inlet Enthalpy	186.1 - 487.2 Btu/lbm
Average Heat Flux	0.179 - 0.539 MBtu/hr-ft^2
Average Mass Flux	1.072 - 3.519 Mlbm/hr-ft^2

Question 4.

Section 5.8.2 also discusses the selection of the turbulent momentum factor (FTM) from the sensitivity study performed with the FTM of 0.0, 0.8, and 1.0. (a) Justify the selected value which is not the most conservative value as shown in Table 5-4. (b) Explain how and why only the three values of FTM were selected for the sensitivity study.

Response

The turbulent momentum mixing between channels is included as a force in the momentum balance. The total axial force on the control volume due to turbulent mixing, F_n, is calculated as

$$F_n = -FTM\,\Delta x \sum_{k \varepsilon i} w'\ \Delta u$$

where w' is the crossflow per unit length, u is the axial velocity difference between the control volume under consideration and an adjacent one, and FTM is a constant correction factor to account for the imperfect analogy between turbulent transport of thermal energy and momentum. As discussed in the topical report, if the turbulent moment factor is 1.0, energy and momentum are mixed with equal strength. If FTM is 0.0, only energy is mixed by the turbulent crossflow. These two extreme values and the value recommended in ref. 1, FTM = 0.8, were studied to determine the effect that the turbulent momentum factor has on the MDNBR. As expected, FTM = 0.0 (no momentum mixing) yields the most conservative MDNBRs and FTM = 1.0 results in the least conservative MDNBRs (see Table 5-4). Battelle found in ref. 1 that DNBR is not sensitive to changes in the turbulent momentum factor and the results in Table 5-4 show that changing FTM from 0.0 to 0.8 changes the MDNBR by less than 1.5 %. Using FTM = 0.8 reasonably assumes that there is momentum mixing which benefits the hot channel, but not by the maximum possible amount. Thus, Duke Power has elected to use FTM = 0.8 because it is a reasonable value, is the recommended value in ref. 1 (which has been approved by the NRC), and the DNBR sensitivity to FTM is low as demonstrated by both Duke Power and Battelle. As discussed in the response to question 3, a conservatively low turbulent mixing coefficient will be used in all Oconee thermal-hydraulic analyses, thus the amount of turbulent mixing (energy and momentum) will be conservatively predicted.

Reference

1. J. M. Cuta, et al., "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores", EPRI-NP-2511-CCM, Vol. 1-5, Battelle Pacific Northwest Laboratories, July 1, 1985.

Question 5.

Section 5.10 states that a new routine is added to the VIPRE-01 code to generate axial power shapes with inlet, symmetric, or outlet peaks. Provide sufficient details of this routine.

Response

The axial power shape routine added to VIPRE-01 is based on the following mathematical constraints on an axial power shape:

(1) $$F(B) = PB$$
(2) $$F(M) = P$$
(3) $$F(E) = PE$$

(4) $$\text{Max } F(x) = F(M) = P$$

(5) $F(x)$ is continuous from (B,E)

(6) $F'(x)$ is continuous from (B,E)

(7) $$\frac{1}{E - B} \int_B^E F(x)\, dx = 1.0$$

where $F(x)$ = axial power shape as a function of the axial location, x

B,E = beginning and ending normalized location of the active length

P = axial peak

M = normalized axial location of the axial peak

PB,PE = axial flux at the beginning and ending location of the active length, respectively

Based on the constraints given above, the following generalized expression was developed

(8) $$F(x) = P + C(x - L)^I$$

where C = a constant based on the axial peak (P) and the axial flux at the beginning and ending location of the active length (PB,PE) and the respective axial locations (M, B, and E). Different expressions are used to determine C based on the axial location (x).

