June 7, 1984

Docket No. 50-255
LS05-84-06-013

Mr. David J. VandeWalle
Nuclear Licensing Administrator
Consumers Power Company
1945 W. Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO THE BASIS FOR THE THERMAL MARGIN/LOW PRESSURE TRIP SETTING

Re: Palisades Plant

The Commission has issued the enclosed Amendment No. 82 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment is in response to your application dated September 29, 1983.

This amendment modifies the basis for the thermal margin/low pressure trip setting based on the results of a reanalysis of the control rod withdrawal transient and also modifies the basis for the limit on linear heat rate.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on November 22, 1983 (48 FR 52811). No request for hearing was received and no comments were received.

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice Publication in the Federal Register.

Sincerely,
Original signed by
Walter A. Paulson, Project Manager
Operating Reactors Branch #5
Division of Licensing

SE01 04
DSU USE

Enclosures:
1. Amendment No. 82 to License No. DPR
2. Safety Evaluation

cc w/enclosures:
See next page

ADD: J.N. GRACE (1)
L. HARMON (2)
L. SCHNEIDER (1)
T. BARNHART (4)
E. JORDAN (1)
J. COLLINS (4 cys) 314-

DISTRIBUTION

Docket	NRC PDR	Local PDR
ORB #5 Reading	NSIC	DCrutchfield
LA	OELD	ELJordan
JNGrace	ACRS (10)	SEPB
RDiggs (fee form)	LSchneider	TBarnhart (4)
LHarmon (2)		

DL:ORB #5	DL:ORB #5	DL:ORB #5	OELD	DL:AD/SA
MShuttleworth:jc	WPaulson	DCrutchfield		FMiraglia
6/5/84	6/5/84	6/5/84	/ /84	6/7/84

8406130094 840607
PDR ADOCK 05000255
P PDR

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

June 7, 1984

Docket No. 50-255
LS05-84-06-013

Mr. David J. VandeWalle
Nuclear Licensing Administrator
Consumers Power Company
1945 W. Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO THE BASIS FOR THE THERMAL MARGIN/LOW PRESSURE TRIP SETTING

Re: Palisades Plant

The Commission has issued the enclosed Amendment No. 82 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment is in response to your application dated September 29, 1983.

This amendment modifies the basis for the thermal margin/low pressure trip setting based on the results of a reanalysis of the control rod withdrawal transient and also modifies the basis for the limit on linear heat rate.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on November 22, 1983 (48 FR 52811). No request for hearing was received and no comments were received.

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice Publication in the Federal Register.

Sincerely,

Walter A. Paulson

Walter A. Paulson, Project Manager
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 82 to License No. DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

cc
M. I. Miller, Esquire
Isham, Lincoln & Beale
Suite 4200
One First National Plaza
Chicago, Illinois 60670

Mr. Paul A. Perry, Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Judd L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

James G. Keppler, Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Township Supervisor
Covert Township
Route 1, Box 10
Van Buren County, Michigan 49043

Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Palisades Plant
ATTN: Mr. Robert Montross
Plant Manager
Covert, Michigan 49043

U.S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

Resident Inspector
c/o U.S. NRC
Palisades Plant
Route 2, P. O. Box 155
Covert, Michigan 49043

Lee E. Jager, P.E., Chief
Environmental and Occupational
Health Services Administration
Michigan Department of Public Health
3500 N. Logan Street
Post Office Box 30035
Lansing, Michigan 48909

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

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 82
License No. DPR-20

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consumers Power Company (the licensee) dated September 29, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Provisional Operating License No. DPR-20 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, (Environmental Protection Plan) as revised through Amendment No. 82, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter A. Paulson

for Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 7, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 82

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Technical Specifications by removing the following pages and by inserting the enclosed pages. The revised pages contain the captioned amendment number and marginal lines indicating the area of change.

Remove Pages	Insert Pages
2-8	2-8
2-10	2-10
3-104	3-104
3-106	3-106

Basis (Cont'd)

(TM/LP) trip will occur before these limits are reached. Reference 13 forms the basis for Figure 2-3 for 4-pump operation. For 2- and 3-pump operation the flow instability criterion is more limiting than the MDNBR criterion. Reference 7 forms the basis for Figures 2-1 and 2-2.

The trip is initiated whenever the pressurizer pressure drops below the minimum value given on Table 2.3.1, or a value computed as described below, whichever is higher. The computed value is a function of reactor inlet temperature and reactor outlet temperature, and takes the form $P_{Trip} = AT_H - BT_C - C$ where A, B and C are constants and T_H and T_C are the hot and cold leg coolant temperatures, respectively. The minimum value of reactor coolant flow and the maximum expected values of axial and radial peaking factors are assumed in generating this trip function.

The TM/LP trip set points are derived from the 4-pump operation core thermal limits (Figure 2-3) through application of appropriate allowances for measurement uncertainties and processing errors. A maximum error of 165 psi is assumed to account for expected instrument drift and repeatability errors, process measurement uncertainties, flow stratification effects, and calibration errors. As such, a maximum error in the calculated set point of -165 psi has been assumed in the accident analysis.[12]

An analysis has been performed[14] which verifies that the TM/LP trip for 4-pump operation provides adequate thermal margin when RTD time delays and conservative assumptions regarding part power radial peaking factors are compensated for by an improved pressurizer model, a primary coolant flow update and the XNB DNB correlation for control rod withdrawal transients. The XNB DNB correlation has been shown to be applicable to the Palisades Plant in Reference 15.

For two- and three-pump coolant pump operation, power is limited to 21% and 39% of rated power, respectively, for a maximum of 12 hours. During either of these modes of operation, the high power level trip in conjunction with the TM/LP trip (minimum set point = 1750 psia) and the secondary system safety valves (set at 1000 psia) assure that the limits shown on Figures 2-1 and 2-2 will not be violated.

5. Low Steam Generator Water Level - The low steam generator water level reactor trip protects against the loss of feed-water flow accidents and assures that the design pressure of the primary coolant system will not be exceeded. The specified set point assures that there will be sufficient water inventory in the steam generator at the time of trip to provide a 15-minute margin before the auxiliary feedwater is required.[9]

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Cont'd)

References (Cont'd)

(8) XN-NF-77-18, Section 3.8.

