OPPD

Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

February 12, 1992
LIC-92-020R

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

References: 1. Docket No. 50-285
2. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated November 27,1991 (LIC-91-320A)
3. Letter from NRC (D. L. Wigginton) to OPPD (W. G. Gates) dated December 26, 1991

Gentlemen:

SUBJECT: Additional Information Concerning Fort Calhoun Station Cycle 14 Reload

Attached are the Omaha Public Power District (OPPD) responses to the eight questions contained in Reference 2. Also included is the response to an additional question raised at a meeting between OPPD and the NRC on January 13, 1992.

The responses provide the NRC additional information on OPPD's submittal for the proposed Technical Specification change on the negative limit for the Moderator Temperature Coefficient (MTC) for Operating Cycle 14.

In response to Question 3 of the attachment, OPPD has referenced the report CE-CES-129, Revision 1-P. Twenty-three (23) copies of this report (copy numbers 29-53) are attached for your information. Pursuant to 10 CFR 2.790, ABB-CE (Combustion Engineering) has determined that CE-CES-129, Revision 1-P, contains proprietary information to be withheld from public disclosure. Appropriate documentation is attached justifying this determination.

180003

Apol 1/23

If you should have any questions, please contact me.

Sincerely

J K Gasper

W. G. Gates for
Division Manager
Nuclear Operations

WGG/sel

Attachments

c: LeBoeuf, Lamb, Leiby & MacRae, (w/o Attachments)
D. L. Wigginton, NRC Project Manager, (with Attachments)
S. D. Bloom, NRC Project Engineer, (with Attachments)
R. D. Martin, NRC Regional Administrator, Region IV, (w/o Attachments)
R. P. Mullikin, NRC Senior Resident Inspector, (w/o Attachments)

Additional Information Concerning Fort Calhoun Station Cycle 14 Reload Application

1. **Explain why the beginning-of-cycle, hot zero power (HZP) steam line break accident is specified in Section 5.1.1 as the most limiting in determining required shutdown margin. From Table 5-1, both the moderator temperature coefficient and the doppler coefficient are most negative at end-of-cycle (EOC) and, therefore, the EOC event would appear to result in a larger reactivity insertion with cooldown.**

   The larger reactivity insertion rate at EOC would provide more limiting consequences for the accident and forms the basis for the required shutdown margin. However, the BOC conditions provide the minimum available scram worth in assessing the margin to the Technical Specification shutdown margin limit (which is currently 4.0 $\%\Delta\rho$). This is shown in the review of the scram worths available at BOC and EOC. The BOC scram worth is 5.0596 $\%\Delta\rho$ while the EOC scram worth is 5.9833 $\%\Delta\rho$.

   The response to Question 2 provides additional discussion of shutdown margin and scram worth.

2. **Provide a table of CEA reactivity worths and allowances similar to Table 5-2 for Cycle 14 for EOC HZP conditions.**

   The following table is similar to Table 5-2. It includes the requested EOC, HZP limiting values of reactivity worths and allowances. The Table also illustrates the difference between BOC, HZP and EOC, HZP limiting CEA shutdown worths for all events, including the main steam line break accident. From the table below, use of the BOC, HZP CEA worth value as the most limiting value is appropriate since the BOC, HZP excess shutdown margin is 0.92 $\%\Delta\rho$ less than the EOC, HZP excess shutdown margin.

FORT CALHOUN UNIT NO. 1, CYCLE 14
LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES
FOR HOT ZERO POWER

	BOC, HZP (%Δρ)	EOC, HZP (%Δρ)
1. Worth of all CEAs Inserted	7.52	8.86
2. Stuck CEA Allowance	1.17	1.43
3. Worth of all CEAs Less Worth of Most Reactive CEA Stuck Out	6.35	7.43
4. Power Dependent Insertion Limit CEA Worth	1.19	1.33
5. Calculated Scram Worth	5.16	6.10
6. Physics Uncertainty plus Bias	0.10*	0.12*
7. Net Available Scram Worth	5.06	5.98
8. Technical Specification Shutdown Margin	4.00	4.00
9. Margin in Excess of Technical Specification Shutdown Margin	**1.06**	**1.98**

* 1.96 % of calculated scram worth using Nodal Expansion Method (see Question 3, Reference 2).

