LIC-03-0122 Attachment 6

Non-Proprietary Framatome Evaluation

PWR Fuel Design Criteria and Statistical Setpoint Calculations for Fort Calhoun Station Measurement Uncertainty Recapture Power Uprate (EMF-2904(NP))

(No changes from LIC-03-0067 Attachment 6)

EMF-2904(NP)
Revision 1

PWR Fuel Design Criteria and Statistical Setpoint Calculations for Fort Calhoun Station Measurement Uncertainty Recapture Power Uprate

June 2003

Framatome ANP, Inc.

EMF-2904(NP)
Revision 1

PWR Fuel Design Criteria and Statistical Setpoint Calculations for Fort Calhoun Station Measurement Uncertainty Recapture Power Uprate

Prepared:	G. S. Uyeda, Engineer PWR Engineering	06/13/03 Date
Prepared:	K. M. Duggan, Engineer Reload Mechanical Analysis	6/13/03 Date
Prepared:	for B. A. Reeves, Engineer PWR Safety Analysis	6/13/03 Date
Approved:	for J. J. Cudlin, Manager Analysis Services / PWR Safety Analysis	6/13/03 Date
Approved:	David M. Brown D. M. Brown, Manager PWR Engineering	6/16/03 Date
Approved:	for D. E. Garber, Manager BWR Engineering	6-17-03 Date
Approved:	J. S. Holm, Manager Product Licensing	6/13/03 Date
Approved:	E. M. Miller, Project Manager Contract Management & Proposals	6/13/03 Date

:sct

P104,103 Document Review Checklist

Document Number: EMF-2904(NP) Revision 1

Document Title: PWR Fuel Design Criteria and Statistical Setpoint Calculations for Fort Calhoun Station Measurement Uncertainty Recapture Power Uprate

Items (at minimum) to be checked:

1. Verify that calculation results in document match values in supporting calculation notebooks (including any revisions) or appropriate references.
2. Verify that document is complete (i.e. no dropped lines / pages, page numbers, etc.).
3. Verify that correct figure and table titles.
4. Verify that Table of Contents is correct.
5. Verify that reference numbers in text match reference numbers in reference section.

The signatures below indicate that the results presented in this document have been verified in accordance with EMF-1928, P104,103.

Section(s)	Reviewer	Date
1.0, 2.0, 5.0, 6.0		06/16/03

Number of changes made as a result of this P104,103 review: Ø

Change Number	Description of Change

P104,103 review of On-Line Test Document completed: ____________

Customer Disclaimer

Important Notice Regarding the Contents and Use of This Document

Please Read Carefully

Framatome ANP, Inc.'s warranties and representations concerning the subject matter of this document are those set forth in the agreement between Framatome ANP, Inc. and the Customer pursuant to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Framatome ANP, Inc. nor any person acting on its behalf:

a. makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights;

or

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 document.

The information contained herein is for the sole use of the Customer.

In order to avoid impairment of rights of Framatome ANP, Inc. in patents or inventions which may be included in the information contained in this document, the recipient, by its acceptance of this document, agrees not to publish or make public use (in the patent use of the term) of such information until so authorized in writing by Framatome ANP, Inc. or until after six (6) months following termination or expiration of the aforesaid Agreement and any extension thereof, unless expressly provided in the Agreement. No rights or licenses in or to any patents are implied by the furnishing of this document.

Nature of Changes

Item	Page	Description and Justification
1.	Section 2.0	Summary section revised.
2.	Section 5.0	Revised setpoint verification writeup.

Contents

Tables

Figures

Nomenclature

AOO	Anticipated Operational Occurrences
APD	Axial Power Distribution
ARO	All Rods Out
ASI	Axial Shape Index
ASME	American Society of Mechanical Engineers
BASSS	Better Axial Shape Selection System
BOC	Beginning of Cycle
BOL	Beginning of Life
CE	Combustion Engineering
CEA(D)	Control Element Assembly (Drop)
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CUF	Cumulative Usage Factor
DNB(R)	Departure from Nucleate Boiling (Ratio)
EOC	End of Cycle
EOL	End of Life
ESFAS	Engineered Safeguard Feature Actuation System
FANP	Framatome ANP, Inc. (Advanced Nuclear Power)
FCM	Fuel Centerline Melt
HFP	Hot Full Power
HPSI	High Pressure Safety Injection
HTP	High Thermal Performance
LCO	Limiting Conditions for Operation
LHR	Linear Heat Rate
LOCA	Loss of Coolant Accident
LOCF	Loss of Coolant Flow
LSSS	Limiting Safety System Setting
LTIL	Long Term Insertion Limit
LTP	Lower Tie Plate
LWR	Light Water Reactor
M&TE	Measurement and Test Equipment
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MFIV	Main Feedwater Isolation Valve
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MUR	Measurement Uncertainty Recapture (Power Uprate)

NAF	Neutron Absorber Fuel
NRC	(U. S.) Nuclear Regulatory Commission
OPPD	Omaha Public Power District
PDIL(-ΔP)	Power-Dependent Insertion Limit (minus Delta-Power) ≡ sub-PDIL insertion
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
RTD	Resistance Temperature Detector
SAF	Shape Annealing Factors
SAFDL(s)	Specified Acceptable Fuel Design Limit(s)
SRP	Standard Review Plan
TMLL(s)	Thermal Margin (Safety) Limit Line(s)
TM/LP	Thermal Margin/Low Pressure
TPC	Thermal Power Calculator
USAR	Updated Safety Analysis Report
UTP	Upper Tie Plate
VHPT	Variable High Power Trip

1.0 Introduction

This report documents the FANP fuel design and safety analysis calculations and event dispositions supporting the Fort Calhoun Station MUR Appendix K power uprate project.

OPPD is evaluating the impact of the installation of an enhanced feedwater flow measurement system. This system would allow Fort Calhoun Station to perform a power uprate up to a maximum of 1.7%.

This report documents the following engineering analyses, considering a 1.7% power uprate to the pre-MUR rated power of 1500 MWt:

- Mechanical design analyses evaluating the impact of the MUR power uprate upon the FANP PWR generic design criteria.
- A disposition of the Main Steam Line Break Incident analysis of record. In parallel, OPPD examined all of the USAR Chapter 14 analyses of record and concluded that the transient analyses of record remain bounding and do not require re-analysis, per Reference 9. (The Main Steam Line Break examination documented in this report is a *more in-depth disposition than that performed by OPPD*). *Since all of the Chapter 14* transient analyses of record remain bounding, the associated MDNBR and fuel centerline temperature results also remain bounding.
- Statistical setpoint analyses evaluating the impact of the MUR power uprate changes upon the LSSS and LCO functional margins related to protecting fuel SAFDLs. Additionally, the TM/LP safety limit lines were reevaluated.

2.0 Summary

The following subsections summarize the results of the various calculations/dispositions documented herein.

2.1 *PWR Fuel Design Criteria*

The mechanical analyses and evaluations of previous analyses confirm that both the FTC-6 and FTC-7 fuel continue to meet the approved design criteria for Cycle 21 with the MUR power uprate.

2.2 *Disposition of the Main Steam Line Break Incident*

The Main Steam Line Break analysis of record remains applicable for MUR power uprate conditions.

2.3 *Statistical Setpoint Analyses*

The limiting statistical setpoint analyses of record were evaluated under MUR power uprate conditions to assess the shifts in margins due to the uprated power and decreased power measurement uncertainty.

The TM/LP safety limit lines also considered all relevant plant changes since the analysis of record was conducted (Cycle 20).

Positive pressure/power margin to the respective SAFDLs was calculated for the analyzed LSSSs and the DNB LCO, therefore, DNB and FCM are both avoided with at least a 95% probability (DNB-related setpoints at a 95% confidence). In addition, the Technical Specifications limit of 15.5 kW/ft is supported. Therefore, the current configurations for these setpoint functions are verified for the Fort Calhoun Station MUR, subject to analysis conditions and assumptions.

The FCM limit of 22.0 kW/ft was demonstrated to be conservative for the MUR, based upon Cycle 21 power distributions and core design.

The TMLLs depicted in Figure 1-1 of the Fort Calhoun Station Technical Specifications (Reference 6) continue to conservatively represent the frontiers of hot leg saturation and DNB for post-MUR power uprated conditions.

3.0 Mechanical Evaluation

Mechanical design analyses of the Fort Calhoun Station MUR 1.7% power uprate have been performed using NRC-approved mechanical design analysis methodology (References 1 and 2). The analyses address the FANP PWR generic design criteria (Reference 3).

The analyses demonstrate that the mechanical design criteria for the fuel rod and fuel assembly design are satisfied for the MUR. The evaluation was performed to a peak assembly average exposure of 58000 MWd/MTU and a peak rod average exposure of 62000 MWd/MTU when the fuel is operated within the peaking limits given in the Technical Specifications. The analyses and evaluations of previous analyses confirm that both the FTC-6 and FTC-7 fuel continue to meet the approved design criteria for Cycle 21.

Table 3.1 provides a summary of the reactor information that was used for the mechanical design evaluations and compares that information with the current reactor information.