L = M, B, or E

I = integer relationships based on the axial peak (P) and the beginning and ending flux values (PB and PE)

Symmetric axial power shapes calculated using the new routine are compared with axial shapes calculated using the VIPRE-01 symmetric cosine routine for axial peaks of 1.2 and 1.5 in Figures 1 and 2. These

figures clearly show the agreement between the two methods of generating symmetric axial flux shapes. The new flux shape routine was added to generate skewed axial flux shapes. As discussed in Section 6.5 of the topical report, Maximum Allowable Peaking (MAP) limits are calculated for a range of axial peaks with the location of the peak varied from the bottom to the top of the core. As an example of the flux shapes used to calculate the MAP limits, three axial shapes are shown in Figure 3 for an axial peak of 1.3 at X/L = 0.3, 0.5, and 0.7.

FIGURE 1.

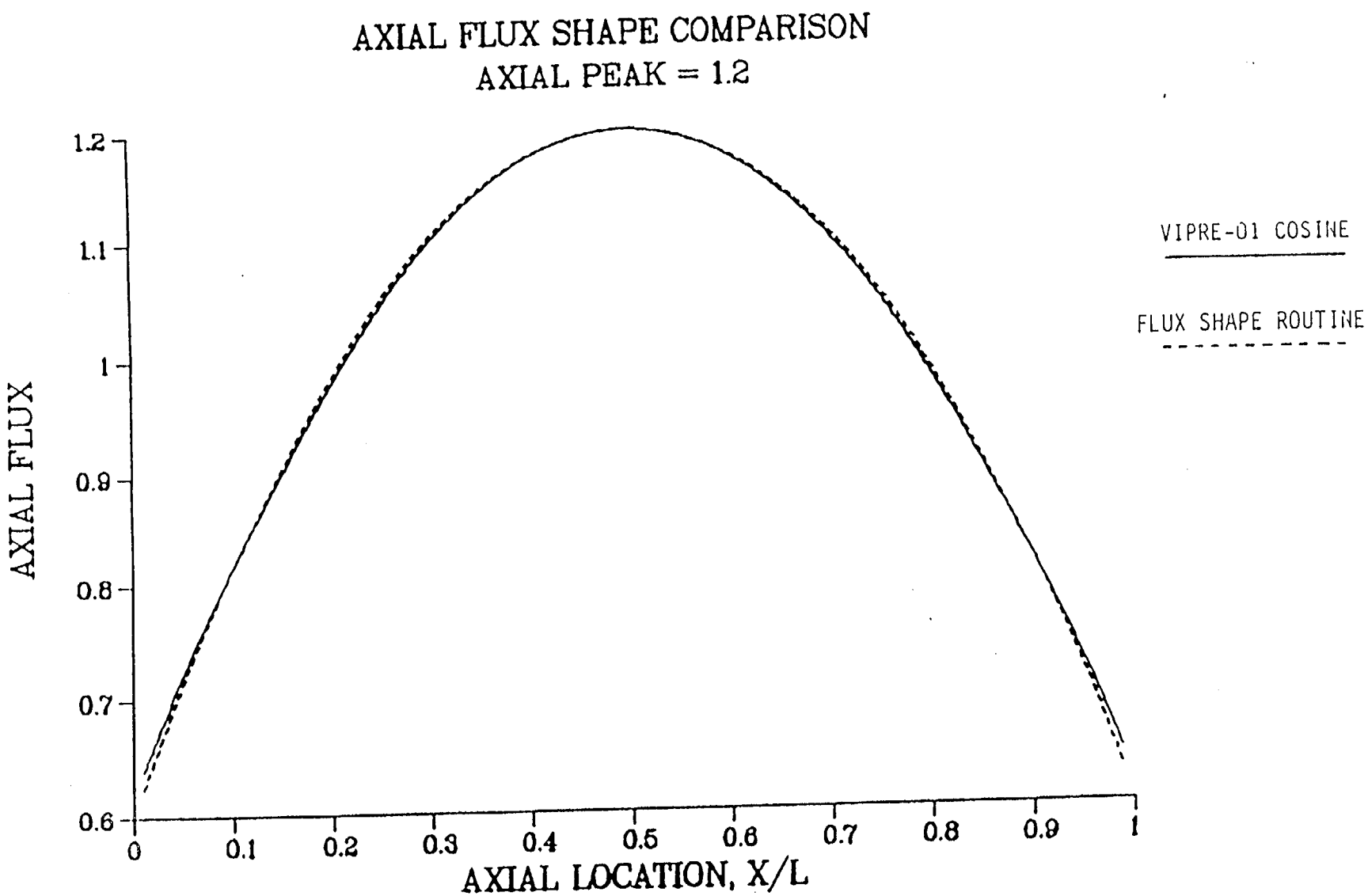

AXIAL FLUX SHAPE COMPARISON
AXIAL PEAK = 1.2
AXIAL FLUX
1.2
1.1
1
0.9
0.8
0.7
0.6
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
AXIAL LOCATION, X/L
VIPRE-01 COSINE
FLUX SHAPE ROUTINE

FIGURE 2.