Due to the applicability of BOC, HZP conditions for not only the main steam line break accident but for all other events in determining the Cycle 14 minimum excess shutdown margin, a clarification is required in the last paragraph of Section 5.1.1, page 21. The revised paragraph should read as follows:

> BOC, HZP conditions for all events are the most limiting conditions used in the determination of available shutdown margin for compliance with the Technical Specifications. The minimum available shutdown margin is 1.06 %Δρ with respect to the Technical Specification limit of 4.0 %Δρ. Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for Cycle 14. The Cycle 14 CEA worth values, used in the calculation of minimum scram worth, exceed the minimum value required by Technical Specifications and thus provide an adequate shutdown margin.

3. **Justify the reduction in the physics uncertainty and bias of the calculated scram worth to 1.96 percent as shown in Table 5-2 and reference all appropriate reports.**

Use of upgraded reactor physics codes necessitates the use of uncertainties and biases consistent with the application of the new methods. Scram worths calculated using the new methods, with biases and uncertainties applied, provide comparable results to those obtained using the former methods (with biases and uncertainties included). The NRC-approved reference cycle (i.e., Cycle 13) reload submittal utilized the Higher Order Difference (HOD) method which is described in Reference 1. For the Cycle 14 submittal, the Nodal Expansion Method (NEM), which is also described in Reference 1, was implemented which increases calculational accuracy of the nuclear design codes. Specific changes incorporated into the new methods include:

1. Implementation of NEM into the ROCS code;
2. Improvements in accountability of anisotropic scattering and higher order interface current angular distributions in the DIT code;
3. Introduction of assembly discontinuity factors between the ROCS and DIT codes;
4. Update of biases and uncertainties applied to calculated parameters.

The revised biases and uncertainties associated with the application of NEM are described in Reference 2. Introduction of the improved methods required the re-evaluation of the biases and uncertainties. The Combustion Engineering data base used to establish the biases and uncertainties was expanded to reflect recent reload cycles with low leakage and high burnup fuel management. The data base was derived from the following sources:

Plant	Cycle	Number of Banks
Palo Verde 1	2	7
Palo Verde 2	2	7
Palo Verde 3	2	7
Palo Verde 1	3	7
Palo Verde 2	3	7
Calvert Cliffs 1	10	5
Calvert Cliffs 2	8	5
Fort Calhoun	13	6

Total Cycles: 8
Total Banks: 51

In addition, Calvert Cliffs 2, Cycle 9 data was added later and found to be consistent with the above data base.

For calculating the FCS Cycle 14 (N-1) scram worths, the uncertainty plus bias terms used are found in Reference 2, Table C (Item C-2), page C-1. The use of NEM in ROCS underpredicted scram worth by 4.32%, thus, calculated values must be increased by 4.32%. The uncertainty term for ROCS-NEM scram worths is 6.28% which is applied in the conservative direction. Since the bias term and the uncertainty term, when taken individually, are applied in different directions, the resultant bias plus uncertainty term is 1.96%. Using the former ROCS-HOD method, the scram worths were overpredicted by 4%, thus the calculated value must be reduced by 4% to obtain the biased scram worth. The uncertainty term for the ROCS-HOD method is 9%. Since both terms must be applied in the same direction, the combined bias plus uncertainty term is 13%. Therefore, in order for both NEM and HOD to produce similar net scram worths, the NEM combination of bias and uncertainty terms must be smaller in value than the HOD combination of bias and uncertainty terms.

To verify that use of the HOD method or the NEM method produces similar results, a Cycle 13 scram worth model using ROCS-NEM was generated and compared to the NRC-approved Cycle 13 ROCS-HOD results. These results are presented below along with results from Cycle 14 using ROCS-NEM:

FORT CALHOUN UNIT NO. 1, CYCLE 14
LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCE FOR
BOC, HZP (%$\Delta\rho$)

	Cycle 13 (HOD)	Cycle 13 (NEM)	Cycle 14 (NEM)
1. Worth of all CEAs Inserted	9.23	7.93	7.52
2. Stuck CEA Allowance	1.83	1.47	1.17
3. Worth of all CEAs Less Worth of Most Reactive CEA Stuck Out	7.40	6.46	6.35
4. Power Dependent Insertion Limit CEA Worth	1.23	1.10	1.19
5. Calculated Scram Worth	6.17	5.36	5.16
6. Physics Bias plus Uncertainty	0.80*	0.11**	0.10**
7. Net Available Scram Worth	**5.37**	**5.25**	5.06
8. Technical Specification Shutdown Margin	4.00	4.00	4.00
9. Margin in Excess of Technical Specification Shutdown Margin	**1.37**	**1.25**	1.06

* 13% of calculated scram worth using Higher Order Difference (HOD) method.

