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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE VIPRE-01 CODE AND WRB-1 CORRELATION

FOR FACILITY OPERATING LICENSE NOS. DPR-42 AND 60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated April 19, 1985 (Ref. 1), N thern States Power Company
(NSP) submitted Revision 3 to NSPNAD-8102P, "Reload Safety Evaluation
Methods for Application to Prairie Island Units," for staff review.
Revision 3 includes changes to: (1) replace the thermal-hydraulic hot
channel code COBRA-IIIC/MIT with the VIPRE-01 Code, (2) add the WRB-1
Critical Heat Flux (CHF) correlation for use with Westinghouse OFA fuel,
and (3) revise the acceptance criteria for Condition IV accidents. Our
evaluation with regard to these revisions follows.

2.0 STAFF EVALUATION

2.1 Use of VIPRE-01/WRB-1

The thermal margin analyses for the Prairie Island Units were done with
the COBRA-IIIC/MIT (Ref. 2) subchannel thermal-hydraulic code and the
W-3 CHF correlation. In the new reload methods, NSP proposes to use
VIPRE-01 (Ref. 3) in place of COBRA-IIIC/MIT and use the WRB-1 CHF
correlation in place of W-3.

VIPRE-01 is a subchannel thermal-hydraulic code developed by Battelle
Pacific Northwest Laboratories under the sponsorship of the Electric
Power Research Institute (EPRI). In December 1984, the Utility Group

for Regulatory Applications (UGRA), which consists of more than 20 utilities, submitted the VIPRE-01 topical reports for staff review. VIPRE-01 was developed from the COBRA series code including COBRA-IIIC (Ref. 4), COBRA-IV (Ref. 5), COBRA-IIIC/MIT and COBRA-WC (Ref. 6) by incorporating many features of these codes into one package. The staff review (Ref. 7) has concluded that the VIPRE-01 code is acceptable for PWR application with the following conditions:

(1) The application is limited to the heat transfer modes up to critical heat flux.

(2) An analysis is made to ensure that the minimum departure from nucleate boiling ratio (DNBR) limit of a CHF correlation used in VIPRE-01 can predict its data base of DNB occurrence with at least a 95 percent probability at a 95 percent confidence level.

(3) Documentation is submitted by each user to provide justification for the modeling assumptions, choice of particular two-phase flow models, correlations and input values of plant specific data, etc.

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

(5) Each user should abide by the quality assurance program established by EPRI for the VIPRE-01 code.

Appendix F to NSPNAD-8102P, Revision 3, provides a description of the intended use of VIPRE-01 by NSP, by comparing the analysis results using VIPRE-01 and COBRA-IIIC/MIT, and an analysis of the DNBR limit of the WRB-1 correlation and the NSP thermal margin methodology using VIPRE-01.

A summary of VIPRE input information is provided in Table F.5 regarding the single phase friction factor, two-phase friction multiplier, subcooled void and bulk void correlations, turbulent mixing model, transverse momentum factor and crossflow resistance factor. Most of the input values and correlations selected area consistent with those used in the previous analysis using COBRA-IIIC/MIT. The heat transfer correlations with respect to forced convection, and subcooled and saturated nucleate boiling chosen in VIPRE-01 and the reactor core modeling are also the same as for COBRA-IIIC/MIT as indicated in the NSP response to a staff question (Ref. 8). One difference from the COBRA-IIIC/MIT input is the subchannel turbulent mixing. Turbulent mixing was ignored in the previous NSP analysis due to the limitation that the COBRA-IIIC/MIT turbulent mixing model is not adequate for mixing between lumped assemblies. Since VIPRE-01 has the capability of specifying the turbulent mixing parameters for each channel connector, the analysis will consider the turbulent mixing effect between subchannels while ignoring turbulent mixing between lumped assemblies or lumped assemblies and subchannels. A turbulent momentum factor (FTM) of 0.8 will be used in the analysis. The turbulent momentum factor, which has a value from 0 to 1.0, is analogous to the turbulent Prandtl number. An FTM value of 0.0 indicates that the turbulent crossflow mixes enthalpy only and not momentum, and an FTM value of 1.0 indicates that it mixes momentum with the same strength as it mixes enthalpy. VIPRE-01 is not very sensitive to the value of FTM and the VIPRE-01 manual recommends that a value of 0.8 be used in the analysis. Therefore, the NSP approach is acceptable.

