WOLF CREEK
NUCLEAR OPERATING CORPORATION

Forrest T. Rhodes
Vice President
Engineering & Technical Service

May 29, 1992

ET 92-0103

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Reference: 1) Letter dated December 27, 1991 from W. D. Reckley, USNRC, to B. D. Withers, WCNOC
2) Letter ET 90-0140 dated August 21, 1990, from F. T. Rhodes, WCNOC, to the USNRC

Subject: Docket No. 50-482: Response to Request for Additional Information on the Core Thermal-Hydraulic Analysis Methodology for the Wolf Creek Generating Station

Gentlemen:

The purpose of this letter is to submit Wolf Creek Nuclear Operating Corporation's (WCNOC) response to a Request for Additional Information (RAI) provided in Reference 1. The RAI concerns WCNOC's "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station" which was submitted in Reference 2.

At a meeting on January 28, 1992, WCNOC and the Nuclear Regulatory Commission (NRC) staff discussed the questions in the RAI and agreed that WCNOC would provide a draft response to the staff and submit a formal response after further discussion with the staff. Following a telephone conservation on February 21, 1992, between Mr. W. D. Reckley, Project Manager, NRC and Mr. S. G. Wideman, WCNOC, a draft response to the RAI was provided by WCNOC. On April 28, 1992, Mr. Reckley indicated in a telephone conversation with Mr. Wideman that WCNOC should submit the formal response to the RAI. The Attachment provides WCNOC's formal response to the questions provided in Reference 1.

A001
1/1

If you have any questions concerning this matter, please contact me or Mr. S. G. Wideman of my staff.

Very truly yours,

Forrest T. Rhodes
Vice President
Engineering & Technical Services

FTR/jra

Attachment

cc: A. T. Howell (NRC), w/a
R. D. Martin (NRC), w/a
G. A. Pick (NRC), w/a
W. D. Reckley (NRC), w/a

**Response to Request for Additional Information for the Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station**

Wolf Creek Nuclear Operating Corporation (WCNOC) submitted "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station" Topical on August 21, 1990. Letter dated December 27, 1991, from the Nuclear Regulatory Commission (NRC) requested WCNOC provide additional information on the topical. At a meeting on January 28, 1992, WCNOC and the NRC staff discussed these questions with WCNOC agreeing to submit a draft response and to submit a formal response after further discussion with the staff. Following a telephone conversation on February 21, 1992, between Mr. W. D. Reckley, NRC Project Manager, and Mr. S. G. Wideman, WCNOC, a draft response to the questions was provided to the NRC staff. During a telephone conversation on April 28, 1992, between Mr. W. D. Reckley and Mr. S. G. Wideman, Mr. Reckley indicated that WCNOC should transmit the formal response to the request for additional information. Provided below is the NRC requested information with WCNOC's response immediately following.

**<u>Request 1</u>: Provide justification for the assumption that the three-step radial power distribution and the modified Core Statepoint-1 used in determining the core-wide protection limits are bounding. What evaluation will be performed to confirm these assumptions for a specific reload cycle?**

<u>Response</u>:

The analysis of a number of "real" power distributions and a bounding three-step distribution were used to demonstrate that the hot pin statistical design limit (SDL) provides a conservative means of demonstrating core-wide protection. The power distributions used for this evaluation were chosen to be representative of limiting actual power shapes but were not considered to be bounding. The three step power distribution was chosen to be bounding in terms of the number of pins with peaking factors at high enough levels to be limiting. The evaluation of this power distribution demonstrates that, even with one-third of the core at the design peaking limit, the statistical design limit derived from the hot pin protection criteria is limiting.

The conservative nature of the core-wide protection results obtained with the modified Core Statepoint-1 was assured by examination of each of the core states described in Table 4-13, TR-90-0025 for which the mean Departure from Nucleate Boiling Ration (DNBR) represents an allowable core state. For example, the mean DNBR at core state 5, Table 4-13 was approximately 1.2 while the corresponding SDL is 1.33. The use of unallowable core states for the determination of the hot pin SDL is acceptable since the relevant parameter is the uncertainty on DNBR, not the absolute magnitude of DNBR. However, for core wide protection, core peaking distributions must be analyzed only for core states that feature a mean DNBR that is greater than or equal to the hot pin statistical design limit. Results of the analysis of these additional core states indicate that the modified core statepoint-1 yields a limiting core-wide SDL.