Table 3.1 Comparison of Reactor Operating Conditions for MUR Mechanical Evaluations

Parameter	Current Value	MUR
Core Thermal Power, MWt	1500	**1526**
System Pressure, psia	2100	2100
Number of Assemblies	133	133
Nominal Total Core Flow Rate, Mlbm/hr	78.0	**78.3**
Core Inlet Temperature, °F	543.0	543.0
Core Outlet Temperature, °F	596.0	**596.8**
Maximum Overpower, %	112	112
Fraction of Heat from Fuel Rods	0.975	0.975
Core Average LHR, kW/ft	6.02	**6.12**
Maximum Peak Power Factor, F_q^T	2.57	**2.53**
Maximum Rod Peaking Factor, F_R^T	1.853	1.853
Peak Assembly Burnup, GWd/MTU	58.0	58.0
Peak Rod Burnup, GWd/MTU	62.0	62.0

4.0 Main Steam Line Break Incident Disposition

Any changes to the following Main Steam Line Break analysis parameters can potentially have significant effects on the analysis results:

- Initial core-average moderator temperature
- Steam generator outlet nozzle flow area
- Most-negative MTC
- Minimum shutdown margin
- Power peaking with all CEAs inserted (except most reactive CEA stuck out)
- ESFAS design that responds to Main Steam Line Break event by closing MSIVs and MFIVs and actuating safety injection (including setpoints and delays) but not actuating auxiliary feedwater
- HPSI pump minimum flow curve
- Total safety injection line purge volume

Only one of these key parameters—the initial core-average moderator temperature—is changing in connection with the MUR power uprate project. The effect of that change is discussed below (in the third following paragraph).

It should be noted that the rated thermal power, which is increasing by 1.7%, is not a key Main Steam Line Break analysis parameter. This is discussed in the following paragraph.

The full-power cases of the Main Steam Line Break analysis of record were initiated at the nominal rated power in effect prior to the power uprate. The analytical methodology used for the analysis does not require that the initial power level be biased to account for measurement uncertainty, because the initial power level used for such analyses has an insignificant effect on the post-scram return to power. Thus, from the standpoint of the initial power level, essentially the same results for the full-power cases would be obtained if they were to be rerun with a 1.7% greater initial power level.

The core-average moderator temperature at full power subsequent to the power uprate will be slightly greater (by 0.4°F) than the initial value used for the full-power cases of the Main Steam Line Break analysis of record. To view this in perspective, the inlet temperature of the affected core sector was calculated to decrease by more than 280°F during the limiting full-power Main Steam Line Break event. Thus, from the standpoint of the initial core-average moderator temperature, essentially the same results for the full-power cases would be obtained if they were to be rerun with a 0.4°F greater initial core-average moderator temperature.

Therefore, it may be concluded that the Main Steam Line Break analysis of record remains applicable for the power uprate conditions.

5.0 Statistical Setpoint Verifications

The LSSS and LCO setpoints that protect the DNB and FCM SAFDLs are evaluated for each cycle of operation. The LSSSs that are assessed are the TM/LP LSSS, which protects the DNB SAFDL and precludes hot leg saturation, and the APD LSSS, which protects the FCM SAFDL. Also verified every cycle are the DNB LCO, which protects the DNB SAFDL, and the LHR LCO, which protects the LOCA LHR limit. All of the setpoints are verified to ensure that they preclude violation of these limits with at least a 95% probability (DNB related setpoints at a 95% confidence level) throughout the cycle.

These setpoint functions were re-evaluated for shifts in margins due to the implementation of the 1.7% Appendix K power uprate for the MUR project.

The suite of axial shapes, generated for the Cycle 21 setpoint analyses, conservatively bounds the range of possible CEA insertions by considering ARO to sub-PDIL (PDIL-ΔP) positions. The measurement uncertainties associated with monitored plant inputs to the various setpoints are typically treated in a statistical fashion. Table 5.1 contains general plant uncertainties for Fort Calhoun Station that were supported throughout the various statistical setpoint calculations. Other variables may be treated statistically in specific setpoint calculations; these are discussed topically within the pertinent sections.

The reduced power measurement uncertainty, shown in Figure 5.1, is an assumed value to be supported in the setpoint calculations. If calibration calculations on the new feedwater system demonstrate that the actual uprated power and calorimetric uncertainty deviate slightly from the assumed values shown in Table 5.2 and Figure 5.1, then the calculated margins may shift, but not significantly enough to invalidate the fact that adequate margins exists for the MUR power uprate.

At the time these analyses were being conducted, there were no Cycle 22 core design data or neutronics inputs to the setpoint analyses available. Therefore, the calculations documented herein are based upon Cycle 21 pin power distributions, core design, and setpoint axial data. The setpoint axials and assembly pin power distributions should remain representative for uprated conditions.

5.1 *Analytical Methodology*

The analyses herein have been performed in accordance with the NRC-approved statistical setpoint methodology for verifying analog LSSSs and LCOs in plants of CE design (Reference 4).

5.2 *Acceptance Criteria*

The LSSSs and LCOs that are the subject of this report are designed to preclude fuel failure during normal operation and AOOs.

The DNB SAFDL precludes fuel failure due to DNB. When the MDNBR on the limiting pin is above the upper 95/95 bound on the applicable critical heat flux correlation (adjusted for mixed core penalties), DNB is precluded with at least a 95% probability, at a 95% confidence level.

The FCM SAFDL precludes fuel failure due to FCM. The current FCM limit of 22.0 kW/ft for UO_2 fuel, per Technical Specification 1.3(8) (Reference 6), will be supported for the MUR power uprate analysis. The verification analysis performed on the APD LSSS function confirms that the 22.0 kW/ft limit is not exceeded during a limiting AOO of maximum FCM challenge. A FCM power analysis confirms that the 22.0 kW/ft LHR limit is lower than the minimum LHR at which FCM will be experienced in HTP fuel.

In summary, the following are the acceptance criteria for the specific LSSSs and LCOs discussed herein:

<u>TM/LP LSSS</u>

Positive pressure margin exists between the pressure at which the TM/LP LSSS trip occurs and the pressure at which DNB would be experienced, for any conditions expected during Cycle 21 with MUR uprated power and reduced power measurement uncertainty. The margin is a statistically adjusted 5/95 bound, and is based upon the upper 95/95 limit on the HTP DNB correlation.

<u>APD LSSS</u>

Positive power margin exists between the power at which the APD LSSS trips and the power at which FCM would be experienced, for the conditions of maximum severity expected during any of the limiting AOO events for the LSSS during Cycle 21 with MUR uprated power and reduced power measurement uncertainty. The margin is a statistically adjusted lower 95% bound, and is based upon the FCM limit as documented in Technical Specification 1.3(8) (Reference 6).

DNB LCO

All statepoints at which DNB is experienced will lie outside the region of allowable operation as described by the DNB LCO barn, for the conditions of maximum severity expected during any of the limiting AOO events for the LCO during Cycle 21 with MUR uprated power and reduced power measurement uncertainty. The power margin between the maximum allowed power and the power at which DNB occurs will be a 5/95 bound, and is based upon the upper 95/95 limit on the HTP DNB correlation.

Excore LHR Monitoring LCO

All statepoints at which the LOCA LHR limit is experienced will lie outside the region of allowable operation as described by the Excore Monitoring of LHR LCO barn, for any steady-state condition expected during Cycle 21 with MUR uprated power and reduced power measurement uncertainty. The margin between the power corresponding to the LOCA LHR limit and maximum allowed LCO power is a statistically adjusted lower 95% bound, and is based upon the 15.5 kW/ft limit as documented in Figure 3 of Reference 7.

TMLLs

The TMLLs should at all points lie under the frontier of hot leg saturation and DNB, whichever is more limiting. The actual frontier of hot leg saturation and DNB will be determined at a statistically adjusted 5/95 bound, with the DNB frontier determined with 95% confidence, based upon the upper 95/95 limit of the DNB correlation. If positive power margins between the TMLLs and the frontier of hot leg saturation/DNB exists, then the TMLLs are verified.

Table 5.1 Uncertainties Applied in Setpoint Verifications

Uncertainty Parameter	Value[a]
Integrated Radial Peaking Factor (F_r) Measurement	6.0% (one sided)
Total Peaking Factor (F_Q) Measurement	6.2% (one-sided)
Axial Shape Index (ASI)[b]	
LSSS Measurement	± 4.98%
LCO Measurement	± 6.59%
Measurement Bias (nonrandom)	0.01719 asiu
Inlet Temperature Measurement	± 2.0°F
Core Inlet Flow Rate Measurement	± 4.29%
Pressure Measurement	± 22.0 psi
HTP DNB Correlation	See footnote [c]
Engineering Allowance	± 3%

Table 5.2 Modified Parameters for the MUR Power Uprate Analyses

Parameter	Value
Rated Thermal Power[d]	1.7% uprate (1525.5 MWt)
Power Measurement Uncertainty	See Figure 5.1

[a] Unless otherwise noted the distributions are treated as normal, two-sided, and the uncertainty represents a 95% bound on the distribution (1.96σ).

[b] The LSSS ASI uncertainty is the uncertainty in the ASI signal at the point where it enters the TM/LP and APD calculators. The LCO ASI uncertainty is the uncertainty in the tilt meter readings (CB 4), as read in the control room. Two ASI nonrandom bias constituent terms contribute to the ASI uncertainty (I_P uncertainties arising from core physics codes and the incore-excore calibration process). These systematic measurement biases have been incorporated directly into the barn itself.

[c] See Reference 5 for a description of the HTP correlation and its associated uncertainties. The upper 95/95 limit on the correlation is biased upward by a deterministically applied mixed core penalty.

[d] The uprated thermal power supported in the setpoint calculations, 1525.5 MWt, is insignificantly different than the 1526 MWt supported in the mechanical design calculations. This difference will have a negligible effect on the calculated margins.