In order to assess the VIPRE-01 capability, Appendix F provides comparisons between the calculations of VIPRE-01 and the approved COBRA-IIIC/MIT code. The analyses were performed for Prairie Island-1, Cycle 9 using the W-3 CHF correlation. For steady-state calculations, comparisons are made between the VIPRE-01 and COBRA-IIIC/MIT analysis

results on the core safety limit and the axial offset effect. The core safety limits are the loci of points of thermal power, pressurizer pressure and inlet temperature for which the minimum DNBR limit is not violated. These core safety limit curves are then used in the derivation of the over temperature ΔT (OTΔT) trip setpoints to ensure that the DNBR limit will not be violated for normal operation and the anticipated operational transients.

Since the OTΔT setpoints are derived with zero axial offset, an axial offset penalty function $f(\Delta I)$ is used to lower the trip setpoints when highly skewed axial power shapes are encountered. The resulting core thermal limits (core inlet temperature as a function of thermal power and pressurizer pressure) and the average heat flux for DNB occurrence for various axial offsets are shown in Figures F.1, F.2, F.7 and F.8 with comparisons between the VIPRE-01 and COBRA-IIIC/MIT analysis results. These comparisons show that the VIPRE-01 calculations are either the same as or slightly conservative to the COBRA-IIIC/MIT calculations.

The transient comparisons between VIPRE-01 and COBRA-IIIC/MIT are made for the rod withdrawal at power, turbine trip, 2/2 pump trip, and locked rotor events. The results of minimum DNBR as a function of time are shown in Figures F.3 through F.6. The comparisons show that the calculations from both codes are essentially the same for rod withdrawal and turbine trip cases and the VIPRE-01 calculations are slightly more conservative for the pump trip and locked rotor cases. NSP attributed this conservatism to a slightly higher crossflow out of the hot channel in the VIPRE-01 calculations which leads to higher local quality and therefore lower CHF and DNBR. The overall comparisons show VIPRE-01 to be conservative relative to the approved COBRA-IIIC/MIT code and is therefore, acceptable.

The WRB-1 CHF correlation was developed by Westinghouse using the THINC thermal hydraulic code. WRB-1 has previously been reviewed and approved for application to the Westinghouse standard low parasitic fuel and the optimized fuel assembly (Ref. 9, 10, 11). A minimum DNBR limit of 1.17 is acceptable for both the standard R-grid assembly and OFA fuel designs. NSP obtained the WRB-1 correlation from Westinghouse and incorporated it into the VIPRE-01 code. Since the DNBR limit of 1.17 is acceptable for WRB-1 in connection with the THINC code, analysis must be done to show that the same limit provides at least the same degree of assurance in the DNB prediction when WRB-1 is used with the VIPRE-01 code.

NSP has re-analyzed the Westinghouse CHF test data obtained with bundle geometry representative of the 14x14 OFA design using the VIPRE-01/WRB-1 package. The results of the analysis for the measured-to-predicted CHF ratios (M/P) are shown in Table F.2. Based on these measured CHF to predicted CHF (M/P) ratios and the use of standard statistical methods widely used in the industry, a minimum DNBR limit is obtained which would ensure avoidance of DNB with 95 percent probability at a 95 percent confidence level. Since this DNBR limit is less than 1.17, the use of 1.17 as the DNBR limit for VIPRE-01/WRB-1 is acceptable. NSP has indicated that the WRB-1 correlation will be applied to the improved OFA fuel design to be loaded in the Prairie Island units. The improved OFA is essentially the same as the OFA design except for a six inch natural uranium axial blanket at the top and bottom of the fuel rod in the improved OFA. Other design features which are important to the CHF behavior such as grid design, grid spacing, pin diameter, etc., remain the same for both the OFA and improved OFA. Therefore, we conclude that the application of WRB-1 to the improved OFA is acceptable.

The staff has asked what steps will be taken by NSP to assure that only the approved version of VIRPE-01 will be used in licensing analyses.

NSP in its response (Ref. 8) stated that the VIPRE code will be controlled according to NSPNAD Policies and Procedures NAP5.001A, Revision 4, "Computer Program Control" which covers the use of codes and making modifications to the codes. This computer code control procedure has been audited by NRC in 1983. NSP provided a brief description of the control procedure which we find to be acceptable. However, this procedure deals with only the modifications to the code inside NSP. Since VIPRE-01 is an EPRI code developed by PNL to be used by the utilities belonging to the UGRA, improvements and modifications to the code will be made by the utilities other than NSP and PNL. Therefore, we will require that NSP also abide by the quality control procedures established by EPRI which the UGRA committed to follow for the VIPRE code.