Inherent in the selection of the core state points for use in establishing both the hot-pin and core-wide SDL is the definition of the midpoint or "nominal" condition state point. Small changes in nominal or design values of the core state variables included in the determination of the statistical design limit are accounted for in the parameter ranges used to define the $\alpha$ points on the response surface model (RSM) and by the large variations in core states used in establishing the DNB protection criteria. Evaluations of the design values of the core state variables along with their associated uncertainty ranges are made by WCNOC during the reload safety evaluation for each cycle design. Should a change in a plant parameter be made which would significantly affect the definition of the center point on the RSM, (i.e., power uprate, large reduction in thermal design flow, large increase in design $F_{\Delta H}$), it would be necessary to perform an analysis to demonstrate that a core wide statistical design limit determined based upon a RSM referenced to some previous "nominal" core state point remains valid or to develop a new RSM that represents the new midpoint values for the core state variable and subsequently, to establish a new statistical design limit.

<u>Request 2</u>: **How will it be ensured that the power distribution assumed in the 17-Channel Model (of Section-3) used to determine the DNBR safety limit lines are bounding for a specific reload cycle?**

<u>Response</u>:

The design power distribution described in the development of the base thermal-hydraulic model was derived from scrutiny of actual power distributions for the Wolf Creek Generating Station (WCGS). The bounding nature of this distribution will be verified for each reload cycle through three mechanisms. First, the maximum allowable peaking (MAP) limits are utilized in a physics maneuvering analysis in which predicted three dimensional peaking distributions throughout cycle life are compared to the peaking limits defined by the MAP's. Core power distributions are influenced by cycle specific design parameters and operational conditions. During cycle operation the core power distribution is dependent primarily upon fuel depletion, power level, control rod position, and xenon distribution. In the maneuvering analysis, peaking distributions resulting from power operation over the entire expected range of these parameters are calculated. The peaking distributions which challenge the maximum allowable peaking limits are identified and core thermal-hydraulic check cases using the predicted power distributions are analyzed to confirm that positive margin to peaking limits is maintained. Secondly, a check is made to guarantee that the peak predicted $F_{\Delta H}$ is less than the design limit (i.e., 1.55) assumed in the design distribution. Finally, a check is made on the pin-to-box factors for predicted peaking distributions for each reload to insure that the 1.05 peak-to-average from the design distribution remains bounding.

**Request 3: How will cycle-to-cycle variations in fuel design and core loading be accounted for in the VIPRE-01 model?**

Response:

Typically, a reload design will feature fuel which is functionally identical to the fuel resident in the core. The design parameters for the fuel for a proposed reload are verified during the reload safety evaluation process. Should a fuel design change occur, such as the addition of intermediate flow mixing grids, modifications to the base thermal-hydraulic model would be made which reflect the design change. Appropriate sensitivity studies would be made to demonstrate that the model yielded a conservative thermal-hydraulic environment for use in the core thermal-hydraulic design. It should be noted that significant model changes would require that analyses be performed to either confirm the statistic design limit or which establish a new SDL. This is because the RSM is optimized to reflect the DNB response for a particular VIPRE-01 model. When this model is changed, the DNB response of the VIPRE-01 model would be expected to also change.

The effect of changes in core loading primarily affects peaking distributions in the core. These effects are considering in the physics maneuvering analysis (see response to request 1).

**Request 4: How are assembly rod-wise power distributions that are not octant symmetric, due either to fuel design or global core power distribution, accounted for in the VIPRE-01 model?**

Response:

The effects of assembly rod-wise power distributions which are not octant symmetric are accounted for in the plant physics maneuvering analysis. Penalty factors are applied on calculated power distributions to account for core tilt. The augmented power peaking is then compared to the maximum allowable peaking limits established in the core thermal-hydraulic analysis. It should be noted that core loading patterns for WCGS are designed such that 1/8 core symmetry will be maintained.

**Request 5: Does the VIPRE-01 axial representation assume that the minimum DNBR (MDNBR) occurs between the 68 and 130 inch elevations and, if so, how are situations where the MDNBR occurs outside this region treated?**

Response:

The axial noding scheme examined in the development of the base thermal-hydraulic does assume that the minimum departure from nucleate boiling ratio will occur within the span from 68 to 130 inches. The combination of core thermal conditions and power distributions typical of most core thermal-hydraulic evaluations will place the point of MDNBR within this axial range. However, certain transients may feature unusual thermal conditions or power distributions which could cause the elevation of MDNBR to fall outside the range in which the axial node size in base thermal-hydraulic model is small. Example of transients of this nature

would include the steam line break event and the rod withdrawal from subcritical events. For core thermal-hydraulic analysis of events in which the point of MDNBR is determined to fall outside the 68 to 130 inch elevation, modifications to the axial noding in base thermal-hydraulic model will be made to insure accurate resolution of the solution. These modifications will be made using the same guidelines used to develop the base thermal-hydraulic model as described in the WCNOC submittal.