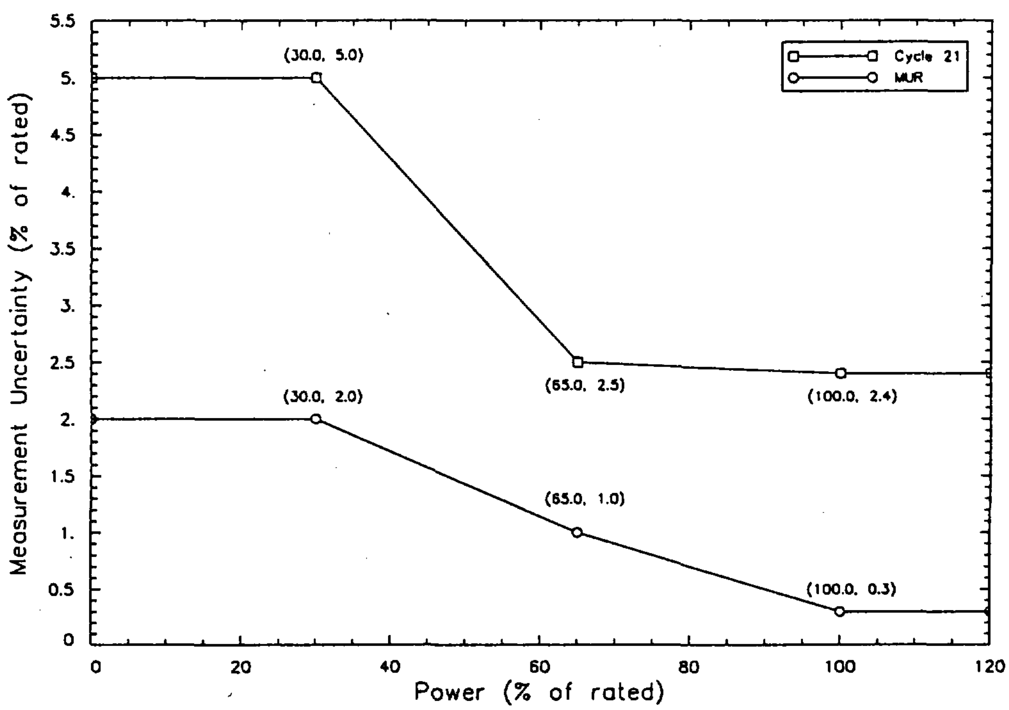

Figure 5.1 Power Measurement Uncertainty

5.3 *Limiting Safety System Settings*

Verification calculations were performed on both the TM/LP and APD LSSSs. A description of the former calculation is described in Section 5.3.1, while the latter calculation is summarized in Section 5.3.2. The purpose of the verification calculations was to assess the impact of anticipated plant changes for the MUR power uprate upon existing margins to the LSSS functions protecting fuel SAFDLs.

Both LSSSs were verified to protect against their limiting AOOs for any set of conditions expected based upon Cycle 21 plant and neutronics data, a 1.7% power uprate from the Cycle 21 rated thermal power of 1500 MWt, and a reduced power uncertainty as shown in Figure 5.1.

5.3.1 Verification of the TM/LP (DNB) LSSS

The TM/LP LSSS, alternatively known as the DNB LSSS, is designed to protect the DNB SAFDL with at least a 95% probability at a 95% confidence level. Additionally, the TM/LP should preclude the occurrence of hot leg saturation (bulk boiling), at a 95% probability. The TM/LP LSSS accomplishes this by monitoring cold leg RTD temperature, pressurizer pressure, synthesized internal ASI, and auctioneered power, then converting them into a floating trip pressure below which the system pressure cannot fall without initiating a reactor trip.

The TM/LP LSSS calculates this floating trip pressure based upon the auctioneered maximum of the calculated variable trip pressure P_{VAR} and a fixed floor pressure:

$$P_{trip} = MAX(P_{VAR}, P_{floor})$$

where P_{VAR} is calculated based upon B, the auctioneered maximum of the nuclear and ΔT-power signals the ASI-adjustment function $A_1(Y)$, the power-adjustment function B·PF(B), the monitored cold leg RTD temperature, and the trip coefficient potentiometer settings. The variable trip pressure P_{VAR} is given (per Section 3.0 of Reference 7) as:

$$\begin{aligned} P_{VAR} &= \alpha \cdot A_1(Y) \cdot B \cdot PF(B) + \beta \cdot T_{in} + \gamma \\ &= 29.6 \cdot A_1(Y) \cdot QR1 + 20.63 \cdot T_{in} - 12372 \end{aligned}$$

where α, β, and γ are TM/LP trip coefficients, $A_1(Y)$ is a function adjusting the trip pressure in response to axial power distribution, and QR1 (or B·PF(B)) is a function adjusting the effective trip power in response to control bank position and auctioneered maximum power B.

The floor pressure P_{floor} is a fixed reference pressure set to a minimum of 1750 psia per Technical Specification 1.3(4). If the monitored system pressure falls below the auctioneered maximum of the floating trip pressure or the floor pressure, a TM/LP trip is signaled.

Three limiting AOOs form the basis for the verification of the TM/LP trip: the RCS Depressurization Event, the Sequential CEA Withdrawal at Power, and the Excess Load Increase. Since the TM/LP is an uncompensated trip, dynamic measurement deviation effects arise in all of the monitored inputs and are accounted for in the trip response. The Cycle 21 TM/LP LSSS verification analysis demonstrated that the limiting trip basis AOO was the Excess Load Increase. Uprated conditions are not anticipated to have any significance in shifting the limiting trip basis AOO. Therefore, only a single verification was performed, using transient biases corresponding to the Excess Load Increase event.

5.3.1.1 TM/LP LSSS Configuration

The TM/LP configuration used in the MUR analysis is identical to that used in the Cycle 21 verification. The functional form of the PVAR variable trip pressure was discussed in Section 5.3.1. The QR1 and $A_1(Y)$ functions are shown in Figures 5.2 and 5.3. The Technical Specification/COLR TM/LP LSSS settings were used for the verification; no attempt to credit plant setting biases was made.

The TM/LP verification methodology credits the actions of the APD LSSS, which will serve to trip the plant in case power and ASI exceed the maximum allowable ASI-dependent power described by the APD LSSS barn. If the APD LSSS is predicted to intercede with a probability of 95% or greater, then the case is rejected as not being primarily protected by the TM/LP LSSS. The configuration of the APD LSSS used in the verification is summarized in Section 5.3.2.1.

5.3.1.2 *TM/LP Verification*

The methodology used to verify the TM/LP LSSS is summarized in Sections 2.2.1 and 2.2.2 of Reference 4. A single verification was performed, corresponding to the limiting TM/LP trip basis AOO (Excess Load Increase). A conservative set of transient shifts were applied to the trip to account for event-specific trip delays. Table 5.3 summarizes the transient shifts used in the TM/LP verification. Table 5.4 provides parameters and uncertainties that are used in the TM/LP verification in addition to the general parameters from Table 5.1.

Axials corresponding to power levels of 60% and above and CEA insertions to ARO, LTIL, PDIL, and PDIL-ΔP positions were considered. The PDIL-ΔP (sub-PDIL) axial shapes are used to bound potential transient situations where a mismatch may arise between the power-dependent PDIL insertion and the actual power of the plant. These axials were generated for the Cycle 21 setpoint verification analyses, and will remain representative for uprated conditions.

A disposition of the Chapter 14 analyses of record was conducted by OPPD, and it was concluded that the Excess Load Increase (as well as the Uncontrolled CEA Withdrawal and RCS Depressurization) analyses of record remain valid for post-MUR uprated conditions (Reference 9). As such, the transient biases used as a basis for the Cycle 21 TM/LP LSSS verification remain applicable to the MUR verification analysis.

The verification of the TM/LP LSSS demonstrated that the trip conservatively protects against DNB for the Excess Load Increase, with a 95% probability at a 95% confidence, and by a substantial amount of margin. Because the Excess Load Increase remains more limiting with respect to the TM/LP than either the Uncontrolled CEA Withdrawal Incident or the Reactor Coolant System Depressurization Incident, the TM/LP is implicitly verified for the other two events as well.

The TM/LP LSSS also precludes hot leg saturation for all three events, at a 95% probability.

Since the TM/LP is characterized by positive pressure margin to the occurrence of both hot leg saturation and DNB, both at a 95% probability (with DNB protection provided at a 95% confidence level), the TM/LP LSSS settings in Fort Calhoun Station are applicable to the MUR, based upon the analysis conditions and assumptions.

Table 5.3 Transient Shifts Applied in the TM/LP LSSS Calculations[a]

TM/LP Input Parameter	Excess Load Increase
Auctioneered Power	5.94%
Pressure	0.23 psi
Cold Leg Temperature	3.17°F
Hot Leg Temperature	0.11°F

Table 5.4 Additional Parameters Applied in TM/LP LSSS Verification

TM/LP Parameter	Value[b]
Thermal power calculator coefficients	
K_α	$1.483\ \frac{\%\ \text{power}}{°\text{F}}$
K_β	$2.824 \times 10^{-3}\ \frac{\%\ \text{power}}{°\text{F}^2}$
K_γ	$2.866 \times 10^{-3}\ \frac{\%\ \text{power}}{°\text{F}^2}$
TM/LP trip uncertainty	± 70.74 psi
Axial Shapes	Complete set of Cycle 21 setpoint axials used. Bounds all possible insertions from ARO to sub-PDIL positions.

[a] The transient shifts are generically defined as the difference between the indicated value of the monitored input at the time a trip setpoint is reached, and the actual value at the time of the MDNBR. The sign convention on the transient shift is such that a positive shift results in a penalty to DNB, with the exception of the cold leg temperature shift, where the convention is switched.

[b] Unless otherwise noted the distributions are treated as normal, random, two-sided, and the uncertainty represents a 95% bound on the distribution (1.96σ).