(9) XN-NF-77-18, Section 3.7.

(10) FSAR, Amendment No 17, Item 4.0.

(11) XN-NF-77-18, Section 3.6.

(12) XN-NF-77-18, Section 3.1.

(13) XN-NF-77-22, Section 3.4.

(14) XN-NF-83-57.

(15) XN-NF-709.

3.23.1 LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

ACTION 3:

If the incore alarm system is inoperable and the excore monitoring system is not being used, operation at less than or equal to 85% of rated power may continue provided that incore readings are recorded manually. Readings shall be taken on a minimum of 10 individual detectors per quadrant (to include 50% of the total number of detectors in a 10-hour period) within 4 hours and at least every 2 hours thereafter. If readings indicate a local power level equal to or greater than the alarm setpoints, the action specified in ACTION 1 above shall be taken.

Basis

The limitation on LHR ensures that, in the event of a LOCA, the peak temperature of the cladding will not exceed 2200°F.(1) In addition, the limitation on LHR for the highest power fuel rod, narrow water gap fuel rod and interior fuel rod ensures that the minimum DNBR will be maintained above 1.30 for the W-3 correlation or above 1.17 for the XNB correlation during anticipated transients; and, that fuel damage during Condition IV events such as locked rotor will not exceed acceptable limits.(2)(3)(5) The inclusion of the axial power distribution term ensures that the operating power distribution is enveloped by the design power distributions.

Either of the two core power distribution monitoring systems (the incore alarm system or the excore monitoring system) provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its limits. The incore alarm system performs this

3.23.1 LINEAR HEAT RATES (LHR

LIMITING CONDITIONS OF OPERATION

Basis (Cont'd)

uncertainty factor of 1.03, a thermal power measurement uncertainty factor of 1.02 and allowance for quadrant tilt.

References

(1) XN-NF-77-24

(2) XN-NF-77-18

(3) XN-NF-78-16

(4) XN-NF-80-47

(5) XN-NF-83-57

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 82 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

1.0 INTRODUCTION

By letter dated September 29, 1983, Consumers Power Company (the licensee) proposed changes to the basis for the thermal margin/low pressure trip setting in the Palisades Technical Specifications by including the acceptance criterion and results of a reanalysis of the control rod withdrawal transient that takes into account the response time of the temperature detectors providing input to these safety system instruments. This change would also be reflected in the basis for the limit on linear heat rate.

At the request of the NRC staff, by letter dated November 1, 1983, Consumers Power Company submitted the Exxon Nuclear Company Report, XN-NF-709, "Justification of XNB Correlation for Palisades," May 1983, which was referenced in the September 29, 1983 application. By letter dated May 11, 1984, the licensee provided additional clarifying information in response to the NRC staff's request for additional information dated May 3, 1984.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on November 22,, 1983 (48 FR 52811). A request for hearing and public comments were not received.

2.0 DISCUSSION

In a Licensee Event Report (LER) 83-20 submitted on April 5, 1983, Consumers Power Company reported they had discovered that the safety analyses contained in XN-NF-77-18, "Plant Transient Analyses of the Palisades Reactor for Operation at 2530 MWt" did not account for the response times of the resistance temperature detectors (RTDs) in the primary coolant system. The hot and cold leg temperature RTD measurements are used in the thermal margin/low pressure (TM/LP) trip function for termination of rod withdrawal transients. The proposed change in the basis of technical specification 2.3 reflects the reanalysis of rod withdrawal transient for the Palisades reactor as described in XN-NF-83-57 "Rod Withdrawal Transient Reanalysis for the Palisades Reactor."

The reanalysis used the UFEB82 version of the PTSPWR2 code (Ref. 1) including a pressurizer model which calculates the pressure increase in the pressurizer during rod withdrawal transients. The effect of the RTD response time on the

TM/LP trip was included and the coolant flow update and the ENC XNB correlation to calculate DNB were used. The radial peaking values were held constant instead of decreasing as the reactor power increased during the rod withdrawal. The current Palisades Technical Specfication part power peaking limits were used for the analysis.

The following control rod withdrawal transients were analyzed:

(1) transients initiated from 102% of rated power at reactivity addition rates bounding the possible range.

(2) transients initiated from 52% of rated power at reactivity addition rates bounding the possible range.

Beginning of cycle, (BOC), mid-cycle (MC) and end of cycle (EOC) kinetics parameters were used in the analysis.

3.0 EVALUATION

Analysis of the rod withdrawal transient for the Palisades Reactor using the previously used methodology and RTD delay times produced results that did not meet DNB limits. The reanalysis was undertaken using the UFEB82 version of the PTSPWR 2 Code (Ref. 1). This Code and methodology is under staff review which has progressed sufficiently that we have reasonable assurance that the results for rod withdrawal events will not be significantly altered by completion of our review. Enclosure 1 to this safety evaluation is the NRC staff's evaluation of the ENC XNB correlation for Palisades (Ref. 2). The staff finds that the XNB correlation is acceptable for application to Palisades with a minimum DNBR limit of 1.17. Therefore, the staff concludes that it is acceptable to use the UFEB82 version of the PTSPWR 2 code and the XNB correlation for the Palisades reanalysis.

The previous analysis had allowed radial peaking values to increase as reactor power increased. For the reanalysis, the more conservative approach, holding of the radial peaking values constant was used. Since the RTD time constant was given as 7±2 seconds, the values used in the analysis were 9 seconds for the hot leg and 5 seconds for the cold leg in order to be conservative.

Three sets of kinetic parameters were used.

(1) beginning of cycle (BOC) minimum feedback
(2) end of cycle (EOC) maximum feedback
(3) mid-cycle

Except for the EOC doppler coefficient, parameters used for BOC and EOC are identical to those previously used and are expected to bound values for future cycles. The hot zero power EOC doppler coefficient was made more negative by 20% in order to ensure bounding the feedback effects. An analysis using mid-cycle parameters had not been performed previously.

