CEN-257(O)-NP SUPPLEMENT 1-NP

STATISTICAL COMBINATION OF UNCERTAINTIES

PART 1

AUGUST, 1985

POWER SYSTEMS
COMBUSTION ENGINEERING, INC.

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ABSTRACT

This supplement updates Part 1 of the original report (CEN-257(0)-P, November, 1983) to account for the changes made to the Fort Calhoun Station Unit 1 TM/LP trip. Because this trip has been modified to include explicit monitoring of the shape index, the descriptions of the protection system and the analysis of the net uncertainties and the reported value of the net uncertainty have been changed.

The numbered sections of this report supersede the same numbered sections of the original report. Only Sections 2.3, 3.1, 3.2, Figure 2-1, and Tables 3-1, 3-2 and B1-lof the original report are changed.

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SUPPLEMENT 1 TO

STATISTICAL COMBINATION OF

UNCERTAINTIES

PART I

(CEN-257(0)-P)

REPORT SUMMARY

Analysis of the DNB LSSS uncertainty factor accounting for the addition of the ASI processing to the TM/LP trip follows almost the same procedure as used prior to the modification. That procedure was described in CEN-257(O)-P, Part 1. In this new procedure, a change has been made to the process for evaluating the allowance for TM/LP processing uncertainty. The addition of ASI monitoring to the TM/LP trip system allows [ ] and results in a new processing allowance of [ ] (2 standard deviations) rather than the [ ] allowance without the direct ASI processing. The net effect of these changes is that the uncertainty factor applied to the revised TM/LP system is [ ].

2.3 TM/LP LSSS STOCHASTIC SIMULATION

For the TM/LP LSSS, DNB overpower (P_{fdn}) is the dependent variable of interest. Core coolant inlet temperature, reactor coolant system pressure, peripheral axial shape index, and total core power are monitored directly by the TM/LP trip system. Total integrated radial peaking factor and RCS coolant flow rate are monitored by other systems and must be included in the TM/LP LSSS evaluations.

Figure 2-1 is a flow chart representing the simulation sequence for the TM/LP LSSS. For each simulation trial, a value of overpower obtained using sampled values of uncertainties about nominal conditions is calculated. This value is compared to the overpower calculated at nominal conditions by taking the ratio of the two values. This simulation sequence is repeated over several thousand sets of nominal operating conditions covering the operations space for the plant. The resulting distribution of the ratio of nominal overpower to overpower incorporating uncertainties is used to determine the overall uncertainty factor on the TM/LP LSSS.

The operating space for the analysis is defined by establishing ranges for RCS pressure, flow, radial peaking ($F_R{}^T$), axial shape index (ASI) and core coolant inlet temperature. The ranges for $F_R{}^T$ and ASI are set to range of flow rate is set to cover flows from a minimum to a maximum value which might occur with the present plant configuration. The pressure range is bounded by the value of the high pressurizer pressure trip setpoint and the lower pressure limit of the TM/LP trip system. Core coolant inlet temperature covers a range established by the combination of core power and inlet temperature resulting in the highest temperature at which the secondary safety valves opean and the lowest temperature at which the low secondary pressure trip occurs.

The ranges of conditions used in the analysis are listed in Table 2-1.

Note: A line is drawn in the right hand margin where a change has been made.

3.1 RESULTS OF ANALYSES

The analytical methods presented in Section 2 have been used to show that a stochastic simulation of uncertainties associated with the APD LSSS and the TM/LP LSSS results in combined uncertainties of [ ]% and [ ]%, respectively, at a 95/95 probability/confidence limit.

Table 3-1 shows the values of the individual uncertainties which were statistically combined to yield the above combined uncertainties. Appendix B contains a further discussion of the bases for these individual uncertainties.

The combined uncertainties are in units of percent overpower (P_{fdl} and P_{fdn}) and are applied in the generation of the APD and TM/LP LSSS as discussed below (Reference 3-1).