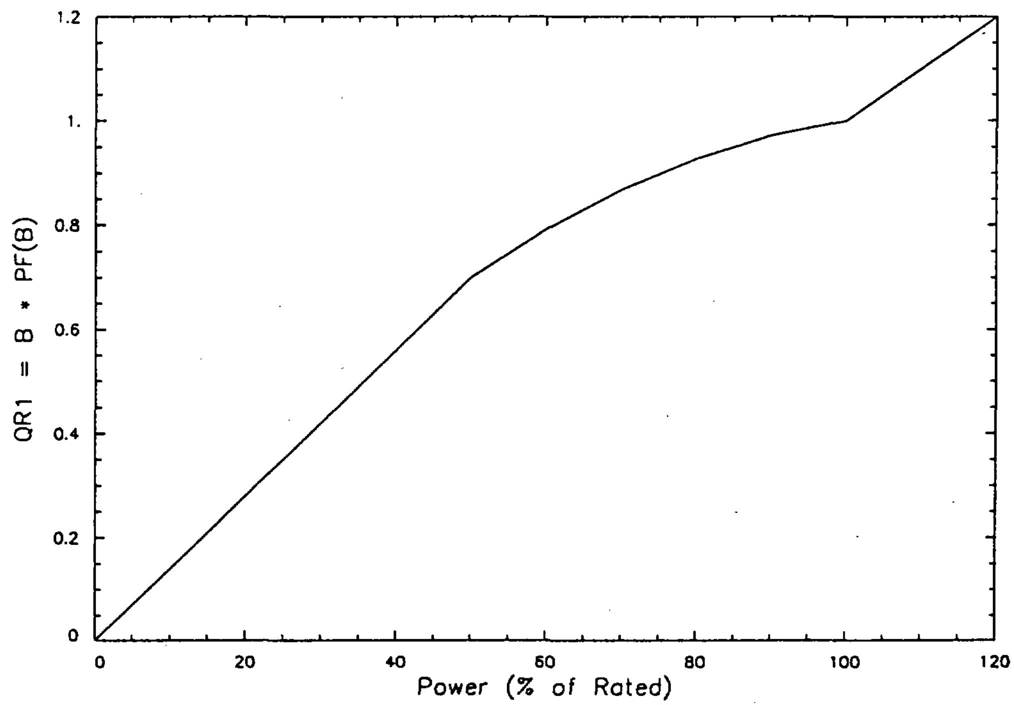

Figure 5.2 TM/LP QR_1 Function ($B \cdot PF(B)$)

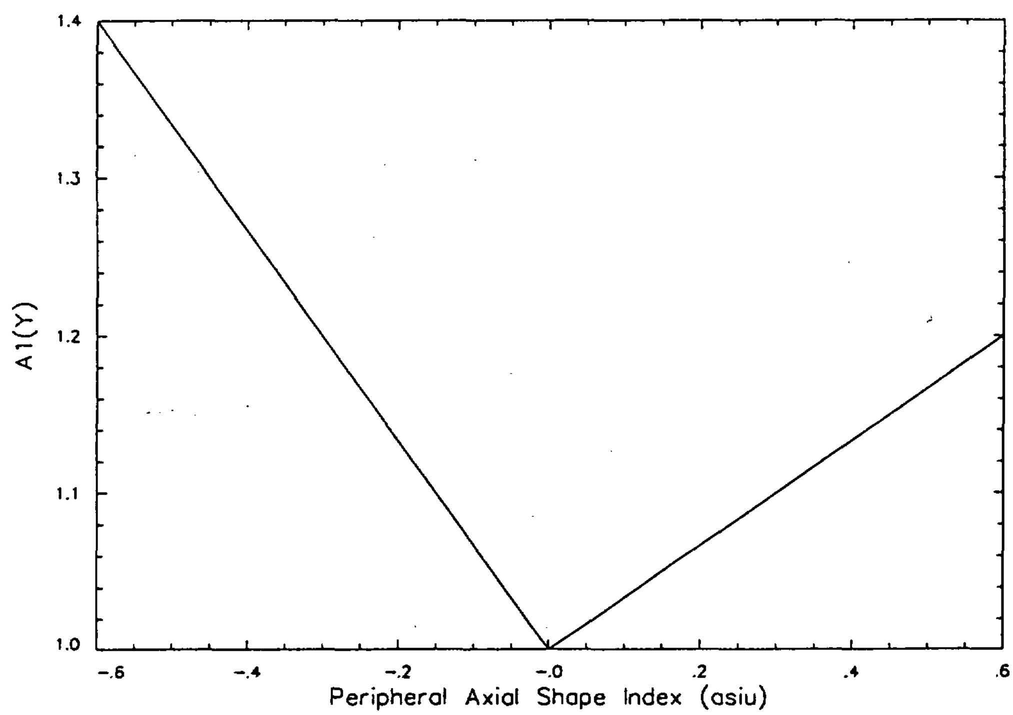

Figure 5.3 TM/LP $A_1(Y)$ Function

5.3.2 Verification of the APD LSSS

The APD LSSS trip, in conjunction with the VHPT, the Rod Block System, and the radial peaking LCO, protects the FCM SAFDL against axial power maldistributions during normal operation and AOOs. The APD LSSS verification calculation ensures that, for any axial power shape that can be achieved during the cycle, the maximum transient LHR does not exceed the limit given the current configuration of the APD LSSS trip function.

The APD LSSS trip may also intercede to protect the DNB SAFDL. Therefore its actions are credited in the verification of the TM/LP LSSS (as discussed in Section 5.3.1).

The APD LSSS synthesizes an internal ASI, based on measurements by the excore nuclear detectors. This internal ASI signal, common to the TM/LP LSSS and other LSSS and LCO functions, is then compared against a maximum allowable ASI for the indicated auctioneered power level. If the maximum allowable ASI as a function of power is exceeded by the internal ASI signal, a reactor trip is generated. The maximum allowable power function is expressed in terms of a shape in internal ASI/auctioneered power space, and is frequently referred to as a "barn", or "tent". The barn is designed to protect the FCM SAFDL with at least a 95% probability throughout the cycle.

The APD LSSS will potentially protect against a variety of transients, particularly those which produce axial power redistributions. Of these, the most limiting AOO transients (taking into account transient measurement effects) are the Uncontrolled CEA Withdrawal Incident and the Excess Load Increase events. As with the TM/LP LSSS, the APD LSSS is an uncompensated trip and dynamic measurement biases are explicitly accounted for in the setpoint confirmation.

5.3.2.1 APD LSSS Configuration

The APD LSSS allowed power versus peripheral ASI "barn" was revised for Cycle 21 to provide more operating flexibility. This "barn" was used as a basis for the MUR analysis. The plant settings for the APD LSSS barn is depicted in Figure 5.4. The peak (deposited) LHR the APD LSSS protects against is 22.0 kW/ft, per Specification 1.3(8) of Reference 6.

5.3.2.2 APD LSSS Verification

The methodology used to verify the APD LSSS is summarized in Sections 2.1.1 and 2.1.2 of Reference 4. A disposition of the Chapter 14 analyses of record was conducted by OPPD, and it was concluded that the Excess Load Increase (as well as the Uncontrolled CEA Withdrawal) analyses of record remain valid for post-MUR uprated conditions (Reference 9). As such, the transient biases used as a basis for the Cycle 21 APD LSSS verification remain applicable to the MUR verification analysis.

Uprated conditions are not anticipated to have any significance in shifting the limiting trip basis AOO. The Cycle 21 APD LSSS verification demonstrated that the limiting basis AOO event for the APD LSSS trip was the Excess Load Increase. Therefore, a single verification was performed based upon this AOO. The other APD LSSS basis event (Uncontrolled CEA Withdrawal Incident) will continue to be less limiting than the Excess Load Increase. A conservative set of transient shifts corresponding to the deterministic transient analyses were applied to the trip to account for event-specific overshoots and decalibration. Table 5.5 summarizes the transient shifts applicable to the APD LSSS verification. Table 5.6 provides parameters and uncertainties that are used in the APD LSSS verification in addition to the general plant parameters documented in Table 5.1.

The APD LSSS verification methodology credits the actions of the VHTP as part of a case rejection criterion. An overall 15% power offset on the VHPT (10% nominal offset, deterministically adjusted upward by 5% uncertainty) was supported in the APD LSSS verification.

Axials corresponding to power levels of 60% and above and CEA insertions to ARO, LTIL, PDIL, and PDIL-ΔP positions were considered. The PDIL-ΔP (sub-PDIL) axial shapes are used to bound potential transient situations where a mismatch may arise between the power-dependent PDIL insertion and the actual power of the plant. These axials were generated for the Cycle 21 setpoint verification analyses, and will remain representative for uprated conditions.

The verification of the APD LSSS demonstrates that the trip conservatively protects against FCM for the Excess Load Increase event, with a 95% probability. Because the Excess Load Increase remains more limiting with respect to the APD LSSS then the Uncontrolled CEA Withdrawal, the APD LSSS is implicitly verified for the latter event.

Since the APD LSSS is characterized by positive power margin to the 22.0 kW/ft FCM limit as described by the APD LSSS barn, both at a 95% probability, the APD LSSS shown in Figure 5.4 is verified for the analysis conditions and assumptions.

Table 5.5 Transient Shifts Applied in the APD LSSS Calculations

Parameter	Excess Load Increase
APD Power Bias[a]	3.50%
APD Trip Decalibration[b]	9.60%

Table 5.6 Additional APD LSSS Verification Parameters

Parameter	Value
Axial Shapes	Complete set of setpoint axials used. Bounds all possible insertions from ARO to sub-PDIL positions.

[a] The transient bias is the difference between the maximum calculated power in the event and the calculated power at the time the trip setpoint is reached.

[b] The APD trip decalibration is defined as the uncertainty associated with the APD power measurement bias. Effectively this accounts for all power measurement uncertainty, including calibration and drift allowances, instrument reference accuracy and uncertainty, and M&TE uncertainties.