The conservative value of 1.5%$\Delta\rho$ for maximum rod worth withdrawn was used for the BOC and EOC cases. This corresponds to the combined rod worth of banks 3 and 4, while the power dependent insertion limits (PDILs) for banks 3 and 4 allow only 20% and 80% insertion at 50% power. For the MC cases, the maximum rod worth withdrawn was 1%$\Delta\rho$ which is more reactivity than the PDILs allow to be inserted.

The minimum DNBR condition for all transients considered occurred for the mid-cycle case initiated from 52% power for a reactivity addition rate of less than $3\times10^{-5}\Delta\rho$/second.

The minimum DNBR was 1.40 compared to the XNB DNBR limit of 1.17. All BOC, EOC and MC cases initiated from 52% rated power with high or intermediate reactivity addition rates terminated on the over power neutron flux trip on high pressure trip. The BOC kinetics and low reactivity addition rate transients also trip on the high pressure trip. For the EOC kinetics and low reactivity addition rate transient, no reactor trip occurred. For the MC and low reactivity addition rate transients, the transients trip on the thermal margin/low pressure trip (TM/LP).

The rod withdrawal transients from 102% power was also analysed for BOC, EOC and MC kinetics. No limits were placed on bank worths. The range of reactivity addition rates was 1.0×10^{-5} to $3\times10^{-4}\Delta\rho$. The transients were all terminated by the high pressure trip, the TM/LP trip or the nuclear flux trip in all cases. The lowest value of MDNBR was greater than 1.7.

4.0 CONCLUSIONS

The reanalysis of the rod withdrawal transients indicates that the use of the new PTSPWR 2 model, the XNB correlation and the coolant flow update more than compensate for the DNBR reducing effects of the RTD response time and the more conservative assumptions regarding part power radial peaking factors which were used in this analysis.

Based on our review of the report XN-NF-83-57 "Rod Withdrawal Transient Reanalysis for the Palisades Reactor," we agree with the licensee's conclusion that no fuel rod in the Palisades core will experience DNB during an uncontrolled rod bank withdrawal transient. Therefore, we find the proposed change in the basis of technical specification 2.3 to be acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

The staff has determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, the staff has further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 ACKNOWLEDGEMENT

M. Chatterton prepared this evaluation. Y. Hsii prepared the evaluation for the "Justification of XNB Correlation for Palisades," (Enclosure 1).

Dated: June 7, 1984

8.0 REFERENCES

1. Description of the "Exxon Nuclear Plant Transient Simulation Mode for Pressurized Water Reactors," (PTS-PWR), XN-75-5, Revision 2, August 1983.

2. Macduff, R. B., "Justification of XNB Correlation for Palisades," XN-NF-709, Rev. 0, May 12, 1983.

(Enclosure 1)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

JUSTIFICATION OF XNB CORRELATION FOR PALISADES

1.0 INTRODUCTION AND BACKGROUND

The Exxon Nuclear Company's (ENC) XNB critical heat flux (CHF) correlation as described in XN-NF-621(P) Revision 1 (Ref. 1), has been reviewed previously by the staff (Ref. 2) with technical assistance from Idaho Nuclear Engineering Laboratory. As a result, the staff had concluded that XNB with a minimum departure from nucleate boiling ratio (DNBR) of 1.17 is acceptable for licensing calculations when it is used with the XCOBRA-IIIC code and is applied within its applicability range.

The CHF test data provided in XN-NF-621(P), Revision 1, as a basis for the development of XNB, consist of test sections representative of fuel assemblies designed by various fuel vendors such as Exxon Nuclear, Westinghouse and Combustion Engineering. Since this data base does not include explicitly a rod bundle prototypical of the Palisades fuel design, the application of XNB to the Palisades reload could be outside of the XNB applicability range. By letter dated November 1, 1983 (Ref. 3), Consumers Power Company submitted an ENC report, XN-NF-709, "Justification of XNB Correlation for Palisades" to justify the use of XNB for Palisades fuel. In XN-NF-709, an additional CHF test section, ENC-204, is provided. This test section is representative of the Palisades fuel design. The staff evaluation of the applicability of XNB to the Palisades fuel is addressed in the following section.

2.0 EVALUATION

The CHF test data presented in XN-NF-621(P), Revision 1, consist of test sections representative of various fuel designs and a variety of axial power distributions expected for power operation. The ENC-204 test data presented in XN-NF-709 for justification of the application of XNB to Palisades fuel consist of uniform axial power shape data only. Therefore, rather than using the ENC-204 data alone to derive the DNBR limit, it must be shown that the ENC-204 test data belongs to the same population of the data which were used in the XNB development. In addition, it must be shown that the DNBR limit of 1.17 is correct or conservative limit relative to the ENC-204 data base.

The CHF data reduction for the ENC-204 test section is performed using XCOBRA-IIIC thermal-hydraulic code for the determination of subchannel fluid conditions. The same computer code and method were used previously in the data reduction for XN-NF-621(P), Revision 1, and had been found acceptable. Therefore, this safety evaluation will be concentrated on the statistical analysis of the CHF data.

In the treatment of CHF test data, the statistical method used by ENC was to evaluate the predicted-to-measured (P/M) ratios of CHF data. This is a deviation from general practice of using the measured-to-predicted (M/P) CHF ratios. However, the previous staff review (Ref. 2) had determined that this statistical characterization of the CHF data is acceptable. The same method is used in the treatment of the ENC-204 data and is also acceptable.

In the staff evaluation of XN-NF-621(P), Revision 1, a One-Way analysis of variance was performed by the staff technical consultant at INEL on the ungrouped test data to test the equality of means of the P/M ratios. The results of this analysis revealed three separate populations among the test data presented in XN-NF-621(P), Revision 1. The DNBR limit of 1.17 was the highest limit obtained among the three populations and could be a conservative limit when compared to the one-sided tolerance limit derived from a particular population. For example, the one-way analysis of variance had determined that the test sections ENC-3, ENC-4, ROSAL-4 and WH-164 were of the same population. The DNBR limit derived from this population is 1.15 with a 95 percent probability at 95 percent confidence level of avoiding DNB. These test sections consist of axial power distributions ranging from uniform, cosine U and U sine U shapes, and rod diameters ranging from 0.374 to 0.422 inches.