** 1.96% of calculated scram worth using Nodal Expansion Method (NEM).

The results show the difference between the HOD and NEM methods for calculating the minimum Cycle 13 scram worth to be 0.12 %$\Delta\rho$, which is considered acceptably small. It can also be determined from the above results that ROCS-NEM produced a more conservative net available scram worth than ROCS-HOD.

In summary, the application of the revised physics uncertainty and bias value documented in Table 5-2 of the reload application is based upon the use of ROCS-NEM methods rather than the ROCS-HOD method, and both methods are described in Reference 1. OPPD's application of the ROCS-NEM method in Cycle 14 was consistent wit the same method used in the derivation of the biases an uncertainties of Reference 2.

References:

1. "The ROCS & DIT Computer Codes for Nuclear Design", CENPD-266-P-A, April 1983.

2. "Physics Biases and Uncertainties", CE-CES-129, Revision 1-P, August 1991.

4. Explain why there is no change in the maximum radial peaking factor or in the maximum ejected CEA worth between BOC and EOC conditions (Table 5-3).

The most limiting radial peaking factor was calculated (including uncertainties and biases) for the BOC and EOC conditions. The peaking factors were then raised to a more bounding value and the largest value was transmitted to Westinghouse. This value was then applied in a conservative manner to conditions during a cycle to ensure that the existing and future operating cycles would be bounded by the Westinghouse CEA ejection analysis.

5. Discuss in more detail the justification for using the CE fuel rod bow penalty for both the Westinghouse and CE fuel coresident in the c[illegible]s.

The design basis for the amount of fuel rod bow allowed in the Westinghouse fuel and for the CE fuel design is the same. Westinghouse has identified in the fuel mechanical design report that the amount of deflection does not require a DNB penalty to be applied under Westinghouse analysis requirements. Thus, the CE DNB penalty was applied to the Westinghouse fuel to ensure that the OPPD statistical combination of uncertainties was still valid and that conservative input assumptions were used in the analysis.

6. Section 6.1 implies that the steady-state DNBR analysis for Cycle 14 differs from that used in previous cycles because of the use of the TORC code rather than the CETOP-D code. However, this does not appear to differ from the methodology specified in the previous version (Rev. 3) of OPPD-NA-8301. Please clarify this point and explain any DNBR methodology difference from the previous cycle in more detail.

The DNBR analysis applications and methods did not change from previous cycles, with the exception that the TORC computer code was used to calculate the minimum DNBR rather than the CETOP-D computer code. Both codes are approved for use with the OPPD methods. The CETOP-D code was developed to run much faster on the mainframe computer system than TORC. With the large number of computer runs required during the reload process, there was a considerable savings in using CETOP-D in the DNBR analyses.

Since these calculations are now done on engineering workstations by OPPD, no cost savings are realized in the use of CETOP-D for DNBR work. Any thermal margin lost in the tuning process for CETOP-D can now be recovered by running TORC for DNBR calculations rather than the CETOP-D. The OPPD topical report was revised to show the use of TORC rather than CETOP-D for calculating the MDNBR. All of the other aspects for calculating MDNBR remain the same as in previous versions of the methodology topical report.

7. **Explain why the critical boron concentration with all rods out assumed in the Cycle 14 boron dilution event during refueling was 1180 ppm whereas the TS minimum refueling boron concentration is 1900 ppm. Why have the critical boron concentration values for the other modes decreased significantly from the previous cycle values and why have the inverse boron worths remained the same?**

The 1180 ppm value listed was used to determine the minimum boron concentration which in accordance with the Fort Calhoun Station Technical Specifications must include at least a 5 %$\Delta\rho$ shutdown margin. In addition the 30 minute dilution to critical time criterion must be met. The current TS value was compared to the 1180 ppm value adjusted by 5.0 %$\Delta\rho$ and was found to be conservative. The TS value is adjusted as necessary for each cycle to ensure that the time to criticality meets the minimum requirements for operator action. For Cycle 14 no adjustment was required and the margin noted in the above question exists.

The use of the integral fuel burnable absorber (IFBA) fuel design caused the large reduction in the critical boron concentration requirements. Since this is the first Westinghouse fuel to be loaded into Fort Calhoun Station it is anticipated that there will continue to be changes in the boron requirements for future cycles as more of the fuel displacing shims are replaced by IFBA rods.