**<u>Request 6</u>: Provide the basis for the uncertainties and the assumed (normal and uniform) distributions for the variables given in Table 4-12.**

**<u>Response</u>:**

The uncertainties propagated through the RSM to establish the statistical design limit were selected to reflect or bound plant specific uncertainties for WCGS. The distributions assumed for these uncertainties were either normal or uniform. When a clear basis for use of the normal distribution was uncertain, the more conservative uniform distribution was used.

The 2% of rated thermal power uncertainty applied to the core power accounts for the error associated with a normalization of the reactor primary side power indication to a secondary side calorimetric heat balance and the error can be shown to conform to a normal distribution (Reference 6.1)

Reactor coolant flow is measured by elbow meters located in the cold leg of each primary system loop. The elbow meters are normalized to a precision calorimetric measurement of flow at the beginning of each cycle. The reactor coolant flow uncertainty is determined from the elbow meter differential pressure measurement uncertainty, the precision calorimetric measurement uncertainty, and the Reactor Coolant System (RCS) pressure and temperature measurement uncertainty. Evaluations of the uncertainties associated with the measurement of reactor coolant flow have conservatively established 2.5% as the flow uncertainty for WCGS. A uniform distribution was used for the propagation of the reactor coolant flow uncertainty to insure conservative results.

The bypass flow uncertainty results from uncertainties in the calculation of mass flow rate through each bypass flow path. Five flow paths exist for coolant flow to bypass the core. The outlet nozzle leakage describes the flow that goes directly from the inlet nozzle in to the outlet nozzle through the gap between the vessel and the outlet nozzle. The baffle-barrel region bypass flow is an intentional bypass flow path which provides for cooling of the baffle, formers and core barrel. The upper head cooling spray provides a third path for coolant flow to bypass the core. The upper head cooling spray injects flow into the upper head to reduce relative motion of the upper head at vessel fitup and to minimize wear on the spring seal. Thimble cooling bypass flow describes the coolant flow that bypasses the core to provide cooling for core component rods. The final bypass flow path is the cavity flow. Cavity flow is that portion of the total coolant flow

which passes between the outer fuel assemblies and the baffle (cavity gap). The uncertainty on bypass flow arises from the uncertainties in calculating the flow through each of these paths (i.e., uncertainty in form loss coefficients, etc.). Plant specific analyses of the bypass flow uncertainty for WCGS indicate that the 1.5% used to establish the statistical design limit will conservatively bound the actual bypass flow uncertainty. A uniform distribution was used for the propagation of the bypass flow uncertainty to insure conservative results.

Pressurizer pressure is controlled by comparison of the measured vapor space pressure and a reference value. The uncertainty in the RCS pressure includes allowances for the pressure transmitters, process racks, and controller and pressure overshoot/undershoot due to the interaction and thermal inertia of the heaters and spray. A plant specific evaluation of the uncertainties associated with the RCS pressure control indicates that the 30 psi used to establish the statistical design limit conservatively bounds the actual uncertainty for WCGS (Reference 6.1). A uniform probability distribution was applied to the pressure uncertainty for conservatism.

The average coolant temperature is controlled by a system that compares the auctioneered high $T_{avg}$ from the four coolant loops with a reference value. The reference is derived from the turbine impulse chamber pressure. Allowances for uncertainties are made for the Resistant Temperature Detectors (RTDs), process racks, and turbine pressure transmitter. A plant specific analysis for WCGS has established 4.85 °F as the uncertainty on the average temperature control for the RCS. Since the uncertainty on coolant temperature treated in the determination of the statistical design limit only accounted for a 4.0 °F uncertainty, a 0.85 °F penalty will be compounded in all core thermal-hydraulic analyses. Note this discussion applies only to the errors associated with the $T_{avg}$ allowance for controller deadband and measurement allowance and does not include the margin allocation for steam generator tube fouling. An additional 1.65 °F penalty on RCS coolant temperature must also be compounded in all core thermal-hydraulic analyses to account for the steam generator tube fouling margin. The uncertainty on $T_{avg}$ was applied assuming a uniform probability distribution for conservatism.