In response (Ref. 4) to a staff question on whether the ENC-204 data belongs to the population of data presented in XN-NF-621(P), Revision 1, the licensee performed a one-way analysis of variance for the test sections ENC-3, ENC-4, ROSAL-4 and WH-164, which had been determined previously to be of the same population, and the test section ENC-204. The result showed a F-statistic of 2.50 for all 5 test sections compared to an F-statistic of 3.02 for the 4 test sections including ENC-204. Therefore, it is reasonable to assume that the ENC-204 test section also belongs to the same population of the other four sections. The staff has also performed an independent calculation by combining the means and standard deviations, respectively, of the four test sections to form a group mean and standard deviation of the combined data. An F-test and a t-test are performed to determine the equality of variances and equality of means between the ENC-204 data and the combined data from the four test sections. The results show an F-statistic of 1.024 and a t-statistic of 0.964. Therefore, both null hypotheses of equal variances and equal means can not be rejected at a 5 percent significance level. We, therefore, conclude that ENC-204 test section belongs to the same population and the ENC-204 data can be incorporated with the other four sections. The new combined data mean of P/M ratios and standard deviation are 0.95646 and 0.10188, respectively, for a total 257 data points. The one-sided tolerance DNBR limit derived from these data would be 1.141 with a 95 percent probability at 95 percent confidence level of not experiencing DNB. The 95/95 DNBR limit derived from the ENC-204 data alone is 1.169. Therefore, use of DNBR limit of 1.17 is conservative.

3.0 CONCLUSION

The NRC staff has reviewed XN-NF-709. Based on this review, the staff concludes that the XNB correlation is acceptable for application to the Palisades fuel with minimum DNBR limit of 1.17. This acceptability is subject to other restrictions imposed in the staff safety evaluation report (Ref. 2, copy attached) on XN-NF-621(P), Revision 1.

REFERENCES

1. R. B. Macduff, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," XN-NF-621(P), Rev. 1, Exxon Nuclear Company, April 1982.

2. Letter from C. O. Thomas (NRC) to Dr. Richard B. Stout (Exxon Nuclear Company), "Acceptance for Referencing of Licensing Topical Report XN-NF-621(P), Revision 1, Exxon Nuclear DNB Correlation for PWR Fuel Designs," April 12, 1983

3. Letter from B. D. Johnson (Consumer Power) to D. M. Crutchfield (NRC), "Docket No. 50-255 - License DPR-20 - Palisades Plant - XN-NF-709, 'Justification of XNB Correlation for Palisades,' May 1983," November 1, 1983.

4. Letter from B. D. Johnson (Consumers Power Company) to D. M. Crutchfield, May 11, 1984.

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

APR 12 1983

Dr. Richard B. Stout, Manager
Exxon Nuclear Company
2101 Horn Rapids Road
P. O. Box 130
Richland, Washington 99352

Dear Dr. Stout:

Subject: Acceptance for Referencing of Licensing Topical Report XN-NF-621(P), Revision 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs"

We have completed our review of the subject topical report submitted May 5, 1982 by Exxon Nuclear Company (ENC) letter GFO:034:82. We find this report is acceptable for referencing in license applications for LWR Plants to the extent specified and under the limitations delineated in the report and the associated (NRC) evaluation which is enclosed. The evaluation defines the basis for acceptance of the report.

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

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

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

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
As stated

1 INTRODUCTION

In XN-NF-621, Revision 1, Exxon Nuclear Company (ENC) presented the XNB critical heat flux (CHF) correlation which will be used to assess the thermal margin of pressurized water reactors (PWRs). The XNB is an empirical relationship which specifies CHF (i.e., the heat flux at which departure from nucleate boiling, DNB, occurs) as a function of local coolant conditions and fuel assembly geometry. It is based on 14 test series with a total of 714 data points and three different PWR fuel vendor designs. The 14 test series include variations in grid design, heated length, grid span, rod diameter, and axial and radial power distributions.

The local coolant conditions in the rod bundle were calculated using the XCOBRA-IIIC computer code which is described in XN-NF-75-21(P) and the range of coolant conditions tested were typical of an operating PWR.

Based on the XNB's ability to predict the test data, Exxon has proposed a departure from nucleate boiling ratio (DNBR) limit of 1.17 for the correlation. This limit corresponds to a 95% probability of not experiencing DNB at a 95% confidence level. The comparable value for the W-3 correlation, which is presently used by ENC, is 1.30.

8304200423
PDR 23pp

2 DESCRIPTION OF CORRELATION

The basic form of the XNB correlation is as follows:

$$q''_{uncorrected} = A + B * HLOC \qquad \text{eq. (1)}$$

where $A = f$ (pressure, mass velocity, inlet subcooling)

$B = f$ (pressure, mass velocity, local enthalpy)

HLOC = Reduced local enthalpy

= Local Enthalpy/906.00

All of the parameters used in the XNB are reduced using the critical properties of water (i.e., the water properties at the critical pressure, 3208.2 psi) and using the above method for HLOC.

Additional factors are used as part of the correlation to account for non-uniform axial power distributions, geometry differences such as spacer pitch and mixing vane loss coefficients, and differences in heated lengths. The final form of the XNB is:

$$q''\ critical = (q''_{uncorrected})^{*}\ Correction\ Factors \qquad \text{eq. (2)}$$

The procedure for using the XNB is to initially calculate the heat flux using equation (1), determine the appropriate correction factors, calculate CHF using equation (2), and determine the DNBR, which is the ratio of the actual heat flux to predicted CHF.