3.2 IMPACT ON MARGIN TO SAFDL

The motivation for using a statistical combination of uncertainties is to improve NSSS performance through a reduction in the analytical conservatism in the margin to the SAFDL. This section contains a discussion of the margin obtainable through a reduction in this conservatism.

Table 3-2 lists the uncertainty values previously used on this plant. The approximate worth of each of these uncertainties in terms of percent overpower margin (P_{fdl}, P_{fdn}) is also shown.

The total uncertainties previously applied to the APD LSSS and the TM/LP LSSS are approximately [ ] and [ ], respectively. The uncertainties resulting from the application of the statistical combination of uncertainties program are approximately [ ] and [ ]. The use of the statistical combination of uncertainties provides a reduction in conservatism in the margin to SAFDL of approximately [ ] and [ ], respectively.

Although the conservatism in the margin to SAFDL has been reduced, a high degree of assurance remains that the SAFDL will not be violated.

OMAHA PUBLIC POWER DISTRICT FORT CALHOUN Nuclear Power Plant	THERMAL MARGIN UNCERTAINTY ANALYSIS	Figure 2-1

TABLE 3-1

UNCERTAINTIES ASSOCIATED WITH THE APD LSSS
AND THE TM/LP LSSS

Uncertainty*	APD LSSS	DNB LSSS
Core power (% of rated power)	±2%	±2%
Primary coolant mass flow (% design)**	NA	
Primary coolant pressure (psid)	NA	
Core coolant inlet temperature (F^{o})	NA	
Power distribution (peaking factor)		
Separability	See Table 1 of Appendix B1	
Calibration (asiu)		
Shape Annealing (asiu)		
Monitoring system processing (asiu)		
Monitoring system processing (psia)		

Notes:

*For complete description of these uncertainties, see Appendix B.

**Design flow = 190,000 GPM

***2 σ values

****Includes [ ]

TABLE 3-2

IMPACT OF STATISTICAL COMBINATION OF
UNCERTAINTIES ON MARGIN TO SAFDL

		Approximate Values of Equivalent Overpowr Margin (%)	
		DNB	APD
Uncertainty	Value	LSSS	LSSS
Power	2% of rated		
Core coolant inlet temperature	$2^{o}F$		
Reactor coolant system pressure	22 psid		
Axial shape index:			
Separability	.02 asiu		
Shape annealing	.01 asiu		
Calibration	.01 asiu		
Reactor coolant system flow	5460 gpm		
Peaking factors	5% DNB, 7% APD		
Equipment processing:			
DNB LSSS	42 psid		
APD LSSS	.02 asiu		
Total uncertainty applied previously	Total		
Total uncertainty statistically combined			
Net margin gain			

TABLE B1-1

UNCERTAINTY [AND BIAS] COMPONENTS FOR THE EVALUATION OF THE PERIPHERAL SHAPE INDEX (1)

	K 95/95 (asiu)	K(f) (2)	Bias
I. Separability Uncertainty			
II. Calibration Uncertainty(n)			
III. Shape Annealing Uncertainty(n)			
IV. Monitoring System Processing Uncertainty			
LHR (asiu)			
DNBR (psia)			

Notes on Table 1

(1) All components of the peripheral shape index have been tested for normality, and where indicated, satisfy that distributional requirement(n).

(2) f = degrees of freedom.

(3) This uncertainty is conservatively [ ].

(4) This $K\sigma^{95/95}$ is for consistent sets of input data used by the uncertainty processors.

CEN-257(O)-P, -NP, Supplement 1

STATISTICAL COMBINATION OF UNCERTAINTIES

PART 2

AUGUST, 1985

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B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method or process disclosed in this report.

ABSTRACT

This supplement contains the revised MDNBR limit for Fort Calhoun Nuclear Unit 1. This revision has been made based on NRC acceptance of the reduced (1.15) CE-1 CHF correlation DNBR limit for C-E's 14x14 fuel. This supplement demonstrates that there is at least 95% probability with at least 95% confidence that the limiting fuel pin will avoid departure from nucleate boiling (DNB) so long as the MDNBR found with the best estimate design CETOP-D model remains at or above 1.18. The 1.18 MDNBR limit supersedes the value of 1.22 in the report CEN-257(O)-P.