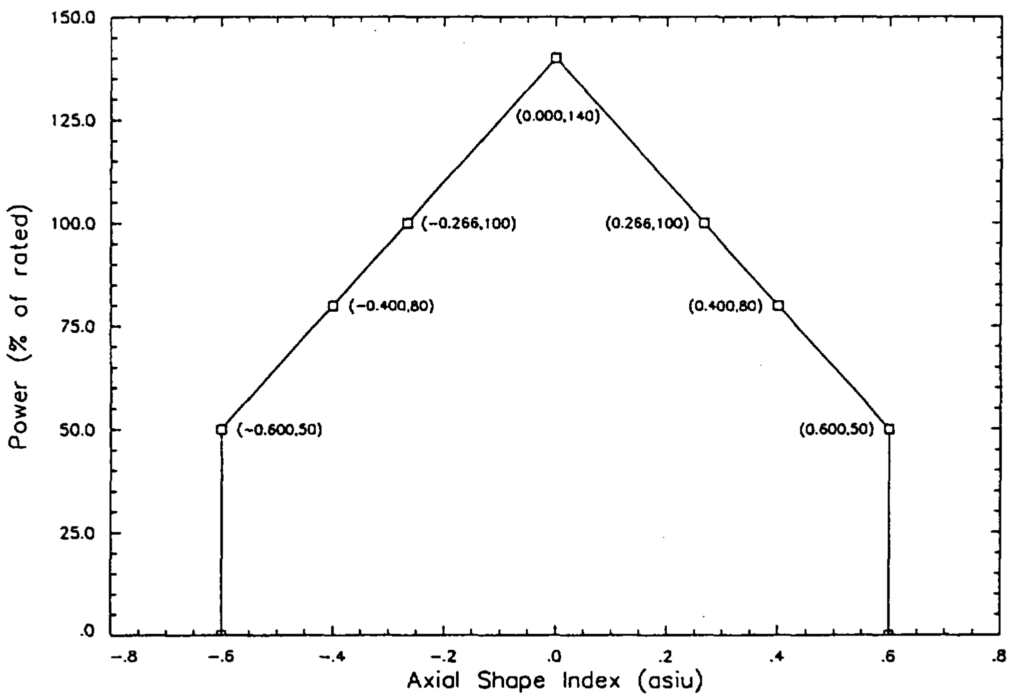

Figure 5.4 APD LSSS "Barn" (Plant Settings)

5.4 *Limiting Conditions for Operation*

The monitoring LCOs on DNB and LHR are verified for each cycle of operation. These LCOs are the DNB LCO and the Excore LHR LCO.

Both LCOs were verified to protect against their limiting AOOs for any set of conditions expected based upon Cycle 21 plant and neutronics data, a 1.7% power uprate from the Cycle 21 rated thermal power of 1500 MWt, and a reduced power uncertainty as shown in Figure 5.1.

The DNB LCO settings were demonstrated to protect against the occurrence of DNB, at a 95/95 level, for the analysis conditions and assumptions. In addition, the LHR LCO was shown to support the Technical Specification LOCA LHR limit.

5.4.1 DNB/BASSS LCO

The DNB LCO is designed to protect the fuel DNB SAFDL for all AOO transients where either initial operational margin or a combination of initial operational margin and RPS protective functions are required. Typically the limiting transient in the latter category is the LOCF Incident, where the combination of initial operational margin and the low coolant flow rate trip are required to protect the DNB SAFDL. In the case of Fort Calhoun Station, where the RCPs have large flywheels with a large moment of inertia, the flow coastdown is relatively slow compared to most plants and the LOCF is a non-challenging event. The limiting transient in the former category is the CEAD Incident. The DNB LCO is typically verified for both events.

For the purposes of the MUR setpoint analyses, only the verification for the CEAD is conducted. The substantiation for this is that the LOCF verification was demonstrated in Cycle 21 as being considerably less limiting than the CEAD verification. Uprated conditions will not result in a shift with respect to the limiting event.

5.4.1.1 DNB/BASSS LCO Configuration

The DNB LCO barn is shown in Figure 5.5. The barn breakpoints coincide with those of the BASSS LCO. Therefore the verification of the DNB LCO will also implicitly verify the BASSS LCO barn, if one conservatively considers the BASSS LCO as responding to excore power signals, rather than incore.

Table 5.1 contains a summary of the uncertainties used in the DNB LCO.

5.4.1.2 DNB/BASSS LCO Verification for CEAD Event

A verification analysis of the DNB LCO barn for the CEAD event was conducted. Table 5.7 contains a summary of the boundary conditions used for the analysis. A standard verification approach was used, with the variables contributing to DNB power treated statistically, the ASI and power variables treated statistically, and the CEAD transient boundary conditions treated in a deterministic fashion.

Axials corresponding to power levels of 60% and above and CEA insertions to ARO, LTIL, PDIL, and PDIL-ΔP positions were considered. The PDIL-ΔP (sub-PDIL) axial shapes are used to bound potential transient situations where a mismatch may arise between the power-dependent PDIL insertion and the actual power of the plant. These axials were generated for the Cycle 21 setpoint verification analyses, and will remain representative for uprated conditions.

A disposition of the Chapter 14 transient analyses of record was conducted by OPPD, and it was concluded that the CEAD analysis of record remains valid for post-MUR uprated conditions (Reference 9). As such, the boundary conditions used as a basis for the Cycle 21 DNB LCO CEAD setpoint calculation remain applicable to the MUR analysis.

Positive power margin exists at all points between the maximum allowed power and the statistically adjusted DNB power at any point on the DNB LCO barn. Thus, the DNB LCO shown in Figure 5.5 protects the DNB SAFDL for the CEAD event for MUR power uprate subject to the analysis conditions and assumptions.

Table 5.7 DNB LCO CEAD Parameters

Parameter	Value
Steady-State Final Power	101.55%[a]
Steady State Final Pressure	2069.9 psia
Steady State Final Inlet Temperature	543.1°F
Radial Peaking Augmentation (CEAD)	1.165
Axial Shapes	Complete set of setpoint axials used. Bounds all possible insertions from ARO to sub-PDIL positions.

[a] The power level used to determine the DNB LCO margin for the CEAD event was conservatively assumed to be the initial power level of 102% of RTP.

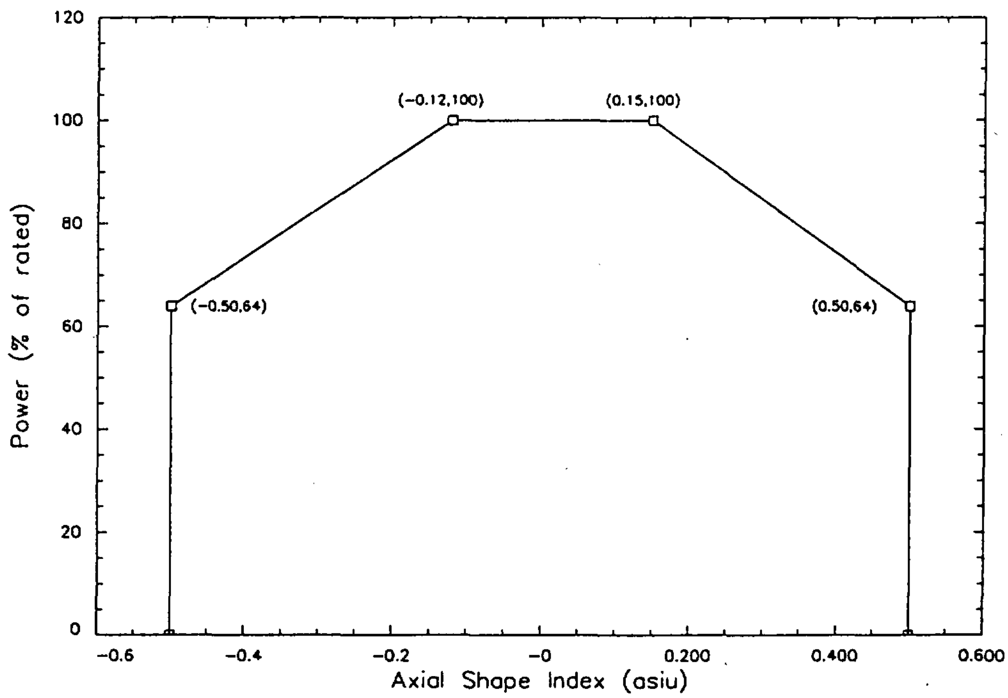

Figure 5.5 DNB/BASSS LCO "Barn" (Plant Settings)

5.4.1.3 DNB/BASSS LCO Verification for LOCF Event

The Cycle 21 DNB LCO LOCF verification demonstrated a minimum power margin significantly greater than the minimum Cycle 21 DNB LCO CEAD power margin (see Sections 8.1.1 and 8.1.2 of Reference 8). None of the MUR power uprate changes are anticipated to change the limiting event from the CEAD to the LOCF. Therefore, only the DNB LCO CEAD event was analyzed for the MUR, and the LOCF event is dispositioned as being less limiting.

5.4.2 Excore LHR Monitoring LCO

The Excore LHR LCO is designed to preclude the maximum LHR from exceeding the LOCA LHR limit when the plant is monitoring LHR with the excore detectors rather than the incore monitoring system. The Excore LHR LCO is a more restrictive mode of operation for the plant due to the increased measurement uncertainties associated with excore monitoring. As with the APD LSSS and the other LCOs, the Excore LHR Monitoring LCO is described by a barn in power-ASI space. The Excore LHR LCO is also intrinsically a steady-state limit, rather than one imposed to intercede in, or protect against, transient situations.

5.4.2.1 Excore LHR LCO Configuration

The Excore LHR LCO barn was modified for Cycle 21 to provide more operating flexibility, and will be used as a basis for the MUR verification. This barn is reproduced in Figure 5.6.

5.4.2.2 Excore LHR LCO Verification

The statistical methodology for the verification of the Excore LHR LCO is essentially the same as that for APD LSSS, except for the revised LHR limit and that the uncertainties associated with the power to meet the limit do not include transient-based uncertainties, biases, or delays. The parameters and uncertainties associated with the verification of the Excore LHR LCO are summarized in Tables 5.1, 5.2, and 5.8.

Due to the requirements laid out in Specification 2.10.4(1)(c) of Reference 6, the full-length CEAs must be withdrawn beyond the LTILs when continuously monitoring via the excore detectors; therefore only ARO and LTIL axial inputs are used in the verification of the Excore LHR LCO. These axials were generated for the Cycle 21 setpoint verification analyses, and will remain representative for uprated conditions.