The calculational and measurement uncertainties on radial peaking, axial peaking, and elevation of the axial peak (i.e., R, A, & Z) were selected to bound the actual uncertainties for WCGS. A total uncertainty on total peak of 7.5% (5.0% applied to the radial component and 2.5% applied to the axial component) conservatively bounds the actual uncertainties for WCGS (Reference 6.2). These uncertainty parameters have been shown to conform to a normal distribution probability function (Reference 6.3) The 3.0% hot channel factor uncertainty and 1.5% initial bundle spacing penalty are plant specific values applicable to the 17 x 17 fuel assembly geometry currently in used at WCGS (Reference 6.4). The hot channel factors have been shown to be well represented by a normal probability distribution function (Reference 6.5), while the penalty due to initial bundle spacing was represented with a uniform probability distribution function for conservatism. The 8 inch uncertainty on the location of the axial peak is a consequence of the

nodal codes used to perform neutronic calculations for WCGS. These codes typically use a node size of 4 inches. Thus an eight inch uncertainty was applied to conservatively bound the nodal uncertainty. This uncertainty has been shown to be represented by a normal probability distribution function (Reference 6.3).

The uncertainty applied for the WRB-1 critical heat flux correlation was derived from the statistical performance of the correlation in the VIPRE-01 code. The calculation of the level of uncertainty for the WRB-1 correlation is given in TR-90-0025. This uncertainty has been shown to be represented by a normal probability distribution function (see response to item #14). The uncertainty on the VIPRE-01 code could be considered to be included in the WRB-1 correlation uncertainty and is therefore accounted for twice in the current application. However, a 5.0% uncertainty was retained for the VIPRE-01 code as an additional conservatism using a normal probability distribution. The RSM to VIPRE-01 fit uncertainty was derived from the statistical performance of the optimized RSM. Calculation of the level of uncertainty for incorporation in the statistical design limit is given in TR-90-0025. This error has been shown to be represented by a normal probability distribution functio

<u>References</u>:

6.1 Carroll, M. E., "WCNOC Nuclear Safety Analysis Setpoint Methodology for the Reactor Protection System", TR-89-001, Wolf Creek Nuclear Operating Corporation, June 1989.

6.2 Jackson, E. W. et al., "Qualification of Steady State Core Physics Methodology for Wolf Creek Design and Analysis", TR-91-0018, Wolf Creek Nuclear Operating Corporation, December 1991.

6.3 Hassan, H. A., et al., "Power Peaking Nuclear Reliability Factor", BAW-10119-P-A, Babcock & Wilcox, Lynchburg, Virginia, February 1979.

6.4 Olson, C. A., "Hot Channel Factors", Westinghouse Electric Corporations, August 1991.

6.5 Chelemer, H, Boman, L. H, and Sharp, D. R., "Improved Thermal Design Procedure", WCAP-8567, Westinghouse Electric Corporation, July 1975.

**<u>Request 7</u>: How do the Wolf Creek Nuclear Operating Corporation (WCNOC) statistical core design (SCD) and response surface methodology differ from the methods described in Reference-16 of the topical report? Explain any differences.**

**<u>Response</u>:**

The WCNOC statistical core design and RSM are based entirely upon the methods described in BAW-10170-P-A. The form of the RSM equation used by WCNOC is identical to that given in the reference. This equation, which contains linear, quadratic, and cross product terms, is comprised

of a maximum of 36 coefficients and involves each of the core state variables. Identical criteria, based upon a central composite design methodology, are utilized in the selection of points for the optimization of the RSM coefficients. The propagation of the uncertainties for the hot pin protection is accomplished using the B&W developed SDLHOT code as described in BAW-10170-P-A. The propagation of uncertainties for core wide protection was accomplished using a modified version of the SDLCORE code.

As was done in BAW-10170-P-A, the WCNOC base thermal-hydraulic model (i.e., VIPRE-01 model) utilized an intrabundle peaking distribution for the hot bundle and then a lumped representation of the remainder of the core in a single pin. For the core wide analysis, WCNOC also applied the hot bundle intrabundle peaking distribution to the predicted radial peak for each fuel assembly in the core. However, as listed in the reference, the B&W core protection code, SDLCORE, does not conservatively account for the highly peaked internal peaking distributions typically found in assemblies with lower powers. Specifically, as it is listed in the reference, the B&W core protection code establishes estimates of pin powers by multiplying pin radial local factors from the design intrabundle peaking distribution with each assembly average power. In the B&W implementation, this results in a peak pin power of 1.05 times the assembly average power. However, as stated in BAW-10170-P-A, the "colder" assemblies typically have internal peaking distributions which are more highly peaked than the design distribution. To conservatively predict the number of pins that could experience DNB in the core wide protection analysis, a modification was made to the SDLCORE core protection code which scales the pin-by-pin radial local peaking factors for the design distribution with the pin-to-box factor for each assembly power. Thus, in the WCNOC application of the core wide protection methodology to actual peaking distributions for WCGS, the number of pins that might contribution to the total number of pins in DNB is established by applying the design hot bundle intrabundle peaking distribution to each assembly in the core while also making the assumption that the highest power rod in each assembly serves as the reference for predicting the number of pins in DNB.