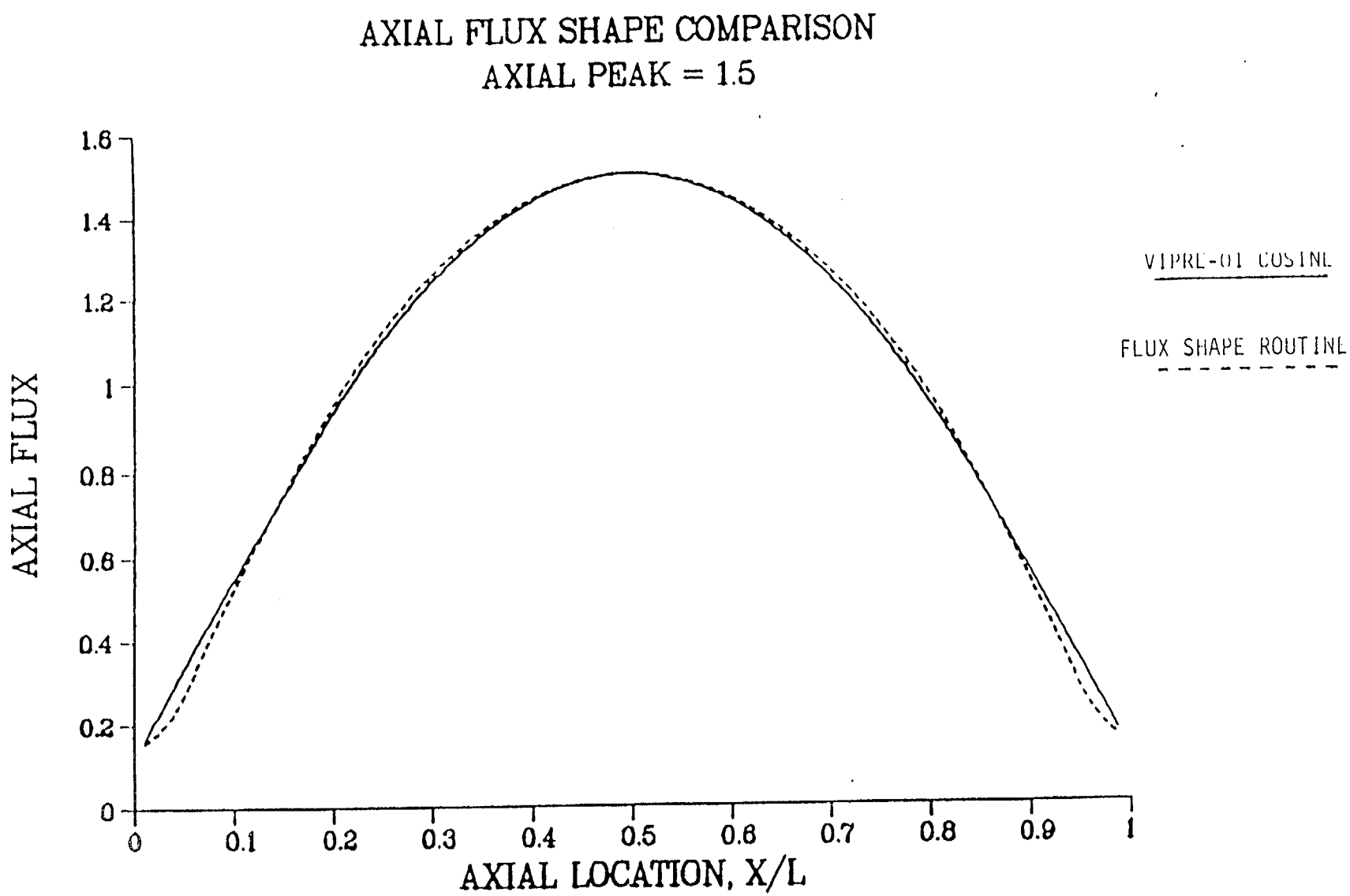