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Supplement 1

STATISTICAL COMBINATION OF UNCERTAINTIES

COMBINATION OF SYSTEM PARAMETER UNCERTAINTIES

IN THERMAL MARGIN ANALYSIS FOR FORT CALHOUN NUCLEAR UNIT 1:

REVISED CHF CORRELATION UNCERTAINTY

Report Summary

As described in Reference 2, the NRC has approved the CE-1 CHF Correlation, Part 2, which contains non-uniform axial power distribution CHF data. As a result, the MDNBR limit for the C-E 14x14 fuel with standard spacer grid is 1.15, higher than the 1.13 limit C-E had originally proposed.

The statistically derived 1.22 MDNBR limit reported for Fort Calhoun Nuclear Unit 1 in Reference 1 had included an NRC imposed interim penalty which increased the CHF correlation DNBR limit from 1.13 to 1.19, pending NRC's review of Part 2 of the CE-1 CHF correlation. Taking into account approval of a 1.15 CHF correlation DNBR limit, the new limit for Fort Calhoun Unit 1 has been determined to be 1.18, replacing the 1.22 limit previously reported in Reference 1.

MDNBR Limit Pending Approval of CE-1 Correlation, Part 2

The MDNBR limit for Fort Calhoun Nuclear Unit 1 was 1.22 as reported in Reference 1. This limit included a penalty on the CE-1 CHF correlation, pending NRC's review of the non-uniform axial power distribution data documented in Reference 2. This penalty was included in the 1.22 limit as a shift of the mean of the DNBR p.d.f by 0.06, a conservative equivalent to a 5% penalty on the 1.13 DNBR limit.

Adjusted MDNBR Limit

After completing a review of CE's non-uniform axial power distribution CHF data (Reference 2), the NRC concluded that a 1.15 DNBR limit should be used for CE's 14x14 fuel design. Since the 1.22 limit was generated prior to the NRC's approval of the 1.15 CE-1 CHF correlation DNBR limit, it included an uncertainty penalty based on the NRC's interim CHF correlation DNBR limit of 1.19. The 1.22 limit can thus be reduced.

The statistical combination of uncertainties analysis for Fort Calhoun Nuclear Unit 1 resulted in a DNBR limit of 1.137 (p. 6-2 of Reference 1), which was increased to include a 0.5% allowance for fuel rod bow, a 0.06 (=1.19 - 1.13) CHF correlation uncertainty penalty and a TORC code uncertainty penalty of 1.1% as follows:

$$1.137 \times 1.005 = 1.1427$$
$$1.1427 + 0.06 = 1.2027$$
$$1.2027 \times 1.011 = 1.216$$

which was rounded up to 1.22, the final MDNBR limit reported in Reference 1. The CHF correlation uncertainty penalty can now be reduced from 0.06 to 0.02 (=1.15 - 1.13). The revised DNBR limit is:

$$1.137 \times 1.005 = 1.1427$$
$$1.1427 + 0.02 = 1.1627$$
$$1.1627 \times 1.011 = 1.1755$$

which can be rounded up to 1.18.

CONCLUSION

Use of a 1.18 MDNBR limit with a best-estimate design CETOP-D model for Fort Calhoun Nuclear Unit 1 will ensure with at least 95% probability and 95% confidence that the hot pin will not experience a departure from nucleate boiling. The 1.18 MDNBR limit includes explicit allowances for system parameter uncertainties, CHF correlation uncertainty, rod bow, the NRC penalty for TORC code uncertainty and the revised penalty on the CE-1 CHF correlation.

REFERENCES

1. "Statistical Combination of Uncertainties, Part 2", CEN-257(0)-P, November, 1983.

2. "C-E Critical Heat Flux, Part 2: Non-Uniform Axial Power Distribution", CENPD-207-P-A, December, 1984.