<u>Request 8</u>: **In the selection of the random variables from the normal and uniform distributions, are values greater than the 95 percent points selected? If not, how is this simplification accounted for?**

<u>Response</u>:

The normal and uniform distribution models which were used in the Monte Carlo propagation of uncertainties on the core state variables are identical to the models used in BAW-10170-P-A. These models, based upon discrete 11 point model generators, have been shown to exhibit excellent attributes for normality and uniformity. As compared to the ANSI standard on normality (Reference 8.1), the sample distributions obtained from the normal distribution model conform to standards of normality even at the most restrictive level (i.e., 20%). Examination of Table 3-3 in BAW-10170-P-A indicates that the 11 point distribution models will indeed result in the selection of points greater than the 95% level.

References:
9.1 American National Standard, "Assessment of the Assumption of Normality (Employing Individual Observed Values)", ANS 15.15-1974, Ameri National Standards Institute, 1974.

**Request 9: Provide justification for the use of the K=1.724 95/95 upper tolerance factor for the RSM fitting error. What error is introduced by this assumption?**

Response:

The use of 1.724 for the Owen's one-sided tolerance factor was based upon the number of experimental points in the WRB-1 correlation database. This was an added conservatism in the WCNOC application of the B&W Statistical Core Design methodology in that in the reference application, an Owen's one-sided tolerance factor of 1.645, corresponding to an infinite population, was used to establish the error on the thermal-hydraulic code to RSM fit.

Argument can be made for the use of a tolerance factor corresponding to the number of points used to derive the RSM to thermal-hydraulic code fit. This would imply the use of a K factor equal to 1.982 for the 73 points in the current application. This would yield a RSM to VIPRE-01 error of 4.05%. Since a 4% error was used to define the error from the RSM to VIPRE-01 fit in the WCNOC application, use of the larger K factor would have negligible effect on the hot-pin statistical design limit.

**Request 10: What evaluation will be performed to ensure that the statepoints used in determining the hot-pin protection statistical design limit are bounding for a specific reload cycle?**

Response:

The state points used in the determination of the hot-pin statistical design limit were chosen to cover the anticipated range of operating limits for WCGS. Both high and low pressure reactor protection system limit state points were analyzed at design overpower conditions. Nominal condition state points were also examined with both symmetric and outlet peaked axial power distributions. Additional cases were also examined in which the core power was increased beyond the operating range and inlet temperatures were artificially increased to yield minimum DNB ratios near the design limit. Finally, the midpoint, or nominal state point from the development of the RSM was analyzed at a low flow condition and at a high power condition.

The purpose of the analysis of many different state points in establishing hot-pin protection was to maximize the coefficient of variation resulting from the error propagation. Results shown in Table 4-14 of the topical indicate that the overall coefficient of variation shows little change compared to the large changes in core states examined. It is therefore concluded that reasonable maximization of the coefficient of variation, and subsequent maximization of the hot-pin statistical design limit, was achieved.

Inherent in the selection of the core state points for use in establishing the hot-pin SDL is the definition of the midpoint or "nominal" condition state point. Small changes in nominal or design values of the core state variables included in the determination of the statistical design limit are accounted for in the parameter ranges used to define the α points on the RSM and by the large variations in core states used in establishing the hot-pin SDL. Evaluations of the design values of the core state variables along with their associated uncertainty ranges is made by WCNOC during the reload safety evaluation for each cycle design. Should a change in a plant parameter be made which would significantly affect the definition of the center point on the RSM, (i.e., power uprate, large reduction in thermal design flow, large increase in design $F_{\Delta H}$), it would be necessary to perform an analysis to demonstrate that a hot-pin statistical design limit determined based upon a previous "nominal" core state point remains valid or to develop a new RSM which represents the new midpoint values for the core state variable and subsequently, to establish a new hot-pin statistical design limit.