Positive power margin exists at all points analyzed. Therefore, the Excore LHR LCO settings in Fort Calhoun Station are applicable to the MUR, based upon the analysis conditions and assumptions.

Table 5.8 LHR LCO Parameters and Uncertainties

Uncertainty Parameter	Value
Axial Shapes	ARO and LTIL axial shapes only

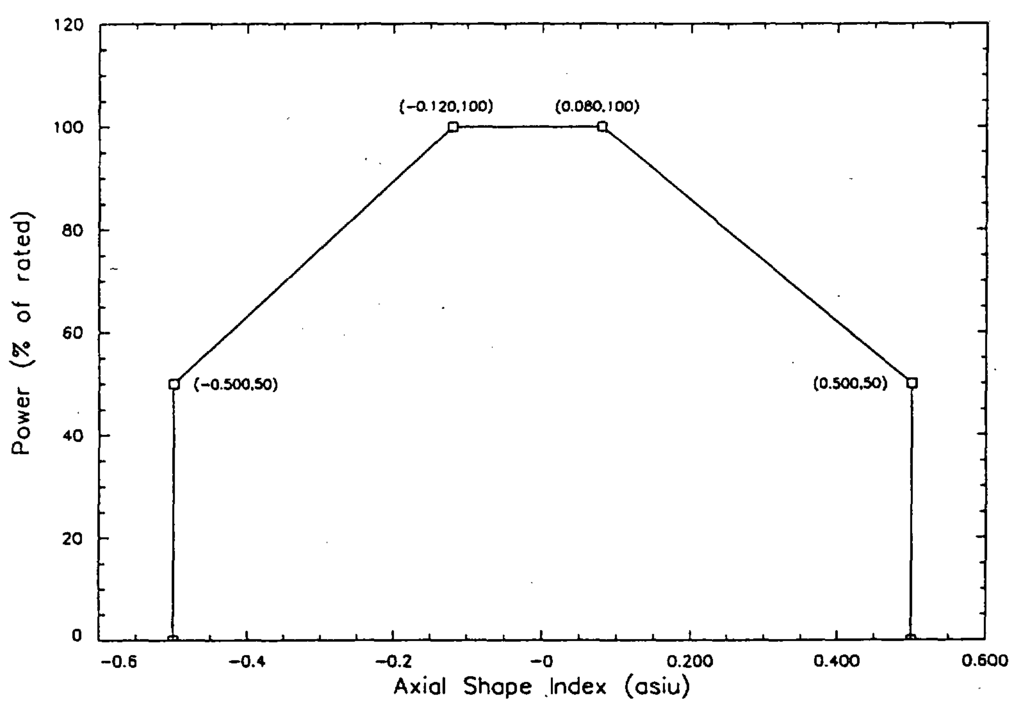

Figure 5.6 Excore LHR LCO "Barn" (Plant Settings)

5.5 *Safety Limits (Thermal Margin Limit Lines)*

5.5.1 Verification of the TMLLs

For changes that may significantly modify the DNB or hot leg saturation performance of the plant, FANP will generally reevaluate the adequacy of the DNB and hot leg saturation safety limits in the Technical Specifications. These limits, designated "Thermal Margin/Low Pressure Safety Limits" or "Thermal Margin Limit Lines", are graphically depicted in Figure 1-1 of Reference 6. They define isobaric frontiers of DNB or hot leg saturation (whichever is more limiting) at a lower 95% bound, in terms of core power and inlet temperature. The slanted portion of the isobars represents the region where DNB is more limiting than hot leg saturation. The cutoff at 580°F represents the region where the action of the MSSVs precludes either hot leg saturation or DNB from occurring.

Although superficially similar to the TM/LP LSSS, the TMLLs are not an LSSS or LCO. The verification of the TMLLs and the TM/LP LSSS within FANP setpoint methodology is distinct, with each being verified relative to their proximity to DNB or hot leg saturation, rather than their proximity to each other. Whereas the TM/LP LSSS is verified using a range of cycle-specific limiting shapes as a function of ASI, the TMLLs are verified using a singular, conservative, cycle-independent axial shape.

The TMLLs were reevaluated as part of the MUR power uprate, in order to assess the changes in margins due to the rated power for power uncertainty tradeoff, as well as other changes related to the mixed core configuration and plant changes since Cycle 20. The TMLLs were last verified prior to Cycle 20.

5.5.1.1 TMLL Configuration

The analysis values of the TMLLs for Fort Calhoun Station are shown in Figure 5.7. These limits are adjusted to add a very slight slope to the 580°F inlet temperature cutoff[a]. Additionally, because the saturation verification is conducted down to 25% RTP, the TMLLs were extended backwards to 0% RTP. Figure 5.8 shows the "design axial" used as a basis for the verification analysis[b].

---

[a] A non-zero slope is required in the methodological implementation to avoid numerical troubles with the underlying verification codes.

[b] A single axial shape forms the basis for the TMLLs. A very conservative top-peaked axial was generated in Cycle 20 such that it would bound the DNB performance of any cycle-specific axial.

5.5.1.2 TMLL Verification

The TMLL verification analysis for the MUR power uprate supports the plant uncertainties documented in Table 5.1, the MUR parameters in Table 5.2, and the additional TMLL-specific parameters in Table 5.9. Because the original basis for the plant TMLLs could not be determined, it was conservatively assumed that the calculated power margins between hot leg saturation/DNB and the TMLLs should be penalized with a statistical penalty resulting from plant uncertainties.

Since positive power margin exists between the TMLLs and the occurrence of DNB or hot leg saturation, the existing TMLLs in Figure 1-1 of the Technical Specifications continue to conservatively represent the frontiers of hot leg saturation and DNB for Fort Calhoun Station MUR, subject to the analysis conditions and assumptions.

Table 5.9 Additional Parameters for the TMLL Verification

Parameter	Value
Design Axial Shape	See Figure 5.8
Inlet Temperature Control Deadband	±2°F

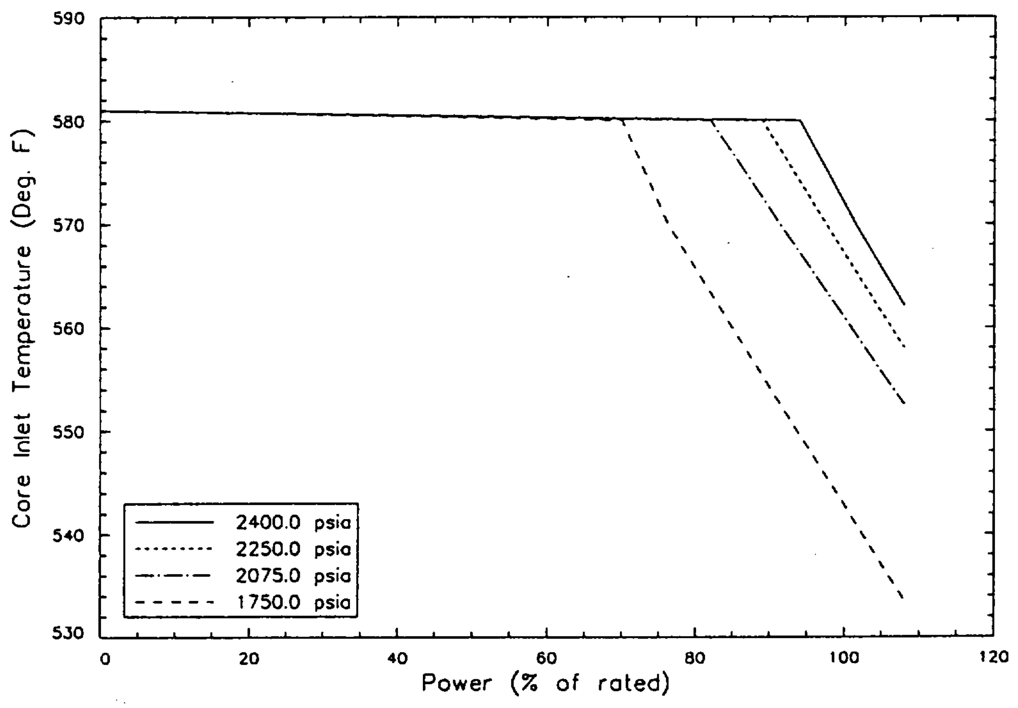

Figure 5.7 Thermal Margin/Low Pressure Limit Lines

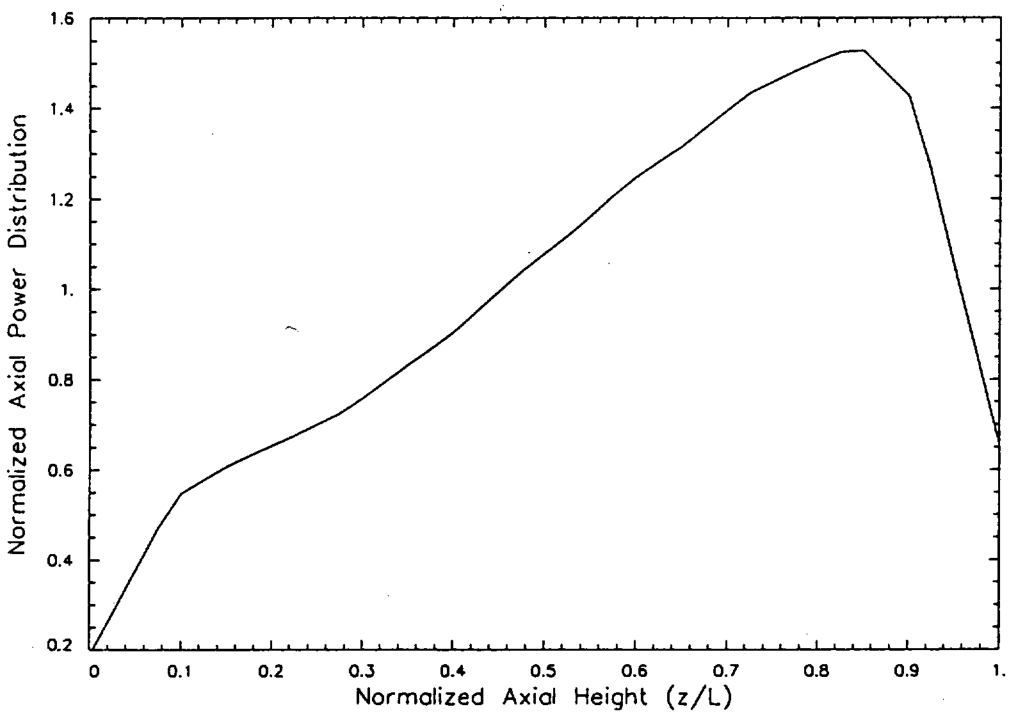

Figure 5.8 Thermal Margin/Low Pressure Limit Line Design Axial

5.6 *Trip Coefficient Settings*

The analyses documented herein are designed to assess the shifts in a margins for the existing LSSS and LCO functional settings from Cycle 21. Therefore, none of the information documented in Section 9 of Reference 8 have been invalidated as a consequence of these analyses, with the following exception.