The inverse boron worths appear to remain the same since a bounding value is used to provide a limiting analysis for each cycle. The table below compares the actual and analysis values of the inverse boron worth for Cycle 14.

FORT CALHOUN UNIT NO. 1, CYCLE 14
INVERSE BORON WORTHS
ACTUAL VERSUS ANALYSIS VALUES

OPERATING MODE	ACTUAL VALUE (ppm/%Δρ)	ANALYSIS VALUE (ppm/%Δρ)
Hot Standby (2)	-95.5	-90
Hot Shutdown (3)	-95.5	-55
Cold Shutdown (4-normal volume)	-70.7	-55
Cold Shutdown (4-minimum volume)	-70.4	-55
Refueling (5)	-79.7	-55

8. **Since Table 5-2 specifies 5.06 percent as the net available scram worth at HZP, why was 6.40 percent used in the HZP CEA withdrawal analysis?**

   The incorrect value was left in the Table from a previous draft. The correct value, from the analysis document, is 5.048 %Δρ for the CEA Withdrawal Analysis at HZP and should replace the 6.40 %Δρ value questioned. The 5.06 %Δρ in Table 5-2 is an input value for the Main Steam Line Break Analysis at HZP conditions.

9. **In accordance with Appendix A of Standard Review Plan 4.2, the NRC requires an evaluation of fuel assembly structural integrity considering the lateral effects of seismic and LOCA loads for transition cores consisting of different fuel types using time history numerical techniques based on the plant specific safe shutdown earthquake (SSE). Verify that this has been performed for Fort Calhoun Cycle 14 and that the results show that all fuel types are structurally acceptable for the transition core. The results should show that the grids will not buckle due to combined impact forces of a seismic/LOCA event, the core coolable geometry is maintained, and the stresses resulting from the seismic/LOCA induced deflections are within acceptable limits.**

In the discussions on January 13, 1992 the NRC noted that for Cycle 8 there was a requirement in the SER (Reference 1) that "the licensee will be required to provide analytical results to the NRC within one year using approved ENC methodology to comply with fuel assembly structural acceptance criteria in Appendix A of SRP-4.2 for the design seismic event." It should be noted that OPPD provided a response to the seismic analysis requirement in Reference 2.

The OPPD response stated:

> "The District has reviewed the seismic analysis contained in Appendix F of the USAR to determine the type of analysis performed for the first core. This type of analysis would be performed on a core containing CE and ENC or ENC fuel, since it is the licensing basis for the core seismic analysis.
>
> Based on the information contained in Appendix F, the District concludes that a dynamic seismic analysis was not performed for the fuel assemblies. It is the District's position that an analysis to show compliance with the fuel assembly structural criteria in Appendix A of SRP-4.2 for the design seismic event is outside the scope of the design basis for the Fort Calhoun Station Unit No. 1 and that an unreviewed safety question does not exist for a core of CE and ENC fuel or ENC fuel with respect to the design seismic event. Therefore, it is the District's position that such an analysis is not required."

Subsequent correspondence,Reference 3, indicated:

> "You further stated that such an analysis is not required. We agree."

Thus, the seismic analysis was not required for a mixed core of CE and ENC fuel.

As described in the fuel mechanical design report, there was a substantial effort made to ensure compatibility between the CE and Westinghouse fuel assembly design parameters. The elevation of the grids in the CE and Westinghouse fuel assemblies are not matched on centerlines, but there is overlap between the adjacent grids. The fuel design also required that the grid crush strength be comparable between CE and Westinghouse fuel assemblies. The crush strength was based on the LOCA blowdown loads and manufacturing tests by the fuel vendors. A peripheral grid was found to have some deformation in the previous CE LOCA load analysis which required a coolable geometry study to be done. In the Westinghouse LOCA analysis, two grids were assumed to fail as a conservative evaluation practice and the impact on coolable geometry and peak clad temperature was also assessed.

Both the CE and Westinghouse LOCA evaluations indicated that a coolable geometry was maintained based on NRC approved acceptance criteria. The assembly and grid stresses were acceptable and the grids of one manufacturer will not crush the grids of the other due to impact loads from a LOCA.