The ranges over which Exxon is requesting the XNB be applied (Chandler; January 6, 1983) are:

Pressure (psia)	1395 - 2425
Local Mass Velocity (Mlbm/hr-ft^2)	0.92 - 3.04
Local Enthalpy (Btu/lb)	594.85 - 821.24
Local Quality	-02 - +0.3
Heated Length (inches)	66 - 168
Grid Spacing (inches)	14.3 - 22.0
Inlet Subcooling (Btu/lb)	37.2 - 336.34

It will also be used for the following geometries:

Vendors:	Exxon Nuclear Combustion Engineering Westinghouse
Fuel Design:	Non-Mixing Vane Mixing Vane
Equivalent Hydraulic Diameter (inches)	0.177 - 0.612
Equivalent Heated Diameter (inches)	0.463 - 0.528

The test series and their associated fuel rod arrays are:

Vendor	Rod Array	Test Series
Westinghouse/ Exxon	14x14, 15x15	ENC-3, 4, and 5 ROSAL-2, 4, 7, and 8
Exxon	17x17	ENC-6
Cumbustion Engineering	16x16	CE-47, CE-59
Westinghouse	17x17	WH-162 and 164

3 STAFF EVALUATION

3.1 Scope of Review

The staff review of XN-NF-621, Revision 1 included an independent audit of the subchannel calculations performed to determine the local coolant conditions in the rod bundle for all 714 data points. This was performed using the COBRA-IV computer code which was derived from and is an ancillary of the COBRA-IIIC program. Our review also included a statistical analysis of the calculated results and a review of the methodology used in combining the XCOBRA-IIIC code and the correlation. During the review, requests were made for data clarification and additional or corrected information was received in several areas.

The above reviews were performed by the Idaho National Engineering Laboratory (INEL) under the direction of a cognizant staff member.

3.2 Results of Audit Calculations

The results of the INEL audit calculations are presented in Tables 1 and 2. Table 1 is a comparison of the local conditions at which CHF was predicted as determined by the XCOBRA-IIIC and COBRA-IV codes for a limited number of data points. The comparison indicates good agreement between the two codes and either could be used to establish the local conditions required for the development of a CHF correlation.

Table 2 is a comparison of the mean and standard deviation for each of the data sets and the total population. This comparison shows good agreement for the overall values but contains discrepancies in many of the individual data sets. The possible ramifications associated with these differences are described in the statistical analysis discussion contained in this report.

During our review, the staff requested that Exxon provide a description of how the local conditions for the XNB were determined including a discussion of the subchannel code used, subchannel modeling, axial nodalization, and input assumptions. Exxon responded that the XCOBRA-IIIC code was used to calculate the local coolant conditions. XCOBRA-IIIC is a derivative of the COBRA-IIIC code which was developed at Battelle Pacific Northwest Laboratory. The modifications made by Exxon to COBRA-IIIC include minor improvements in the solution technique, the addition of calculational options, and operational modifications such as streamlining code input.

Exxon further stated that the friction factors used were determined from pressure drop measurements performed on ENC test sections or estimated for geometries for which ENC does not have detailed test data. These loss coefficient estimates are based on the experience gained from measuring actual fuel bundles of Westinghouse or Combustion Engineering (C-E) designs. They also reported that sensitivity studies of CHF test data showed negligible influence on predicted conditions when the form loss coefficients were varied by as much as 15%.

The mixing values ($\frac{Bs}{D}$) chosen were based on spacer design and are dependent on a particular fuel type. These values were determined experimentally for the ENC designed fuel while for non-Exxon fuel a lower bounding value was used for mixing vane grids. For example, in analyzing, the Westinghouse "L" grid design a lower value of 0.010, which was obtained from WCAP-8030-A, was used.

Based on our review of the above information, the staff concludes that the approach taken by Exxon in determining the local conditions used in developing the XNB correlation are acceptable. The XCOBRA-IIIC code is still under staff review, and any limitations resulting from this review will be addressed in our safety evaluation report on XN-NF-75-21(P), Revision 2.

The INEL audit calculations were performed using the same friction factor correlation, two-phase flow correlation, crossflow resistance, momentum turbulent mixing factor, pitch to length parameter, inlet enthalpy and inlet mass velocity as Exxon.

Our review also included an analysis of the correction factors used in the XNB development and the determination of these factors in actual reactor application. Based on this review, we have concluded that the method used to calculate these parameters and their values used in determining the DNBR limit are acceptable.

However, it is the opinion of the staff and our consultant that a change in these parameters, such as determining their values using a prototype and then a full scale bundle, may increase the uncertainty in both the code's prediction of local coolant conditions and the correlations prediction of CHF. This may significantly alter the statistical analyses on which the DNBR limit is based. Therefore, we conclude that the values of these parameters used in the development of the XNB must be used in licensing analyses.

For the uniform heat flux tests, ENC used the end of the heated length as the CHF location while the experiments showed that for the same tests, CHF occurred upstream of the end of the heated length. When asked to justify using this technique in determining the DNBR Exxon responded that the worst local conditions calculated for a bundle having a uniform axial power distribution (APD) are at the end of the heated length. In order to maintain a consistent path between test analysis and reactor design and based on the fact that the DNBR location in a reactor is determined by the code and is not known apriori, the procedures used to determine the DNBR for those tests where burnout occurred upstream of the heated length is acceptable. We have reviewed the additional information provided by ENC and have concluded that the method used by Exxon in determining DNBR is acceptable since the DNBR limit is dependent on the ability of the subchannel code to predict local conditions which produce CHF.

An additional area of concern raised by the staff on the uniform heat flux tests was why CHF occurred at the thermocouple upstream of the end of the heated length rather than at the end of the heated length where the highest quality region should occur. Exxon stated that burnout is a function of the location of the spacer grid and that the grids will improve heat transfer for a distance of 20 or more rod diameters downstream of the spacer. Because the spacer was located slightly downstream of the end of the heated length, heat transfer above the spacer would improve while the local hydraulic conditions downstream of the grid would be more severe. Therefore, for the experimental data in question, the effects of the spacer grid dominated the occurrence of CHF even though a higher quality may occur at the end of the test bundle. The staff has reviewed this information and concludes that ENC has acceptably addressed our concerns on this issue.

Finally in the area of test procedures, the staff requested that Exxon provide a discussion on how the rate of power was increased, what post-test inspections were performed, and what, if any, duplicate runs were made to establish continued integrity of the test bundle. In response to this concern, ENC stated that the power was manually raised in the CHF tests by an increment of less than 1% and held constant until conditions became stable. This process was repeated until CHF occurred. They further stated that duplicate runs were made to establish continued integrity. As an example, they cited the ENC-6 tests, where replicate points were taken during the test and one in between point was taken at the end of the test to confirm continuity and consistency of the test data from beginning to end. At the end of the tests, post-test inspections were performed and, for example, on the ENC-6 bundle there were no visible signs of hot spots on the rods. Based on our review of this information, the staff has concluded that the CHF tests were performed in an acceptable manner.