**Request 11: Are the WCGS fuel designs to which the WRB-1 correlation will be applied included in the presently approved applications of WRB-1/THINC?**

Response:

WCNOC will apply the WRB-1 critical heat flux correlation only to the analysis of fuel designs approved by the Commission in the Safety Evaluation Report of WCAP-8762. For WCGS these fuel types might include the 17 x 17 standard and 17 x 17 optimized fuel assemblies (OFA).

**Request 12: Why is the correlation design limit to be used with WRB-1 in VIPRE-01 proprietary? The design limit value is given in WCAP-8567, as well as in the MR SER, without proprietary brackets, and therefore should not be considered proprietary in this topical report.**

Response:

Proprietary brackets will be removed from both the design limit and standard deviation in the WCNOC topical report.

**Request 13: Why does the number of data points and test series given in WCAP-8762 differ from the number given in Table 2-4? Please justify the use of the smaller number.**

Response:

In response to the Commission's request for additional information during the review of WRB-1 topical report, Westinghouse submitted Supplement 1 to WCAP-8762. This supplement documented additional critical heat flux test data obtained by Westinghouse at Columbia's Heat Transfer Research Facility. This new data was obtained to extend the WRB-1 correlation to 14x14 OFA fuel assembly designs. As stated in the supplement, "DNB testing of the 14x14 OFA typical cell geometry has

shown that the WRB-1 correlation correctly accounts for the geometry changes from the reference design, using the same performance factor as used for the 0.422 inch rod evaluations. The 14x14 OFA data can be added to the WRB-1 R-grid database without changing the DNBR limit of 1.17." Table 3 of this supplement summarizes the WRB-1 database after the addition of the new 14x14 test data. As indicated, the new database consists of a total of 1108 test runs with a mean M/P ratio of 1.0079 and a sample standard deviation of 0.0859. This is the database used in the qualification of the WRB-1 critical heat flux correlation for use with the VIPRE-01 code.

**Request 14: What tests have been performed to ensure that the M/P data is normal?**

Response:

The D' test, based upon the development by D'Agostino (Reference 14.1), was used to ensure that the distribution of the measured to predicted critical heat flux data would approximate a normal distribution. Referring to ANSI N15.15-1974 (Reference 14.2), the critical values for a population of 1108 members at the 5% level of significance are 10344 and 10454. The D' value calculated for the M/P data obtained from the qualification of WRB-1 in the VIPRE-01 code was 10377, indicating that assumption of normality may be accepted.

References
14.1 D'Agostino, R. B. "An Omnibus Test of Normality for Moderate and Large Size Samples", *Biometrika*, Volume 58, 1071, pp. 341-348.

**Request 15: Is the procedure used to determine the steam generator safety valve (SGSV) line (Equation 5-2) the same as is presently used?**

Response:

The formation of the equation defining the steam generator safety value line given in the Core Thermal-Hydraulic Analysis Methodology topical report is the same as is presently used to establish the core thermal limit line for WCGS (Reference 15.1). The steam generator safety valve line is an integral part of the definition of the core thermal limit lines due to the physical limit on reactor power and temperature placed on the plant due to the action of the steam generator safety valves. The temperature drop from the steam generator primary to secondary is proportional to the power transferred. Since the maximum secondary side temperature is constant at the saturation temperature corresponding to the lift pressure of the steam generator safety valves, the primary temperature cannot rise above this saturation temperature plus the temperature drop across the generator tubes. Therefore, the steam generator safety valve line defines one of the boundaries on core power and temperature in the core thermal limits.

References

15.1 Ellenberger, S. L., et al, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions", WCAP-8745-P-A, Westinghouse Electric Corporation, September, 1986.

**Request 16: Are the WCNOC procedures for determining the MAP curves the same as the Babcock and Wilcox Fuel Company methods? If not, discuss any differences.**

Response:

WCNOC utilizes the same procedures as BWFC to produce the Maximum Allowable Peaking curves for use in the plant maneuvering analysis.

**Request 17: Is the part-power multiplier used below 75 percent power? If so, provide the basis.**

Response:

WCNOC has performed analyses which indicate increasing DNB margin at powers below 75% as compared to the part power multiplier line as shown in Figure 6-14 of TR-90-0025. Below 75% power, limitations on the hot channel exit quality imposed by the range of applicability of the critical heat flux correlation place limits on the total peaking allow however, studies show that sufficient margin exists at all powers below 100% to support the use of a part power multiplier coefficient of 0.3.