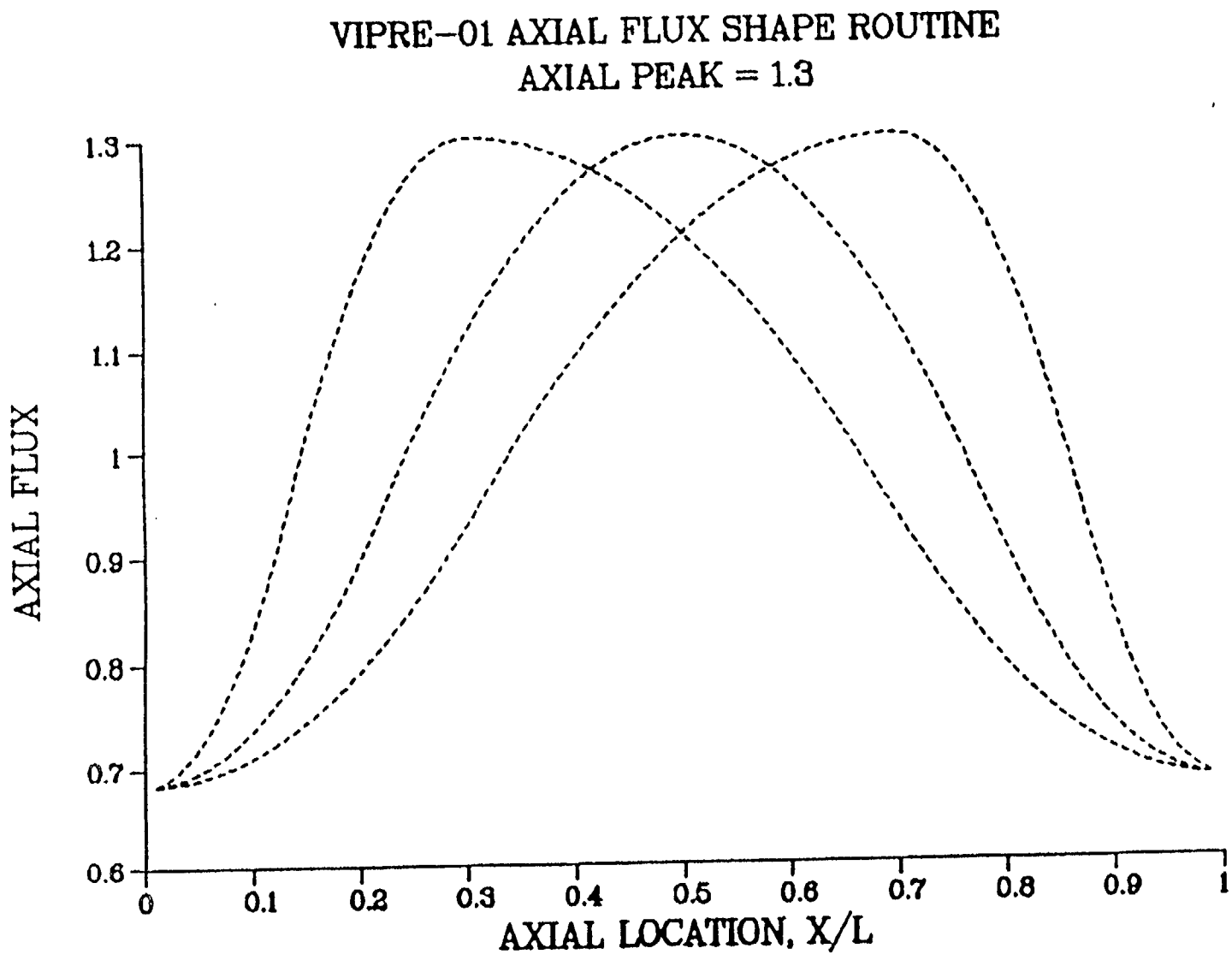