2.2 Changes to the Acceptance Criteria of Condition IV Accidents:

Revision 3 to NSPNAD-8102P proposes to revise the acceptance criteria for a few Condition IV events including locked rotor, steamline break and control rod ejection events. The proposed revisions are as follows:

(1) Eliminate an existing criterion which states: "The number of fuel rods calculated to experience a DNBR of less than 1.3 should not exceed the number which are required to fail in order that the doses due to released activity will exceed the limit of 10 CFR Part 100. This limit is currently the maximum number of failed fuel rods calculated in the FSAR." The proposed revision also eliminates an equivalent criterion for reload analysis: "number of fuel pins above $F_{\Delta H}$ (DNBR = 1.3) < 20%."

(2) Change the maximum clad temperature limit from 2750°F to 2700°F.

The staff has found that the first revision with regards to the elimination of the released activity dose criteria conforming to 10 CFR Part 100 is not acceptable. In a telephone conversation on April 7, 1986, NSP decided that it will withdraw the proposed changes to the acceptance criteria including the change to the maximum cladding temperature.

Since the DNBR of 1.3 in the original acceptance criteria was for the W-3 correlation which does not reflect the DNBR limit of 1.17 for WRB-1, the licensee has agreed to re-install the original criteria into Revision 4 to NSPNAD with an exception that the DNBR limit of 1.3 (W-3) will be changed by adding a DNBR limit of 1.17 for WRB-1. We find this to be acceptable.

3. SUMMARY AND CONCLUSIONS

The staff has reviewed Revision 3 to NSPNAD-8102P. We find that the use of the VIPRE-01 subchannel thermal hydraulic code and the WRB-1 CHF correlation with a minimum DNBR limit of 1.17 for the Prairie Island units is acceptable. We will also require that NSP abide by the EPRI quality assurance procedure regarding the VIPRE-01 modifications.

With regard to the acceptance criteria for the Condition IV events, NSP has decided to withdraw its proposed revisions and has agreed to install the original criteria into Revision 4 to NSPNAD-8102P with an addition of a DNBR limit of 1.17 for the WRB-1 correlation. We find this to be acceptable.

Principal Contributor:

Hsii Yi Hsiung

REFERENCES

1. Letter from D. Musolf (Northern States Power Company), to Director, Office of Nuclear Reactor Regulation, USNRC "Proprietary (and ron-proprietary version) Topical Report Reload Safety Evaluation Methods, NSPNAD-8102P, Rev. 3," April 19, 1985.

2. Bowring, R. W., and P. Moreno, "COBRA-IIIC/MIT Computer Code Manual," Prepared by Massachusetts Institute of Technology for EPRI, March 1976.

3. EPRI-NP-2511-CCM, "VIPRE-01, A Thermal-Hydraulic Analysis Code for Reactor Cores," Volumes 1 through 3, EPRI, April 1983, Revision 1, November 1983, Revision 2, July 1985.

4. D. S. Rowe, "COBRA-IIIC: A Digital Computer Program for Steady-State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements," Richland, Washington: Pacific Northwest Laboratory, March 1973, BNWL-1695.

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

6. T. L. George, et al., "COBRA-WC: A Version of COBRA for Single-phase Multiassembly Thermal-Hydraulic Transient Analysis," Richland, Washington: Pacific Northwest Laboratory, July 1980, PNL-3259.

7. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (NUSCO), "Acceptance for Reference of 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.

8. Letter from D. Musolf (NSP) to Director, Office of NRR, USNRC, "Responses of NRC Questions on Revision 3 of Topical Report NSPNAD-8102P," November 19, 1985.

9. WCAP-8762(P), WCAP-8703(NP), "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1976.

10. Letter from R. L. Tedesco (NRC) to T. M. Anderson (Westinghouse), "Acceptance for Referencing Topical Report-WCAP-9401(P)/WCAP-9402(NP)," May 15, 1981.

11. Letter from C. O. Thomas (NRC) to E. P. Rahe, Jr., (Westinghouse) "Acceptance for Referencing of Licensing Topical Report - WCAP-8762(P)/WCAP-8783(NP), Supplement 1, Basis for the Applicability of the WRB-1 Correlation to 15x15 OFA and 14x14 OFA Fuel," June 29, 1984.

Distribution Copies:
Docket No(s)
NRC PDR
Local PDR
TNovak, Actg. DD
NThompson, DHFT
ELD
EJordan
BGrimes
JPartlow
GLear
DDiIanni
PShuttleworth
ACRS (10)
RLobel
HY Hsiung
PAD#1 r/f
PAD#1 p/f