**Request 18: Do the three points on the core safety limit lines used to determine the MAP curves provide the most limiting MAPs? For example, since the low pressure MAPs are more restrictive, why wasn't the SGSV limit line MAP calculated on the low pressure DNBR limit line?**

Response:

The three points on the core safety limit lines used to determine the MAP curves have been shown to yield a limiting set of Maximum Allowable Peaking limits for WCGS. Examination of Figure 5-2, TR-90-0025 provides insight for not calculating a set of MAP limits at the intersection of the steam generator safety valve limit line and the low pressure DNB limit line. As shown, this intersection occurs well down on the 1860 psia limit line in the region where the plant is vessel exit boiling limited. The core ΔT at this intersection corresponds to a core power significantly less than 100% of rated thermal power. Since it has been shown that increased DNB margin exists with respect to peaking at powers below 100%, calculation of MAP limits at these conditions would be less restrictive that the MAP limits computed at the design overpower conditions.

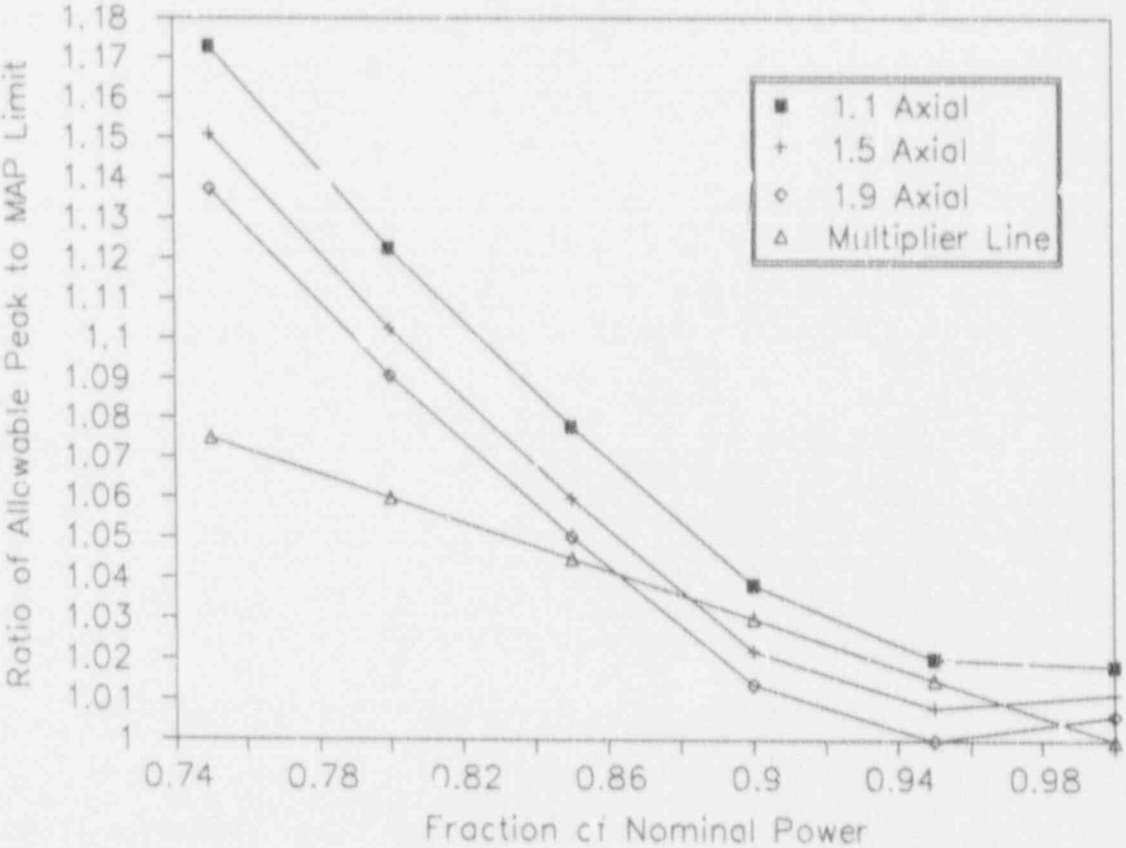
**Request 19: Why are the curves in Figures 6-13 and 6-15 different?**

Response:

The MAP limits shown in Figure 6-13, TR-90-0025 were adjusted, in the more restrictive direction, as a result of the analysis to establish the part power multiplier. As shown in Figure 19-1 below, use of a part power multiplier constant of 0.3 would result in slightly non-conservative results when the ratio of the power allowable peaks is compared to the unadjusted safety limit MAP's. To account for this, the limiting safety limit MAP's were reduced by 1.6%. As shown in Figure 6-14, TR-90-0025, this insures that the peaking restrictions imposed by the safety limit MAP's are conservative at all powers. Figure 6-15, TR-90-0025 is simply the maximum allowable peaking limits shown in Figure 6-13, TR-90-0025 reduced by 1.6%.