AXIAL FLUX SHAPE COMPARISON
AXIAL PEAK = 1.5
AXIAL FLUX
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
AXIAL LOCATION, X/L
VIPRE-01 COSINE
FLUX SHAPE ROUTINE

FIGURE 3.


VIPRE-01 AXIAL FLUX SHAPE ROUTINE
AXIAL PEAK = 1.3
AXIAL FLUX
1.3
1.2
1.1
1
0.9
0.8
0.7
0.6
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
AXIAL LOCATION, X/L

Question 6.

In Section 6.6, the inputs for the fuel gap conduction model are selected through a sensitivity study performed by varying three input parameters, i.e., pellet-cladding gap size, gas composition, and pellet radial power profile. Explain how this study enables the selection of conservative values of the eight parameters for input to the conduction model.

Response

Conduction through the gap between the fuel pellet and the clad is determined using the gap conductance model in VIPRE-01. This model is a simplified form of the models available in the FRAP and GAPCON codes. The NRC stated in the VIPRE-01 SER, ref. 1, that "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." To select the input for the conduction model (pellet diameter, gap width, etc.) sensitivity studies were performed using the 8 channel model discussed in Section 4.0 with a base set of conduction model input.

To investigate the sensitivity of the input gap width on the DNBR during a pump coastdown transient, three cases were run using the nominal, maximum, and minimum pellet/clad gap. The dynamic gap conductance model was used for all of the sensitivity studies. The dynamic gap conductance model calculates any changes in the gap width due to fuel rod deformation and fuel pellet thermal expansion, but it does not determine any changes due to densification, swelling, cracking, or pellet relocation. The maximum cold gap studied was calculated based on a conservative pellet densification and on manufacturing data for the pellet and clad diameters. The input gap width can be varied axially, but all of the cases assumed a constant gap width. The different gap widths were studied using the nominal clad ID and varying the pellet diameter.

The maximum pellet/clad gap case yielded the lowest MDNBR during a 2 pump coastdown transient. The large gap resulted in a lower gap conductance than that for the nominal gap, but the clad surface heat flux increased slightly when using the maximum gap (i.e., more energy was stored and then released at the time of MDNBR).

The gap width cases were run assuming that only helium and nitrogen gases were in the gap. An additional case was run assuming that fission gas had been released into the gap. The fission gas composition was taken from a typical TACO2, ref. 3, run at a burnup of 30,000 MWD/MTU. The VIPRE-01 results showed that the gap conductance and surface heat flux did not significantly change and the MDNBR did not change at all when assuming that fission gas was present in the pellet/clad gap. Since the maximum gap resulted in the lowest MDNBR during the pump coastdown transient and since the maximum gap would occur early in the burnup history of the fuel when peaking is highest, the generic pump coastdown analyses will assume that the gap is filled with only helium and nitrogen.

The base case for the sensitivity studies assumed that the power was uniformly distributed radially through the pellet. Cases were also run using a fuel pellet power profile from a typical TACO2 run. VIPRE-01 integrates the input power profile over the width of each node in the pellet to define the local volumetric heat generation rate. The MDNBR

results assuming a uniform power distribution or a power profile from TACO2 are essentially identical. The generic pump coastdown analyses will be based on a uniform pellet power distribution.

One additional case was run assuming a maximum pellet/clad gap based on the nominal clad OD and nominal pellet diameter and a reduced clad thickness. The MDNBR results for this case were identical to the case with the maximum gap based on the nominal clad OD and ID and a reduced pellet diameter.