CEN-257(O)-NP SUPPLEMENT 1 -NP

STATISTICAL COMBINATION OF UNCERTAINTIES

PART 3

AUGUST, 1985

POWER SYSTEMS
COMBUSTION ENGINEERING, INC

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B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method or process disclosed in this report.

ABSTRACT

This Supplement updates the original report (CEN-257(0)-P, Part 3, November, 1983) to account for the addition of BASSS/MINI CECOR to the DNB LCO monitoring system. Because this new system is based on continual monitoring of the power distribution using in-core detectors, the description of the monitoring system, the analysis of the net uncertainties and the net uncertainty reported have been changed.

The numbered sections of this report supersede the same numbered sections of the original report. In some sections significant additional material has been added. Sections 1.2.1, 1.4, 1.5, 2.3.2, 2.4.1.2, 2.4.1.3, 2.5, 3.1, 3.1.1, 3.1.2, 3.2.1, A.1, A.3, A.4 and Table 3-1 were changed. Figure 2-2 is added.

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SUPPLEMENT 1 TO
STATISTICAL COMBINATION OF
UNCERTAINTIES
PART 3
(CEN-257(O)-P)

REPORT SUMMARY

Analysis of the DNB - LCO uncertainty factor with BASSS/mini-CECOR follows the same procedure as the DNB - LCO uncertainty analysis described in CEN-257(O)-P, Part 3. Two changes in input uncertainties related to power distribution monitoring have been made. The [ ] processing uncertainty is no longer required and the only measurement uncertainty component for [ ] uncertainty. This uncertainty is given in CEN-257(O)-P, Part 3, App. B1, Table B1-1. All of the other non-ASI uncertainties remain the same. The total uncertainty factor is [ ].

1.2.1 PROTECTION AND MONITORING SYSTEM

The basic purposes and interactions of the LSSS and LCO's were previously described in Section 1.2.1 of Part 1 of this report. Part 1. describes the function of the protection system; Part 3 describes the function of the DNB and LHR LCO's.

Operation within the DNB and LHR LCO's provides the necessary initial DNB and LHR margin to prevent exceeding acceptable limits during Design Basis Events (DBE's) where changes in DNBR and linear heat rate are important. A list of the Nuclear Steam Supply System (NSSS) parameters which affect the calculation of these LCO's is shown in Table 1-2. A discussion of C-E setpoint methodology may be found in Reference 1-3.

Either the ex-core or the in-core detectors can be used to monitor the LHR LCO for C-E designed reactors. For Fort Calhoun Station Unit 1, the DNB LCO and axial shape index can now be monitored on in-core detectors as well as on the ex-core detectors. This use of in-core detectors is described in reference 1-5. Although these two DNB LCO monitoring systems are functionally similar in that they correlate allowed power levels and axial shape indices, use of the in-core detectors rather than the ex-core detectors implies different [ ]. These differences are noted herein.

Note: A vertical line is drawn in the right hand margin where a change has been made.

1.4 SUMMARY OF RESULTS

The analytical methods presented in Section 2.0 are used to show that a stochastic simulation of uncertainties associated with the ex-core detector-monitored DNB and LHR LCO's results in aggregate uncertainties of [ ] and [ ], respectively, at a 95/95 probability confidence level. The total uncertainties previously applied to the ex-core DNB and LHR LCO's are approximately [ ] and [ ], respectively. Therefore, the statistical combination of uncertainties program provides a reduction in the conservatism of the uncertainties applied in establishing the ex-core instrument monitored DNB and LHR LCO's of approximately [ ] and [ ], respectively. The stochastic simulation of uncertainties associated with in-core monitoring of the DNB LCO results in an aggregate uncertainty of [ ] at the 95/95 probability/confidence level.

1.5 REFERENCES FOR SECTION 1

1-1 CEN-270-(0)-P, "Statistical Combination of Uncertainties," Part 1, November 1983.

1-2 CEN-270-(0)-P, "Statistical Combination of Uncertainties," Part 2, November 1983.