**WCNOC Response to RAI for TR-90-0025**

Figure 19-1




**Request 20: The Chen heat transfer correlation does not result in the highest fuel and clad temperatures in Table 3-59. How will conservative maximum fuel temperatures be calculated in specific transients?**

Response:

The Chen heat transfer correlation results in the highest fuel and clad temperatures when compared to the cases which do not enter a post-chf heat transfer regime. Comparison of the heat transfer coefficients for

case 1, Table 3-59, TR-90-0025, indicates that the Chen correlation does in fact yield the smallest heat transfer coefficient, 28% less than the Thom correlation and 32% less than Thom plus single phase. As a result, the fuel and cladding temperatures are highest for the Chen correlation for this case.

As would be expected, the Chen correlation does not yield the highest fuel and clad temperatures when compared to correlation sets that enter a post-chf heat transfer regime. Specifically, examination of the results from the 120% power case (i.e., case 2, Table 3-59, TR-90-0025) indicates that while the Chen correlation yields the highest fuel and clad temperature for the correlation sets that remain in the saturated boiling heat transfer regime, case HEAT1 enters the post-chf heat transfer regime and the resulting heat transfer coefficient, obtained with the G5.7 correlation, is significantly less than the Chen coefficient. The reason for the switch to post-chf heat transfer in case HEAT1 is the use of the Electrical Power Research Institute (EPRI) critical heat flux correlation to define the peak of the boiling curve (see table 3-58, TR-90-0025). As noted, the heat transfer mode was switched to the post-chf regime when the MDNBR calculated reach 1.50. Since the WRB-1 correlation generally yields higher DNB ratios than the EPRI correlation, cases HEAT2 through HEAT6 did not enter the post-chf heat transfer regime.

The Safety Evaluation Report issued for the VIPRE-01 code limits use of the code to heat transfer modes up to the point of critical heat flux. Thus, WCNOC will not attempt to predict fuel and cladding temperature when conditions result in heat transfer modes in the transition or film boiling regions.

Finally, the choice of an appropriate heat transfer correlation set in VIPRE-01 must be selected to yield conservative results for the parameter of interest. For departure from nucleate boiling ratio evaluations, heat transfer in the single phase region will be defined by the EPRI correlation, the subcooled and saturated nucleate boiling regions will be characterized by the Thom plus single phase correlations, and the peak of the boiling curved will be defined with the WRB-1 critical heat flux correlation. For fuel and cladding temperature evaluations, heat transfer in the single phase forced convection regime will again be characterized by the EPRI correlation while the subcooled and saturated nucleate boiling regime will be defined by the Chen correlation. Again, the point of critical heat flux will be determined using the Westinghouse WRB-1 correlation. No attempt will be made to use the VIPRE-01 code in any analysis in which the heat transfer mode is predicted to enter the transition or film boiling regions of the boiling curve.

**<u>Request 21</u>: Certain combinations of fluid correlations have not been included in the Section-3.3.4 comparisons. How will these cases compare to the base-case thermal-hydraulic model?**

<u>Response</u>:

The selections of correlation combinations for use in the flow correlation sensitivity study were made such that only correlations with consistent and complementary bases were examined. The objective of the fluid correlation study was to demonstrate that the calculation of a minimum departure from nucleate boiling ratio is relatively insensitive to the fluid correlation set used; not to examine every possible combination and permutation of the fluid correlations available in VIPRE-01. This was accomplished since the variation in calculated minimum DNB ratios was less than 3% for all correlation combinations with the exception of the case which utilized the Beattie two-phase friction multiplier correlation. The reasons for the variation observed with the Beattie correlation are well documented in the topical report.

The fluid sensitivity study found that the EPRI subcooled void, EPRI bulk void, and EPRI two-phase multiplier correlations are sufficient to yield accurate predictions of fluid conditions. Further, the qualification of the WRB-1 critical heat flux correlation using this combination of fluid correlations and the generation of acceptable statistical results as compared to the experimental data in the WRB-1 database, serves to confirm the selection of fluid correlations for use in the base thermal-hydraulic model.