The conduction model input that will be used for the generic pump coastdown analyses was selected based on the sensitivity study results discussed above. The input that results in a conservative pump coastdown analysis is listed in Section 6.6 of the topical report.

References

1. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA), "Acceptance for Referencing of Licensing Topical Report, VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores", EPRI-NP-2511-CCM, Vol. 1-5, May 1, 1986.

2. J. M. Cuta, et al., "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores", EPRI-NP-2511-CCM, Vol. 1-5, Battelle Pacific Northwest Laboratories, July 1, 1985.

3. Y. H. Hsii, et al., TACO2 - Fuel Pin Performance Analysis, BAW-10141, August 1979.

Question 7.

Section 6.6 also indicates that a sensitivity study shows very little difference in the pump coastdown results with regard to the choice of nucleate boiling correlation. Provide more detail of the sensitivity study performed to select the nucleate boiling correlation.

Response

VIPRE-01 contains a number of heat transfer correlations for each of the four commonly recognized models of heat transfer: single-phased forced convection, subcooled and saturated nucleate boiling, transition boiling, and film boiling. Since only conditions up to the point of DNB are of interest during a pump coastdown transient, the code can be restricted to consider only convection and nucleate boiling heat transfer, speeding up the solution procedure.

To quantify the effect of different heat transfer correlations on the local coolant conditions and MDNBR during a pump coastdown transient, the following nucleate boiling correlations were studied using the default EPRI forced convection correlation:

Subcooled Nucleate Boiling	Saturated Nucleate Boiling
THOM	THOM
THSP*	THSP
CHEN	CHEN

* Thom plus the EPRI single-phased forced convection correlation

The results given in Table 1 show that the choice of nucleate boiling correlations makes very little difference in the MDNBR during a two pump coastdown transient. The Thom subcooled and saturated nucleate boiling correlations, which yielded a conservative MDNBR, will be used along with the EPRI single-phased forced convection correlation for the generic Oconee pump coastdown analyses.

Table 1. VIPRE-01 Nucleate Boiling Heat Transfer Correlation Sensitivity Study

MDNBR BWC

Saturated & Subcooled Nucleate Boiling Correlations

Time sec.	THOM THOM	THSP THSP	CHEN CHEN
0.0	1.830	1.830	1.830
0.5	1.807	1.807	1.808
1.0	1.767	1.768	1.770
1.5	1.708	1.709	1.712
2.0	1.636	1.637	1.642
2.5	1.558	1.560	1.565
2.7	1.517	1.518	1.523
2.9	1.489	1.489	1.494
3.1	1.454	1.455	1.460
3.3	1.420	1.421	1.426
3.4	1.376	1.378	1.383
3.5	1.333	1.335	1.341
3.6	1.308	1.310	1.315
3.7	1.285	1.286	1.291
3.8	1.260	1.261	1.267
3.9	1.238	1.240	1.245
4.0	1.221	1.223	1.228
4.1	1.216	1.221	1.224
4.2	1.221	1.234	1.235
4.3	1.250	1.259	1.258

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

June 19, 1989

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Response to Questions Regarding
Differences Between Duke Topical
Reports DPC-NE-2003 and DPC-NE-3000

During a telecon on June 13, 1989, the NRC staff requested additional information to clarify the intended applications and other technical details regarding the VIPRE-01 models for Oconee submitted in DPC-NE-3000, Revision 1 and in DPC-NE-2003. This letter provides that information. In general, the VIPRE-01 models described in DPC-NE-2003 are applied in the thermal-hydraulic design of each reload core. The VIPRE-01 models described in DPC-NE-3000 are applied in the prediction of the minimum DNBRs resulting from FSAR Chapter 15 transients. A more detailed description of the applications of these models follows.