1-3 CENPD-199-P, Rev. 1-P, Rev. 1-P, "C-E Setpoint Methodology," March 1982.

1-4 Letter, E. G. Tourigny (NRC) to W. C. Jones (OPPD) dated March 5, 1983, License Amendment 70 and SER for Cycle 8 Operation of Fort Calhoun Station Unit No. 1, Docket No. 50-285.

1-5 CEN-119(B)-P, BASSS, November 1979.

2.3.2 DNB LCO STOCHASTIC SIMULATION

For the DNB LCO, DNB overpower (P_{fdn}) divided by the required overpower margin (ROPM) is the dependent variable of interest. the core coolant inlet temperature, reactor coolant system pressure and flow rate, peripheral axial shape index and integrated radial peaking factor are the independent variables of interest. As demonstrated in Appendix C, ROPM is relatively insensitive to these independent variables. In addition, the maximum ROPM as a function of shape index is used as input to generate the LCO's. This reduces the analytical evaluation of the dependent variable to consideration of the P_{fdn}'s response to the uncertainties of the independent variables. TORC/CE-1 (References 2-2, 2-3) is used to determine the functional relationship between P_{fdn} and the independent variables. The probability distribution of uncertainties associated with some of the independent variables have been discussed in Appendix A of Part 1 of this report. Those uncertainties specifically associated with the calculation of the core average axial shape index using the in-core detector system to monitor the DNB LCO are discussed in Appendix A of this part of the report.

The core coolant inlet temperature range of interest for the DNB LCO stochastic simulation is defined by:

(1) the temperature at which the secondary safety valves open, and
(2) the temperature at which the low secondary pressure trip would occur

The reactor coolant system pressure range of interest for the DNB LCO stochastic simulation is defined by:

(1) the value of the high pressurizer pressure trip setpoint, and
(2) the lower pressure limit of the thermal margin/low pressure trip

It is noted that these ranges are the same as used in the LSSS stochastic simulation (Ref. 2-1) and as such are bounding for the LCO.

Figure 2-1 is a flow chart representing the ex-core detector monitoring stochastic simulation of the DNB limits. This figure is similar to Figure 2-1 in Part 1. Figure 2-2 is a flow chart representing the in-core detector monitoring stochastic simulation of the DNB limits. This figure differs from

Figure 2-1 in that the [

]

stochastic simulation. The independent variables and their uncertainties are input to CESCU. Each data set generated by the statistical part of CESCU is evaluated with the CETOP-D portion of CESCU to generate a P_{fdn} probability distribution. The ratio of the mean value of P_{fdn} to the lower 95/95 value of P_{fdn} is the parameter of interest. The details of the specific DNB LCO stochastic simulations performed are presented in Section 2.4.

2.4.1.2 AXIAL SHAPE INDEX UNCERTAINTY SIMULATION

2.4.1.2.1 EX-CORE AXIAL SHAPE INDEX

At Fort Calhoun Unit 1, the digital display of safety channel information is used tomonitor the LHR and DNB LCO's, in accordance with Technical Specification 2.10.4. Thus, the basic relationships between the components of the axial shape index uncertainty for LSSS, described in Appendix B1 of Part 1, are also appropriate for the LCO uncertainty analysis.

2.4.1.2.2 CORE AVERAGE AXIAL SHAPE INDEX

The uncertainties associated with the in-core detector system have been developed in support of the better axial shape selection system (Reference 2-5). The magnitude of those uncertainties are defined in Appendix A of this part of the report.

The procedure used to sample the shape index uncertainty distributions for the LCO stochastic simulation are those described in Section 2.4.1.2 of Part 1.

2.4.1.3 PROCESSING UNCERTAINTY SIMULATION

2.4.1.3.1 EX-CORE INSTRUMENT PROCESSING

The signals generated by the ex-core detectors are processed into an axial shape index (ASI) value. The electronic processing equipment introduces further uncertainty in these values. However, the safety channel information used is transferred to the digital display before the information is processed by the trip actuation evaluation circuits. Therefore, only the safety channel processing uncertainties need by included in the LCO processing uncertainties. Since the axial power distribution and the ASI value used in each simulation calculation are correlated, this uncertainty is incorporated in the stochastic evaluation of the LCO.