Our review of the statistical characterization of the XNB results dealt mainly with the method used by Exxon to statistically analyze the data and a review of the analyses. The statistical method used by ENC was to evaluate the predicted-to-measured (P/M) ratio of CHF data. Since in previously approved correlations, the measured-to-predicted (M/P) ratio was used to determine the 95/95 limit, Exxon was asked to justify their technique. ENC responded that the procedure used in determining the 95/95 limit assumed a normal distribution. Transforming the data from P/M to M/P yields two distributions for comparison, both of which may be normal or both may depart from normality. As a verification on the 95/95 limit for the P/M data, Exxon performed a distribution free estimate of the limit and determined the value to be 1.177. For the reverse ratio, and using their original statistical approach, Exxon calculated that 95/95 limit for the M/P data, when a normal distribution is assumed, is 1.191.

ENC further stated that the non-parametric estimate of the 95/95 limit, 1.177, does not make complete use of the actual distribution, and therefore this limit will bound the 95/95 limit obtained from the actual distribution. By considering the first four moments of the P/M data ENC found that the actual distribution is a gamma distribution. On the other hand, the use of the M/P data is overly conservative since, the actual value of the 95/95 limit for the P/M data, when the appropriate distribution is used, lies at some value below the non-parametric limit of 1.177. ENC also stated that the DNBR reported for licensing analyses is defined as P/M ratio. Based on our review of the above information, the staff has concluded that the analysis of the P/M data is acceptable.

As part of the review, the staff requested that Exxon demonstrate that each of the samples, e.g., test series, belong to a single population. ENC responded by initially performing a Bartlett test for homogeneity of variance (Chandler; August 26, 1982). The breakdown was based on both vendor design and fuel assembly geometries. The results of this test showed that the variances do differ among geometry types.

Exxon also performed a K-sample Squared Ranks test of variance using the above groupings (Chandler; August 26, 1982). Results for the population of 6 samples and 5 degrees of freedom indicated that at least two of the variances were unequal. By removing the ROSAL, ENC-1, and 2 data, Exxon found that there exists a significance level between 2.5% and 5.0% that the remaining data were from the same population. Finally, ENC removed the ENC-3, 4, and 5 data and analyzed the remaining population. Based on the results of the third analysis, Exxon concluded that the data comprised of 3 samples and 2 degrees of freedom were likely identical.

An analysis of the means and a comparison of variance analysis showed that for an equivalent sample size of 83.7 with 378.7 degrees of freedom the mean is 0.98502 with a standard deviation of 0.09847. Based on this mean and standard deviation the 95/95 DNBR limit would be 1.168.

The final analysis performed by ENC was the determination of a DNBR limit excluding that data which had the greatest possibility of being from a different population. For all sections less the ROSAL and ENC 1 thru 5 data the DNBR limit was 1.169 while for all sections less the ENC-6, WH-162, WH-164, CE-47, and 49 data, the DNBR limit was 1.176.

The results of the above tests lead ENC to conclude that the data could be treated as a single population and that the 1.17 DNBR limit would cover any deviation within the data sets.

In order to ascertain the validity of these conclusions, INEL performed a series of F-tests to identify any systematic variation among the test series. The tests were performed at a 99% confidence level. Based on the F-test, INEL concluded that there was a variance among tests of different geometries. Additionally, INEL performed a one-way analysis of variance using the ungrouped test series.

For the one-way analysis, INEL used the groupings reported by ENC and calculated a F-ratio of 24.03 for six samples with five and 708 degrees of freedom for the numerator and denominator. This result shows that there is a variance among the tests when they are grouped by geometry type. Removing data sets WH-162, WH-164, ENC-3, 4, and 5 resulted in an F-ratio of 2.40 with three and 392 degrees of freedom for the numerator and denominator. This indicates that the remaining data have a probability of between 5% and 10% of being in the same population.

A second one-way analysis of variance was performed on the ungrouped data. The results of this test are presented in Table 3 and indicate that ENC-1, ENC-2, ENC-6, ROSAL-2, ROSAL-7, ROSAL-8, WH-162, CE-47 and CE-49 are probably of the same population while test series ENC-3, ENC-4, ROSAL-4, and WH-164 are of a second population. ENC-5 is a unique test series and does not fall into either population. Using the above populations, a DNBR limit of 1.21 for the ENC-1, ENC-2, etc. population was determined while the ENC-3, ENC-4, etc. population has a 95/95 limit of 1.133.

Figure 1 is a histogram of the total data set and it shows that the overall population is approximately normally distributed. Histograms for the individual samples (EGG-NTAP-6167) show that ENC-3, ENC-4, ENC-5, ROSAL-4 and WH-164 are skewed to the left of the population mean.

Further analyses were performed to determine if there was a reason for the groupings obtained from the one-way analysis of variance. A number of groupings were examined using different bases such as rod diameter, grid spacing, radial power distribution, axial power distribution, KLOSS, and an unheated guide tube in the bundle. These studies showed no uniqueness in either grouping.

A second evaluation revealed that the modeling of the guide tube was an influence in determining the above grouping. For those bundles containing an unheated guide tube, CHF experimentally occurred in a channel that contained the guide tube; however, in predicting CHF, Exxon often reported burnout in a channel other than the one with the guide tube. Since the guide tube is an unheated wall, CHF occurs at less severe local conditions and has a lower value. If CHF is predicted in a typical channel, four heated rods, when it actually occurred in a guide tube channel, this would be nonconservative. The reason for this is that the predicted local conditions are greater than the conditions which experimentally produced CHF; therefore, the analytical results show that you can go to a higher power than you actually achieved.