DPC-NE-2003 describes the VIPRE-01 models and methodology to be used for reload thermal-hydraulic analyses. The steady-state analyses that determine the thermal-hydraulic limits that define the regions of safe operation in terms of power level, reactor coolant temperature and pressure (Pressure-Temperature curves), and power distribution (RPS Maximum Allowable Peaking (MAP) limits) are described in this report. The steady-state analyses, based on the limiting two-pump coastdown statepoint, that determine the allowable power distribution during the limiting DNBR transient (Operational MAP limits) are also described. The methodology for determining the limiting statepoint during the two-pump coastdown transient is included. These analyses are routinely performed for a reload core to demonstrate that applicable safety criteria are met.

As discussed in DPC-NE-2003, two additional hot channel factors to account for power spikes due to spacer grids, and axial nuclear uncertainty are applied to the local heat flux factor, F_q'', only when calculating MAP limits. The two sets of MAP limits, RPS and Operational MAP limits, are used to demonstrate that peaking will be acceptable during steady-state operation and during anticipated transients. All other core thermal-hydraulic analyses (calculation of pressure-temperature curves, FSAR Chapter 15 analyses) are based on the reference design peaking given in the appropriate reports and F_q'' without the additional hot channel factors. This approach is consistent with the current application of hot channel factors in the NRC-approved methodology described in the Duke Power topical report NFS-1002. The use of the VIPRE-01 code has no impact on this approach.

The reference axial peaking (1.50) used in the two-pump coastdown transient is also used in the FSAR Chapter 15 transients to verify that the results are acceptable. The higher reference axial peaking factor (1.65) given in DPC-NE-2003 indicates the objective of using a higher value which results in less limiting Operational MAP limits. A higher reference axial peaking factor yields a lower two-pump coastdown MDNBR which results in higher allowable peaking. The methodology described in DPC-NE-2003 is applicable to any axial peaking assumption, provided that the resulting DNBRs and other peaking factor-related aspects are addressed. The current value of the reference axial peaking factor used in the MAP methodology is 1.50. Prior to increasing this value to, for example, 1.65, a complete evaluation of all potential safety concerns will be performed.

The VIPRE-01 SER states that "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." VIPRE-01 was used to predict the BWC CHF test results as discussed in Section 5.13 of DPC-NE-2003. The VIPRE-01/BWC results yield a DNBR limit of 1.161; thus, it will be conservative to use the NRC approved BWC correlation limit of 1.18 for all Oconee thermal-hydraulic analyses.

DPC-NE-3000 Section 2.3 describes the VIPRE-01 models to be used for predicting the minimum DNBRs resulting from FSAR Chapter 15 transients. The one exception is the two-pump coastdown described above, which is analyzed with the models described in DPC-NE-2003. The two-pump coastdown is a unique transient in that it is an integral part of the reload thermal-hydraulic design methodology. Therefore, the VIPRE model used for the two-pump coastdown should be the same model used for all other reload design thermal-hydraulic analyses. As discussed in DPC-NE-3000, Section 2.3.4, the VIPRE methodology for transient analyses includes a few differences when compared to the DPC-NE-2003 methodology. These differences when compared to the DPC-NE-2003 methodology. These differences are either necessary for meeting the modeling requirements of transient analyses, or incorporate additional conservatisms beyond those in the DPC-NE-2003 methodology. These additional conservatisms are desired in order to build margin into the transient DNBR results and avoid the need for reanalyzing transients in the future. It would be undesirable to use the DPC-NE-3000 VIPRE models as part of the normal reload thermal-hydraulic design process due to these differences.

In order to support the Oconee Unit 3, Cycle 12 reload licensing effort, an SER on DPC-NE-2003 is needed by August 15, 1989. If you have further questions regarding this matter, please contact Scott Gewehr (704/373-7581) or Gregg Swindlehurst (704/373-5176).

Very truly yours,

H. B. Tucker

SAG171/lcs

cc: L. A. Wrens, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. S. D. Ebneter, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
101 Marietta Street, NW - Suite 2900
Atlanta, Georgia 30323

Mr. P. H. Skinner
Senior Resident Inspector
Oconee Nuclear Station

Mr. D. Katze
Reactor Systems Branch
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Y. H. Hsii
Reactor Systems Branch
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. M. W. Hodges
Reactor Systems Branch
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555