2.4.1.3.2 IN-CORE INSTRUMENT PROCESSING

As noted in Appendix A the processing uncertainty for the in-core instrument signals has been [

]

2.5 REFERENCES FOR SECTION 2

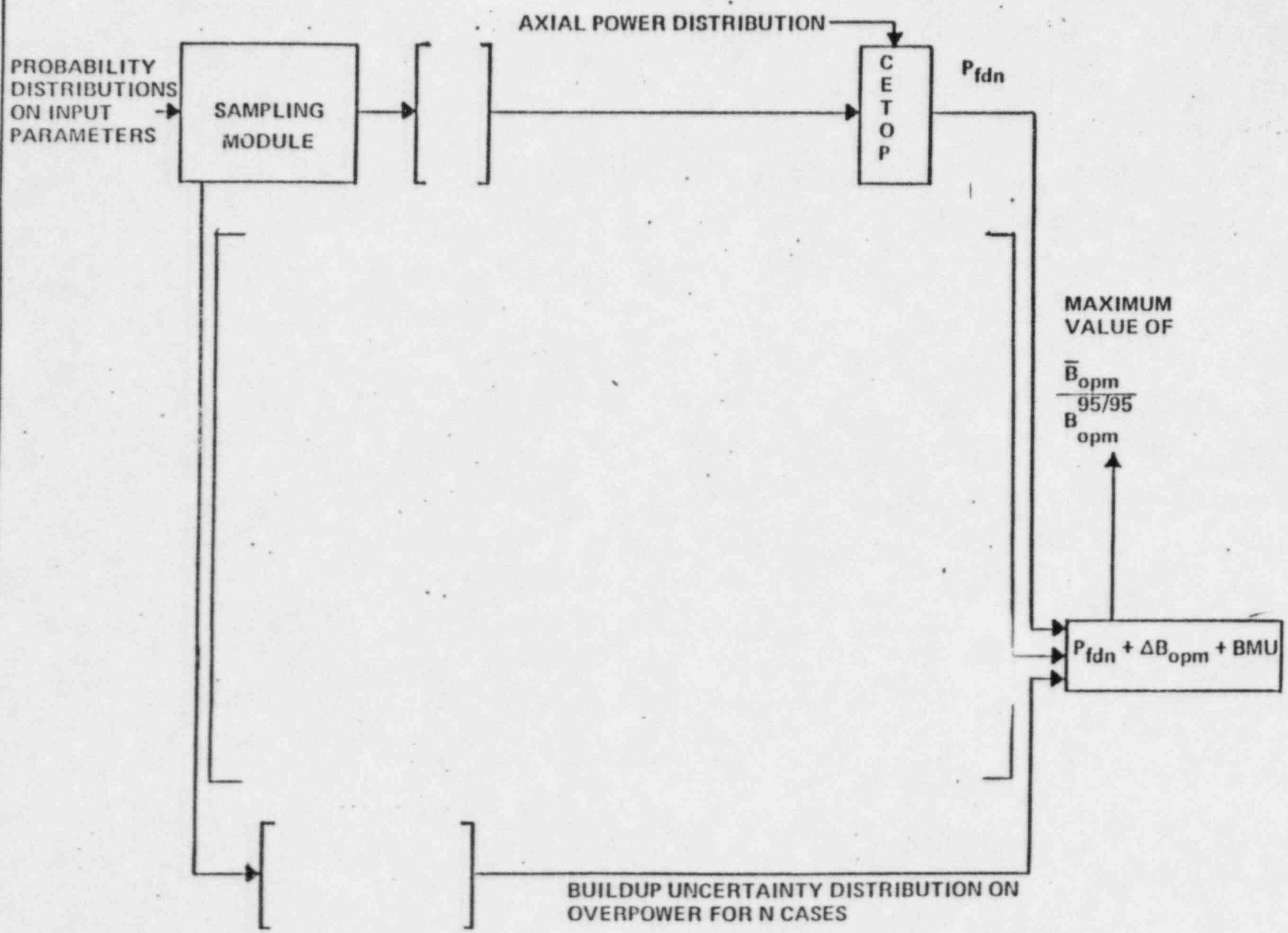
2-1 CEN-257(O)-P, "Statistical Combination of Uncertainties," Part 1, November 1983.