Table 4 presents a summary of the test series that have one or more unheated guide tubes. For all of the series reported in Table 4 ENC predicted CHF in the COBRA hot channel rather than the experimental channels listed in the table. This indicates that the reason ENC-3, ENC-4, and ENC-5 do not belong to the population may be the difference in the channel for the predicted and measured CHF. Test series ENC-6 does not fall from the population because the difference between the COBRA-IV experimental hot channel and the guide tube channel is only 3.0% and the sample mean is closer to the expected mean of 1.0.

In addition to the above analyses, the INEL audit calculations revealed that the ENC-1, 2, 3, 4, 6, CE-59, and ROSAL-8 test series were biased with inlet pressure. For pressures less than 1800 psia the correlation predictions tend to be scattered about some value less than 1.0 while for data above 1800 psia the data is randomly scattered about 1.0. This indicated that the correlation under predicts CHF for the lower pressures but is reasonably accurate for pressures above 1800 psia. Based on this review, the staff has concluded that although these test series statistically belong to one of the two populations, excluding the ENC-5 population, the fact that they are biased with pressure may preclude them from being placed in either population.

Also, the staff statistically analyzed the six different geometry types reported by Exxon. Table 5 contains the results of our analysis based on a geometric characterization. These results show that for the ENC-1 and -2 population the mean, standard deviation, and 95/95 limit are much greater than the mean, standard deviation, and 95/95 limit of the remaining populations when they are compared to the same parameters of the total population.

Based on our review of the ENC statistical analyses, our consultant's analyses, and the result of the staff's statistical analyses, we requested additional information from Exxon which justified treating the 14 samples as one population.

In response to our concerns, Exxon provided plots of DNBR versus inlet pressure for those test series that the staff felt were biased with pressure (Chandler; December 16, 1982). Based on their own pressure plots ENC concluded that there was no significant systematic trends with pressure. We have reviewed the information submitted in the December 16, 1982 letter and have concluded that there is a small trend with pressure; however, the trend is random in nature and does not exhibit any systematic characteristics. Therefore, the staff concludes that the ENC-1, 2, 3, 4, 6, CE-59, and ROSAL-8 test series need not be treated as a single population due to the trends in pressure, since these trends are not systematic.

With respect to the statistical analyses, Exxon requested that the data be reviewed as two separate populations (Chandler; December 22, 1982). One of the populations would be comprised of the test series representing 16x16 and 17x17 arrays (CE-47, CE-59, WH-164, WH-162, and ENC-6) while the second population would represent the 15x15 bundles. As justification for requesting this breakup, ENC provided the range of test conditions and axial power distributions found in each population.

A review of the 16x16 and 17x17 data base showed that only a chopped cosine and uniform axial power distribution (APD) were present. It is the position of the staff that all possible power distributions expected throughout an operating cycle be used in the development of any CHF correlation. Since the 16x16 and 17x17 do not include either an upskew or downskew APD, Exxon cannot remove those test series, e.g. the 15x15 array, that have the upskew APDs. Therefore, the 15x15 test series must remain in the data base until ENC provides additional data for the 16x16 and 17x17 test series which contain an upskew and/or downskew APD.

In a modified response (Chandler; January 3, 1983) Exxon requested that test series ENC-1 and ENC-2 be removed from the data base. The reason for eliminating this data was that ENC-1 contained minimum grids that were not representative of any grid being manufactured by ENC, Westinghouse or CE while ENC-2 had a uniform axial and radial power distribution that was atypical of actual reactor conditions. ENC further stated that a statistical analysis of the data was performed using the populations reported by INEL. The results of these evaluations showed that the worst 95/95 limit was 1.17 for the population containing the CE-47, -59, WH-162, ENC-2, ROSAL-2, -7, and -8 test series. Based on these results, we have concluded that the proposed grouping of data which results in a DNBR limit value of 1.17 is acceptable.

4 CONCLUSION

The staff has reviewed XN-NF-621, Revision 1 and the additional supporting information submitted by Exxon Nuclear Company. Based on this review, we have concluded that XNB correlation is acceptable for use in reactor licensing applications. We have also concluded that the 95/95 DNBR limit of 1.17 reported by Exxon is acceptable. These conclusions are based on the following:

(1) The subchannel code used, XCOBRA-IIIC, is acceptable for predicting local coolant conditions used in the development of a CHF correlation. This is based on a comparison of XCOBRA-IIIC with the staff's audit code COBRA-IV. Since the XCOBRA-IIIC is still under staff review, any limitations resulting from its use will be addressed in our safety evaluation report on the code.

(2) An independent audit, performed by our consultant INEL, using a different subchannel code yielded similar results.

(3) The DBNR data has been statistically characterized in an acceptable manner.

(4) The 95/95 limit is based on three separate populations that were recommended by our consultant; therefore, the 95/95 limit of one population will be conservative when compared to the limit of a population containing all of the test data.

We will require that the correction factors used in analyzing the CHF test data and the mixing factors used in the data reduction be used in reactor design applications, since a change in these factors may alter the code and correlation uncertainties associated with the prediction of CHF. This in turn may raise or lower the 95/95 DNBR limit. Therefore, if any of these parameters are changed, ENC must provide a description of the change and sufficient justification which warrants making this change. Additionally, Exxon should provide the test data which justifies using the XNB on fuel designs not contained in the data base or acceptable justification on why the XNB is applicable to this fuel type. For example, Exxon manufactured fuel for CE reactors is not present in the data base. ENC must provide additional test data for these fuel bundles or a quantified justification of the XNB's applicability to this bundle type.

Finally, it should be noted that the DNBR limit does not include any adjustment which is required when a mixed core, e.g. a core with geometrically different fuel types, is analyzed.

5 REGULATORY POSITION

The staff concludes that the XNB CHF correlation as described in XN-NF-621, Revision 1 is acceptable for use in licensing application when it is used with the XCOBRA-IIIC code and within the range of application reported in Section 2.2 of this safety evaluation report. We also conclude that the 95/95 limit of 1.17 associated with the XNB is acceptable. Use of the correlation should be within the limitations described in the previous section.

Based on our review, the staff finds XN-NF-621, Revision 1 an acceptable and referential report with the restrictions noted in the above paragraph.