References:

1. Letter, E. G. Tourigny (NRC) to W. C. Jones (OPPD) dated March 15, 1983, Facility License Amendment #70.
2. Letter(LIC-83-184)from W. C. Jones (OPPD) to Mr. Robert A.Clark (NRC) dated July 28, 1983.
3. Letter, James R. Miller (NRC) to Mr. W. C. Jones (OPPD) dated August 29, 1983

7/29/83 - WEM, GPD, MUG, KJM, RLJ, WGG, TJM, JCB, [redacted] DDW, PMS, FAT, FFF, TRR, RRM, FCFile

Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

July 28, 1983
LIC-83-184

QUESTION #9
REFERENCE 2

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

Response to Concerns Contained in the
Cycle 8 Reload SER and Cycle 9 Reload Schedule

The Cycle 8 reload SER requested the District address concerns related to fuel assembly seismic analysis, worst case ECCS assumptions, high burnup fuel, and documentation of the District's reload methodology. This letter addresses these concerns and also provides a schedule for submittals related to the Cycle 9 reload.

Fuel Assembly Seismic Analysis

The District has reviewed the seismic analysis contained in Appendix F of the USAR to determine the type of analysis performed for the first core. This type of analysis would be performed on a core containing CE and ENC or ENC fuel, since it is the licensing basis for core seismic analysis.

Based on the information contained in Appendix F, the District concludes that a dynamic seismic analysis was not performed for the fuel assemblies. It is the District's position that an analysis to show that compliance with the fuel assembly structural criteria in Appendix A of SRP-4.2 for the design seismic event is outside the scope of design basis for the Fort Calhoun Station Unit No. 1 and that an unreviewed safety question does not exist for a core of CE and ENC fuel or ENC fuel with respect to the design seismic event. Therefore, it is the District's position that such an analysis is not required.

Worst Case ECCS Assumptions

The SER required that the assumption of a worst single failure in the ECCS be reviewed and verified as more limiting than an assumption of no single failure. In response to the District's request, Exxon Nuclear Company has performed a sensitivity study of the Fort Calhoun Cycle 8 ECCS analysis to determine which assumption provided the most restrictive results.

The worst single failure assumption was the loss of a low pressure safety injection pump. When the ECCS analysis was redone, due to the Fort Calhoun ECCS configuration, the estimated full safety injection flow resulted in a higher reflood rate with no significant effect on containment pressure. Assuming the single failure, the analysis predicted lower reflood rates and a higher peak cladding temperature. Therefore, the Fort Calhoun Cycle 8 ECCS analysis provides the most limiting prediction with the assumption of a single failure.

High Burnup Fuel

The Cycle 8 reload SER stated that batch average burnups exceeding 38,000 MWD/MTU in future cycles would involve an unreviewed safety question related to radiological consequences. Based on a May 16, 1983 telephone conversation between the Commission and District staffs, it is our understanding that these high burnup concerns will be addressed in the extended burnup topicals submitted by the nuclear fuel vendors. In response to your request made during the May 16, 1983 telephone conversation, the anticipated discharge batch burnups for future cycles are provided in Table 1.

TABLE 1
FORT CALHOUN STATION UNIT NO. 1
ANTICIPATED BATCH DISCHARGE BURNUP
(All fuel manufactured by ENC)

Cycle Discharged	Anticipated Shutdown Date	Cycle Loaded	No. of Assemblies	Batch Average Discharge Burnup (MWD/MTU)
9	Sept 1985	6	24	34,000
9	Sept 1985	7	19	35,500
10	Mar 1987	7	17	40,000
10	Mar 1987	8	23	36,000
11	Sept 1988	8	5	39,000
11	Sept 1988	9	8	44,000
11	Sept 1988	9	31	39,500

Documentation of Reload Methodology

The Cycle 8 reload SER requested the District submit methodology reports well in advance of the Cycle 9 reload application date. The scope of these methodology reports was discussed in a May 11, 1983 telephone conversation between members of the Commission and District staffs. Based on requests made by members of the Commission staff during these conversations, the District will submit methodology reports on reactor physics and transient analyses. In addition, the District intends to submit a reload methodology report which will provide an overview of the analyses performed during a reload core analysis and the interfaces between these analyses.

The District also intends to utilize a Statistical Combination of Uncertainties (SCU) in the Cycle 9 reload analysis. The SCU analysis is being performed by Combustion Engineering utilizing the methodology previously submitted and approved on the Calvert Cliffs and St. Lucie 1 dockets. The schedule for submittals and events related to Cycle 9 reload licensing is given in Table 2.