2-2 CENPD-161-P, "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," July, 1975.

2-3 CENPD-206-P, "TORC Code: Verification and Simplified Modeling Methods," January, 1977.

2-4 C. Chiu, J. F. Church, "Three-Dimensional Lumped Subchannel Model and Prediction-Correction Numerical Method for Thermal Margin Analysis of PWR Cores," TIS-6191, June, 1979.

2-5 CEN-119(B)-P, BASSS, November 1979.



OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN
Nuclear Power Plant

IN-CORE DETECTOR MONITORED DNB LCO UNCERTAINTY ANALYSIS

Figure 2-2

3.0 RESULTS AND CONCLUSIONS

3.1 RESULTS OF ANALYSES

The statistical analytical methods presented in Section 2 have been used to show that a stochastic simulation of uncertainties associated with the ex-core monitored DNB and LHR LCO result in combined uncertainties of [ ] and [ ], respectively, at a 95/95 probability confidence level. Stochastic simulation of the in-core monitored DNB LCO results in an aggregate uncertainty of [ ].

Table 3-1 shows the values of the individual uncertainties which were statistically combined to yield the above combination. Appendix A contains a further discussion of the bases for these individual uncertainties.

The combined uncertainties are in units of percent overpower (P_{fdn}, P_{fdl}) and are applied as such in the generation of the LCO limits as discussed below (Reference 3-1).

3.1.1 DNB LCO

The fuel design limit on DNBR for the DNB LCO is represented by a combination of the ordered pairs (P_{fdn}, ASI_{DNB}). A lower bound is drawn under the "flyspeck" data such that all the core power distributions analyzed are bounded. This lower bound is reduced by applicable uncertainties as follows:

[ ] (3-1)

(3-2)

(3-3)

*Equations 3-2 and 3-3 are valid for the excore and incore monitoring systems, respectively.

where:

B_{DNB}^{LCO} - DNB Power limit for LCO after inclusion of uncertainties and allowances

P_{fdn} - Power to fuel design limit on DNB including the effects of azimuthal tilt

SMDO - Statistically combined uncertainties applicable to the DNB LCO

ASI_{DNB} - Axial shape index associated with P_{fdn}

Temperature, pressure and flow components of the DNB LCO are represented by equations as folows:

[ ] (3-4)

(3-5)

(3-6)

where:

F^{DNB}, P^{DNB}, T_{in}^{DNB} = Coolant conditions used in the calculations of (P_{fdn}, I_p) ordered pairs of data.

3.1.2 LHR LCO

The excore detector monitored LCO on linear heat rate is represented by the ordered pairs (P_{fdl}, I_p). A lower bound is drawn under the "flyspec" data such that all the core power distributions analyzed are bounded. This lower bound is reduced by the applicable uncertainties and allowances to generate the LCO as follows:

(3-7)

(3-8)

where:

B^{LCO}_{LHR} - Linear Heat Rate Power Limit for LCO after inclusion uncertainties.

P^{LCO}_{fdl} - The power to the LCO linear heat rate limit including the effects of azimuthal tilt.

SMLO - Statistically combined uncertainty applied to the LHR LCO

The incore detector monitored LCO on linear heat rate will not be modified for a statistical combination of uncertainties.

3.2 IMPACT OF STATISTICAL COMBINATION OF UNCERTAINTIES

3.2.1 IMPACT ON MARGIN

The motivation for using a statistical combination of uncertainties is to improve NSSS performance through a reduction in analytical conservatism in the uncertainties which must be taken into account. This section contains a discussion of the margin obtainable through a reduction in this conservatism.