The K_α coefficient setting in the thermal power calculator was rebalanced for the MUR power uprate, based upon the Cycle 21 settings for K_β and K_γ ($2.824 \times 10^{-3}\ \frac{\%\ \text{power}}{°F^2}$ and $2.866 \times 10^{-3}\ \frac{\%\ \text{power}}{°F^2}$, respectively). The effective value changed from 1.509 % power/°F to 1.483 % power/°F. The latter value is a suggested initial value for the post-uprate plant startup.[a]

[a] A ΔT-power calibration procedure will determine the actual plant setting value of K_α at startup.

6.0 References

1. XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, and 5, *Qualification of Exxon Nuclear Fuel for Extended Burnup*, Exxon Nuclear Company, October 1986.

2. ANF-88-133(P)(A) and Supplement 1, *Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU*, Advanced Nuclear Fuels Corporation, December 1991.

3. EMF-92-116(P)(A) Revision 0, *Generic Mechanical Design Criteria for PWR Fuel Designs*, Siemens Power Corporation, February 1999.

4. EMF-1961(P)(A) Revision 0, *Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors*, Siemens Power Corporation, July 2000.

5. EMF-92-153(P)(A) and Supplement 1, *HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel*, Siemens Power Corporation, March 1994.

6. Fort Calhoun Nuclear Station Facility Operating License and Technical Specifications, through Amendment 213.

7. Fort Calhoun Station Technical Data Book Procedure TDB-VI, Revision 27, Core Operating Limits Report.

8. EMF-2752 Revision 1, *Fort Calhoun Cycle 21 Statistical Verification of LSSS and LCO Setpoints*, Framatome ANP Richland, Inc., May 2002.

9. Letter, T. A. Heng (OPPD) to J. L. Raklios (FANP), "Disposition of USAR Chapter 14 Events for MUR Power Uprate," NPD-03-016, January 24, 2003.

Distribution

K. M. Duggan, 31
R. C. Gottula, 36
R. D. Kliewer, OF12
T. R. Lindquist, 36
T. P. Luedeke, OF12
E. M. Miller, OF11/ OPPD (3)
M. H. Smith, 31
G. S. Uyeda, OF12

Document Control

LIC-03-0122 Attachment 7

Westinghouse Reactor Vessel Structural Evaluation

(No changes from LIC-03-0067 Attachment 7)

Westinghouse Reactor Vessel Structural Evaluation

Core Shroud Under Revised Thermal Loadings[7]

Introduction

In 1980/1981, the impact of the then-proposed Cycle 6 stretch power (1500 MWt) operation on the Reactor Vessel Internal (RVI) structures was assessed. At that time, it was concluded that, of the major RVI components, only the Core Shroud would potentially be adversely affected by the increased thermal loadings associated with such operation. Accordingly, a structural evaluation of the Core Shroud under these increased thermal loadings was performed. That evaluation (documented in Reference 2) determined the impact of stretch power operation on the Core Shroud to be acceptable.

In 1997, the RVI structures were again evaluated; this time to assess the impact of increased flow resulting from the removal of steam generator orifice plates. In that evaluation (documented in Reference 5), it was determined that the increased flow, and the attendant increase in pressure difference across the Core Shroud panels, would increase stresses in the most critically-stressed Core Shroud component; i.e., the anchor block bolts. These stresses were calculated in Reference 5 and were determined to be acceptable.

The currently-proposed Appendix K power uprate, like the Cycle 6 stretch power condition, will increase thermal loadings on the RVI structures. Because the power uprate is small (1.7%), it is assumed that the rationale developed for the stretch power assessment remains applicable, and that only the Core Shroud will incur potentially adverse effects. To quantify these effects, the Reference 12 calculation note reprises the Reference 2 evaluation; modified as necessary to optimize methodology and to incorporate revised thermal loadings associated with the Appendix K power uprate. The increased pressure differential across the Core Shroud panels, as evaluated in Reference 5, is also incorporated. High-temperature effects are considered, and scoping fatigue evaluations are performed for the re-evaluated components.

Limits of Applicability

The Reference 12 calculation note, as summarized in this report, is applicable to OPPD Fort Calhoun Nuclear Station only.

---

[7] **Note: evaluation was performed to a higher power level than what is requested in LAR to be conservative.**

Summary of Results and Conclusions

The results of Reference 2, which evaluated the Core Shroud for stretch power operation, are summarized below:

Component	Stress Category	Calculated Stress (psi)	Allowable Stress (psi)	Margin (%)
Core Shroud Panel @ 6th Girth Rib	Primary plus Secondary	18,203	45,300	60
Core Shroud Panel to Girth Rib Bolts	Secondary Shear	5,507	16,800	67
Core Shroud Panel to Anchor Block Bolts	Primary plus Secondary	15,154	16,800	10
Core Shroud Panel to Anchor Block Bolt Holes	Primary plus Secondary Shear	22,910	45,300	49
Girth Rib Flexure	Secondary	19,631	45,300	57
Girth Rib Flexure To CSB Bolts	Secondary Shear	2,162	16,800	87

In Reference 5, which evaluated the Core Shroud for increased flow resulting from the removal of steam generator orifice plates, the above results were modified as shown below.

Component	Stress Category	Calculated Stress (psi)	Allowable Stress (psi)	Margin (%)
Core Shroud Panel to Anchor Block Bolts	Primary plus Secondary	21,400	23,700[8]	10

Per Reference 12, the above results remain applicable for Appendix K power uprate, except as modified below:

Component	Stress Category	Calculated Stress (psi)	Allowable Stress (psi)	Margin (%)
Core Shroud Panel @ 1st Girth Rib[9]	Primary	15,242	22,200	31
	Primary plus Secondary	31,541	44,400	29
Core Shroud Panel @ 6th Girth Rib[10]	Primary	4,580	22,099	79
	Primary plus Secondary	14,197	44,400	68
Girth Rib Flexure[11]	Secondary	39,981	44,400	10
Girth Rib Flexure To CSB Bolts	Secondary Shear	5,136	16,500	69

Note that the primary plus secondary stress in the Core Shroud Panel at the elevation of the 6th Girth Rib, as calculated in Reference 12, is lower than that calculated in Reference 2. This is because Ref. 2 conservatively calculates primary stress using the maximum ΔP (which occurs at

[8] Allowable stress = as-irradiated yield stress rather than 3 × Sm for bolting material, as used in Ref. 2.
[9] Fatigue usage = .01 < 1.0 allowable fatigue usage
[10] Fatigue usage = .217 < 1.0 allowable fatigue usage
[11] Fatigue usage = .800 < 1.0 allowable fatigue usage

the bottom of the panel), whereas Ref. 12 appropriately uses the (lower) ΔP at the elevation of the 6^{th} Girth Rib.

Assumptions and Open Items

Discussion of Major Assumptions

- Because the Appendix K power uprate is relatively small (1.7%), it is assumed that any adverse effects on the RVI structures resulting from this power uprate will be confined to the Core Shroud, which is more sensitive than the other RVI components to minor variations in thermal loading. More significant thermal loading increases, such as would result from a larger power uprate, could adversely affect additional RVI components and would have to be evaluated accordingly.
- OBE loads were not included among the Core Shroud design loads defined in Reference 12, and are assumed to be negligible.

Open Items

There are no open items associated with the Reference 12 calculation note, as summarized in this report.

Acceptance Criteria

Primary stress limits for RVI structures are defined in Table 3.2-1 of the Fort Calhoun Station Updated Safety Analysis Report (Ref. 3), and include a limiting value of 1.5 S_m on primary membrane (general or local) plus bending stress under design loading plus design earthquake conditions. Reference 3 does not provide specific design criteria for secondary stresses in RVI structures, but does indicate (in Section 3.2.3.4) an intent to satisfy the design criteria defined in Section III, Article 4 of the ASME Boiler and Pressure Vessel Code. Article 4 relates to the design of Class A pressure vessels, and was included in earlier editions of the ASME Code, prior to the adoption of specific design criteria for RVI structures. Per Section 4.2.4 of Reference 3, the Fort Calhoun reactor vessel was designed to the requirements of the 1965 edition of the ASME Code, Section III, Article 4, through and including the 1967 Winter Addenda (Reference 4). The design criteria specified therein for primary plus secondary stress were invoked for the Reference 2 evaluation and are defined below:

Paragraph N-414.4 (Reference 4)

Primary plus secondary stress intensity is the stress intensity derived from the highest value at any point across the thickness of a section of the general or local primary membrane stresses plus primary bending stresses plus secondary stresses produced by specified operating pressure and other specified mechanical loads and by general thermal effects. The effects of gross structural discontinuities but not of local structural discontinuities (stress concentrations) shall be included. The allowable value of this stress intensity is 3 S_m, where S_m is the design stress intensity for the material.