TABLE 2
CYCLE 9 RELOAD SCHEDULE

Event	Date
Submit Reactor Physics and Transient Analysis Methodology Reports	September 23, 1983
Submit Statistical Combination of Uncertainties Report	October 21, 1983
Submit Cycle 9 Technical Specifications	February 10, 1984
Start of Refueling	March 19, 1984
Cycle 9 Startup	May 14, 1984

The District believes this letter addresses all Commission staff concerns discussed in the Cycle 8 reload SER. The submittal of the methodology reports will satisfy all requirements discussed in the SER.

Sincerely,

W. C. Jones
Division Manager
Production Operations

WCJ/JKG:jmm

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

Mr. L. A. Yandell, Senior Resident Inspector

9/7/83 - WEM, GPB, WCJ, MOG, KJM, [redacted], WGG, TJM, JCB, JKG, DDW, PMS, FAT, FFF, DJM, TRR, RRM, FCFile

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

AUG 29 1983

JKG -
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Docket No. 50-285

QUESTION #9
REFERENCE 3

Mr. W. C. Jones
Division Manager, Production
Operations
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Jones:

We have reviewed your letter of July 28, 1983 in which you provided responses to our long range concerns contained in our Cycle 8 reload SER which was issued on March 15, 1983.

We required that if the District intends to use a safety analysis computer code to support reload licensing actions, it should demonstrate its proficiency in using the code by submitting code verification. This would best be accomplished by submitting methodology reports for our review and approval. You have provided schedules for submittal of such reports and they are acceptable.

We required the District to provide analytical results to the NRC within one year using the approved ENC methodology to comply with fuel assembly structural acceptance criteria in Appendix A of SRP-4.2 for the design seismic event. You have stated that an analysis to show that compliance with the fuel assembly structural criteria in Appendix A of SRP-4.2 for the design seismic event is outside the scope of design basis for the Fort Calhoun Station and that an unreviewed safety question does not exist for a core of CE and ENC fuel or ENC fuel with respect to the design seismic event. You further stated that such an analysis is not required. We agree.

We noted that our Cycle 8 approval applies to the requested discharge average exposure of 37,200 MWD/MTU only and that significant increases in this or future cycles would involve safety questions related to radiological consequences. You stated that it is your understanding that these high burnup concerns will be addressed in the extended burnup topicals submitted by the nuclear fuel vendors. We have no objection to addressing radiological consequences for high burnup fuel in vendor topicals as long as they apply to Fort Calhoun fuel and are addressed adequately and documented.

We required that for the large break LOCA, you must demonstrate the worst assumption for ECCS operation since it was shown that for some plants using Exxon fuel, maximum safety injection might be the worst case rather than loss of some ECCS capacity as was believed previously. You stated that the Fort Calhoun Cycle 8 ECCS analysis provides the most limiting prediction with the assumption of a single failure. This is acceptable.

In summary, your July 28, 1983 letter adequately addressed our long range concerns that were discussed in our Cycle 8 safety evaluation of March 15, 1983.

Sincerely,

Charles M Trammell

for James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

cc: See next page

Omaha Public Power District

cc:

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Mr. Jack Jensen
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Regional Administrator
Nuclear Regulatory Commission, Region IV
Office of Executive Director for Operations
611 Ryan Plaza Drive Suite 1000
Arlington, Texas 76011

<u>AFFIDAVIT PURSUANT</u>

<u>TO 10 CFR 2.790</u>

Combustion Engineering, Inc.)
State of Connecticut)
County of Hartford) SS.:

I, S. A. Toelle, depose and say that I am the Manager, Nuclear Licensing, of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations and in conjunction with Omaha Public Power District for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

CE-CES-129 Rev. 1-P, "Methodology Manual - Physics Biases and Uncertainties," 1991.

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790

of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced documents, should be withheld.

1. The information sought to be withheld from public disclosure, which is owned and has been held in confidence by Combustion Engineering, is physics biases and uncertainties applied to PWR nuclear design parameters.

2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to Combustion Engineering.

3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F. M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject document herein is proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because:
   a. A similar product is manufactured and sold by major pressurized water reactor competitors of Combustion Engineering.
   b. Development of this information by C-E required thousands of manhours and hundreds of thousands of dollars. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.
   c. In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of similar physics biases and uncertainties that are applied to PWR nuclear design parameters.

d. The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e. The information consists of physics biases and uncertainties applied to PWR nuclear design parameters, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of Combustion Engineering's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

g. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

S. A. Toelle

S. A. Toelle
Manager
Nuclear Licensing

Sworn to before me
this 7th day of February, 1992

Laurie J. White
Notary Public

My commission expires: 3/31/94