Table 3-2 lists the uncertainty values previously used on Fort Calhoun Unit 1. The approximate worth of each of these uncertainties in terms of percent overpower margin (P_{fdn}, P_{fdl}) is shown.

The total uncertainties previously applied to the excore monitored DNB and LHR LCO are approximately [ ] and [ ], respectively. The use of the statistical combination of uncertainties justifies a reduction in the conservatism in the uncertainty of approximately [ ] and [ ], respectively. The use of the statistical combination of uncertainties and incore detector monitoring of the DNB LCO results in an uncertainty of approximately [ ].

Although the conservatism in the uncertainty has been reduced, a high degree of assurance remains that acceptable limits will not be exceeded.

TABLE 3-1

UNCERTAINTIES ASSOCIATED WITH THE DNB AND LHR LCO'S

Uncertainty	LHR LCO	DNB LCO
Core power (% of rated power)	±2%	±2%
Primary coolant mass flow (% Design flow)*	NA	[ ]
Primary coolant pressure (psia)	NA	[ ]
Core coolant inlet temperature (F^{o})	NA	[ ]
Power distribution (peaking factor)	[	]
Axial Shape Index (Excore Detector System)		
1. Separability (asiu)	See Table A-1 of Appendix A1	
2. Calibration (asiu)	[ ]	[ ]
3. Shape Annealing (asiu)	[	]
4. Monitoring system processing (asiu) (2σ)	[	]
Axial Shape Index (Incore Detector System) (asiu)		[ ]

Note: For complete description of these uncertainties, see Appendix A.

*Design Flow: 190,000 gpm

A.1 SHAPE INDEX UNCERTAINTIES

A.1.1 AXIAL SHAPE INDEX UNCERTAINTIES ASSOCIATED WITH THE EXCORE DETECTOR SYSTEM

At Fort Calhoun Unit 1, the safety channel instruments, which supply some of the input information to the trip system, also supply information for monitoring the shape index of the LCO's. This shape index information is supplied to the LCO monitors before the trip system evaluation of shape index is performed. Consequently, except for the processing uncertainty for shape index, uncertainties described for the LSSS in Appendix A-1 Part 1 (Reference A-1) are appropriate for evaluation of the LCO shape index uncertainty. The LCO shape index processing uncertainty should include only that part of the LSSS processing uncertainty attributed to the ex-core detectors, and must exclude that portion of the uncertainty due to the trip system shape index evaluation circuits. The processing uncertainties are described in A.3 below. The other shape index uncertainties are given in Table A-1.

A.1.2 AXIAL SHAPE INDEX UNCERTAINTIES ASSOCIATED WITH THE INCORE DETECTOR SYSTEM

The incore detectors are used to calculate the core average axial shape index, $\overline{I}$. The value of $\overline{I}$ calculated from the incore signals is used to monitor the DNB LCO in the BASSS/Mini-CECOR system. Mini-CECOR is just a mini computer version of C-E's CECOR code. CECOR and its uncertainties are described in Reference A-2.

A.3 MONITORING SYSTEM PROCESSING UNCERTAINTIES

A.3.1 EXCORE MONITORING SYSTEM

The description of the Trip System Processing Uncertainties given in Appendix B3 of Part 1 is valid for the Fort Calhoun excore monitored DNB LCO because the same instruments are used.

A.3.2 IN-CORE MONITORING SYSTEM

The processing uncertainties for the DNB LCO in-core monitoring system result from the [

].

A.4 REFERENCES FOR APPENDIX A

A-1 CEN-257R-(O)-P, "Statistical Combination of Uncertainties Part 1," November, 1983.

A-2 A. Jonsson, W. B. Terney, M. W. Crump, "Evaluation of Uncertainty in the Nuclear Power Peaking Measuring the Self-Powered, Fixed In-Core Detector System," CENPD-153-P, Rev. 1-P, May, 1980.