Table 1: Comparison of Local Conditions

Case	Enthalpy		Quality		Void Fraction		Mass Flux	
	XCOBRA-III	COBRA-IV	XCOBRA-III	COBRA-IV	XCOBRA-IIIC	COBRA-IV	XCOBRA-IIIC	COBRA-IV
ENC-3-63	656.57	656.49	0.077	0.077	0.610	0.594	1.9046	1.9434
ENC-4-28	703.28	705.78	0.167	0.167	0.709	0.712	1.4897	1.5210
ENC-6-42	616.44	628.00	0.00	0.007	0.318	0.350	2.8655	2.8998
ROSAL-2-18	612.57	627.50	0.001	0.027	0.550	0.554	1.8674	1.8809
ROSAL-2-9	622.44	636.52	0.018	0.043	0.561	0.566	1.9409	1.9601

Table 2: Comparison of Mean and Standard Deviation

Test Section	Number of Data Points	Mean (Meas./Pred) XCOBRA-IIIC	COBRA-IV	Standard Deviation XCOBRA-III	COBRA-IV
CE-47	96	1.028	1.0300	0.0741	0.0804
CE-59	89	1.023	1.0500	0.0820	0.1020
WH-164	53	0.950	0.9727	0.0677	0.0682
WH-162	53	0.992	1.0032	0.0845	0.0736
ROSAL-2	28	0.976	0.9995	0.118	0.0990
ROSAL-4	26	0.933	0.9689	0.0843	0.0832
ROSAL-7	11	0.970	1.0383	0.1043	0.1210
ROSAL-8	32	1.001	1.0586	0.0987	0.1070
ENC-1	28	1.040	1.0504	0.1212	0.1220
ENC-2	24	0.993	1.0119	0.1093	0.1090
ENC-3	73	0.994	0.9458	0.1029	0.0923
ENC-4	80	0.985	0.9712	0.1196	0.112
ENC-5	59	0.911	0.8956	0.0848	0.0811
ENC-6	62	0.995	1.0071	0.0749	0.0868
Total Population	714	0.985	0.99614	0.09847	0.1030

Table 3: One Way Analysis of Variance

Test Series Grouping	F-Ratio	Probability of Being in Same Population
ENC-1, -2, -6 ROSAL-2, -7, -8 WH-162, CE-47, -59	2.47	1 - 2.5%
ENC-1, -2, -4, -6 ROSAL -2, -4, -7, -8 WH-162, -164 CE-47, CE-59	5.57	---
ENC-1, -2, -3, -4, -6 ROSAL-2, -4, -7, -8 WH-162, -164 CE-47, -59	7.84	---
ENC-3, -4, -5 ROSAL-4, WH-164	7.39	---
ENC-3, -4 ROSAL-4, WH-164	1.23	>10%

Table 4: Comparison of Test Series With Unheated Guide Tubes

Test Series	Number of Experimental CHF Predictions: COBRA-IV Hot Channel[1]	COBRA-IV Channel Other Than Hot Channel	Explanation
WH-162	All	-0-	As expected.
ENC-6	20	42	The 42 channels are 3% cooler than the hot channel.
ENC-3	18	53	Five of the indications occur in a channel with 5% less power, 21 in a channel with 0.4% less power and the remaining in a channel with 23% less power.
ENC-4	30	50	Seven of the 50 indications were in a channel with 0.20% less power while the remaining 43 were in a channel with 22% less power.
ENC-5	4	53	Twenty-five of the 53 indications occur in a channel with 0.9% less power while the remaining 28 are in a channel with 22% less power.
CE-47	82	14	The 14 indications occur in a channel with 0.3% less power.
CE-59	85	4	The 4 indications occur in a channel with 0.1% less power.

[1]ENC predicts all CHFs in this channel.

Table 5: Comparison of 95/95 Limit Based on Geometry

Geometry Grouping	Mean	Standard Deviation	95/95 Limit
CE-47, CE-59	1.0256	0.0778	1.169
WH-162, WH-164	0.9710	0.0791	1.123
ENC-6	0.995	0.0749	1.146
ROSAL-2, 4, 7, 8	0.9720	0.1021	1.169
ENC-1, ENC-2	1.0183	0.1173	1.259
ENC-3, ENC-4, ENC-5	0.9503	0.0865	1.109
Total Population	0.985	0.0985	1.163

6 REFERENCES

6.1 Topical Reports

XN-NF-621, Revision 1 "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, April 1982.

XN-NF-75-21(P), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Exxon Nuclear Company, September 1982.

WCAP-8030-A, "Application of Modified Spacer Factor to L Grid Typical and Cold Cell DNB," Westinghouse Electric Corporation, January 1975.

6.2 Other References

Ambrosek, R. G., C-C Tsai, and R. D. Wadkins, "Exxon Nuclear DNB Correlation Review," EGG-NTAP-6167, EG&G Idaho, INC., February 1983.

J.C. Chandler (ENC) to H. Bernard (NRC), Subject: "XN-NF-621(P), 'Exxon Nuclear DNB Correlation for PWR Fuel Designs,' Revision 1," August 26, 1982.

J.C. Chandler (ENC) to J.J. Holonich (NRC), Subject: "XN-NF-621, Revision 1, 'Exxon Nuclear Company DNB Correlation for PWR Fuel Designs,' April 1982," December 9, 1982.

J.C. Chandler (ENC) to L.E. Phillips (NRC), Subject: "XN-NF-621, Revision 1, 'Exxon Nuclear Company DNB Correlation for PWR Fuel Designs,' April 1982," December 16, 1982.

J.C. Chandler (ENC) to L.E. Phillips (NRC), Subject: "XN-NF-621, Revision 1, 'Exxon Nuclear Company DNB Correlation for PWR Fuel Designs,' April 1982," December 23, 1982.

J.C. Chandler (ENC) to L.E. Phillips (NRC), Subject: "XN-NF-621, Revision 1, 'Exxon Nuclear Company DNB Correlation for PWR Fuel Designs,' April 1982," January 3, 1983.

J.C. Chandler (ENC) to L.E. Phillips (NRC), Subject: "XN-NF-621(P), Revision 1, 'Exxon Nuclear DNB Correlation for PWR Fuel Designs,' April 1982," January 6, 1983.