Reference 4 specifies the following additional criteria for peak stress intensity. These were not considered in Reference 2, but are addressed in Reference 12.

Paragraph N-414.5 (Reference 4)

Peak stress intensity is the stress intensity derived from the highest value at any point across the thickness of a section of the combination of all primary, secondary and peak stresses produced by specified operating pressures and other mechanical loads and by general and local thermal effects including the effects of gross and local structural discontinuities. The allowable value of this stress intensity is dependent on the range of the stress difference from which it is derived and on the number of times it is to be applied. The allowable value is obtained by the methods of analysis for cyclic operation described in N-415 through the use of the fatigue curves, Figs. N-415(A) and (B).

Per Subsection N-415, the ratio of the applied number of cycles over the allowable number of cycles (obtained from the fatigue curves), summed for each transient event or combination of events, shall be ≤ 1.0.

Design criteria for nuclear vessels in high temperature service, concurrent with the ASME Code edition of record (Reference 4), were provided in Reference 13. As defined therein, the allowable value of primary membrane plus primary bending stress intensity shall be:

$$1.5 \cdot S_m - \left(\frac{T - T_c}{200}\right) \cdot (0.17 \cdot S_m + 0.33 \cdot S)$$

for $T_c \leq T \leq T_c + 200$

Where: S_m = Design stress intensity @ 800 °F
T = Maximum metal temperature
T_c = 800 °F for austenitic steel
S = Calculated primary membrane plus primary bending stress intensity

The allowable value of primary plus secondary stress intensity shall be the greater of 3 S_m or three times the allowable amplitude of fatigue stress at 10^6 cycles. Revised fatigue curves are provided (in Figure 2 of Reference 13) for temperatures greater than 800 °F.

These high temperature design criteria were not considered in Reference 2, but are addressed in this calculation note.

Bolting design criteria defined in Reference 4 were also invoked for the Reference 2 evaluation; these are defined as follows:

Paragraph N-416.1 (Reference 4)

The maximum value of service stress, averaged across the bolt cross section and ignoring stress concentrations, shall not exceed 2 S_{mb}, where S_{mb} is the design stress intensity for the bolting material. The maximum value of service stress at the periphery of the bolt cross section (resulting from direct tension plus bending) and neglecting stress concentrations shall not exceed 3 S_{mb}. Stress intensity, rather than maximum stress, shall be limited to this value when the bolts are tightened by methods other than heaters, stretchers, or other means which minimize residual torsion.

With the formal introduction (via Reference 9) of specific design criteria for RVI structures, the bolts used to assemble RVI components were re-classified as threaded structural fasteners. Design criteria for threaded structural fasteners are based on material strength values for the non-bolt equivalent of the bolt material, and are less stringent than the bolting design criteria described above, which are based on the much lower material strength values for the bolt material itself. The invocation of bolting design criteria in References 2 and 12 therefore constitutes a conservative measure.

Method Discussion

Reference 2, documenting the evaluation of the Core Shroud for Cycle 6 stretch power operation, calculates stresses for the following Core Shroud components/locations:

a) Core Shroud panels (at locations adjacent to girth ribs)
b) Core Shroud panel-to-girth rib attachment bolts
c) Core Shroud panel-to-anchor block attachment bolts
d) Core Shroud panel-to-anchor block attachment bolt holes
e) Girth rib flexure (longer flexure on straight segment assembly girth ribs only)
f) Girth rib flexure-to-Core Support Barrel attachment bolts

Thermal stresses in the above components were calculated (in Reference 2) using temperature input data provided in References 6 and 7. The applicability of this temperature input data to the proposed Appendix K power uprate condition is summarized in Reference 8. Based on a review of the Reference 2 methodology in combination with the Reference 8 assessment, the applicability of the Reference 2 results may be summarized as follows:

a) Core Shroud panels – Temperature input data remains applicable, however a re-evaluation was performed to include an additional panel elevation and to calculate peak stresses with attendant fatigue usage. High temperature effects were also considered.
b) Core Shroud panel-to-girth rib attachment bolts – Temperature input data, and the associated thermal stresses, remain applicable.
c) Core Shroud panel-to-anchor block attachment bolts – Temperature input data, and the associated thermal stresses, remain applicable.
d) Core Shroud panel-to-anchor block attachment bolt holes – Temperature input data, and the associated thermal stresses, remain applicable.
e) Girth rib flexure – Temperature input data was revised per Reference 8. A re-evaluation was performed to incorporate this revised data and to include the shorter, more highly-stressed flexure on the girth ribs attached to the corner segment assemblies. A fatigue evaluation was included.
f) Girth rib flexure-to-Core Support Barrel attachment bolts – A re-evaluation was performed to incorporate revised loads calculated per item d above. A fatigue evaluation was included.

Reference 5, documenting the evaluation of the Core Shroud for increased flow resulting from the removal of steam generator orifice plates, re-calculated stresses in the Core Shroud panel-to-anchor block attachment bolts (item c above) to account for the increased pressure differential across the Core Shroud panels. These re-calculated stresses remain applicable to the Appendix K power uprate condition. The increased pressure differential used in Reference 5 was also used to re-calculate primary stresses in the Core Shroud panels.

Material properties used in Reference 12 are applicable to the Appendix K power uprate condition, per Reference 10.

References

1) "Nuclear Services Policies & Procedures," WP-4.5 Revision 4, "Design Analysis," effective 10/01/01.
2) Calculation Number 23866-690-008 Rev. 0, Omaha Stretch Power Study: Core Shroud Thermal Stress Analysis (1560 MWT), M. M. Cepkauskas, 5/29/81.
3) Fort Calhoun Updated Safety Analysis Report, Release 4, 5/30/02.
4) ASME Boiler and Pressure Vessel Code, Section III, Article 4, 1965 Edition through and including the 1967 Winter Addenda.
5) Calculation Number O-ME-C-016 Rev. 00, "Evaluation of the Reactor Internals Components Under Increased Flow from Removal of Steam Generator Orifice Plates", R. F. Raymond, 11/26/97.
6) Interoffice Correspondence Number O-TH-170, "Thermal Analysis of the OPPD Lower Core Shroud at 1560 MWT with Revised Heat Generation Rates", L. C. Hwang, 4/28/81.
7) Calculation Number 23866-TH-149, "Omaha Core Shroud & Core Support Barrel Thermal Analysis for Cycle 6 Stretch Power (1560 MWT)", W. R. Moran, 12/6/79.
8) Calculation Note Number CN-PS-03-9 Rev. 00, "Normal Operating Design Metal Temperatures for the Core Shroud for Ft. Calhoun for an Appendix-K Uprate (1526 MWt Power Level)", R. P. Letendre, 6/10/03.
9) ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, 1971 Edition, Winter 1973 Addendum.
10) Letter Number LTR-CI-03-22, "Reactor Vessel Internals Materials Evaluation for Fort Calhoun Appendix-K Power Uprate, J. F. Hall, 3/24/03.
11) Design Criteria Number 23866-6.10a, "Core Shroud and Formers", 11/21/67.
12) Calculation Note Number CN-CI-03-27 Rev. 00, "Evaluation of Core Shroud under Revised Thermal Loadings Associated with Appendix K Power Uprate", P. O'Brien, 6/12/03.
13) Interpretations of ASME Boiler and Pressure Vessel Code, Case 1331-4 (Special Ruling), "Nuclear Vessels in High Temperature Service", Approved by Council August 15, 1967.

LIC-03-0122 Attachment 8

Facility Operating License, TS, and TS bases pages marked up to show the proposed changes

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not to exceed ~~1500~~ 1524 megawatts thermal (rated power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 220, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Physical Security Plan," with revisions submitted through September 30, 1988; "Fort Calhoun Station Guard Training and Qualification Plan," with revisions submitted through August 17, 1979; and "Fort Calhoun Station Safeguards Contingency Plan," with revisions submitted through March 20, 1979. If certain security modifications are delayed beyond expectations of the schedule, approved compensatory measures must be implemented during the transition period.

D. Fire Protection Program

Omaha Public Power District shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report for the facility and as approved in the SERs dated February 14, and August 23, 1978, November 17, 1980, April 8, and August 12, 1982, July 3, and November 5, 1985, July 1, 1986, December 20, 1988, November 14, 1990, March 17, 1993 and January 14, 1994, subject to the following provision:

> Omaha Public Power District may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

TECHNICAL SPECIFICATIONS

DEFINITIONS

The following terms are defined for uniform interpretation of these Specifications.

REACTOR OPERATING CONDITIONS

Rated Power

1524
A steady state reactor core output of ~~1500~~ MWt.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition (Operating Mode 1)

The reactor is in the power operation condition when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is 6.606 x 10^6 lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at ~~1500 MWt~~[3] RATED POWER to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At ~~the power of 1500 MWt~~, RATED POWER sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.[4] These analyses are based on a minimum of four-of-five operable main steam safety valves on each main steam header.

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 in^2.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References

(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Tests (Continued)

Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed at RATED POWER on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, ~~a reactor power level of 1500 MWt~~, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident.[3] The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.

LIC-03-0122 Attachment 9

Revised (clean) Facility Operating License, TS, and TS bases pages

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not to exceed 1524 megawatts thermal (rated power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Physical Security Plan," with revisions submitted through September 30, 1988; "Fort Calhoun Station Guard Training and Qualification Plan," with revisions submitted through August 17, 1979; and "Fort Calhoun Station Safeguards Contingency Plan," with revisions submitted through March 20, 1979. If certain security modifications are delayed beyond expectations of the schedule, approved compensatory measures must be implemented during the transition period.

D. Fire Protection Program

Omaha Public Power District shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report for the facility and as approved in the SERs dated February 14, and August 23, 1978, November 17, 1980, April 8, and August 12, 1982, July 3, and November 5, 1985, July 1, 1986, December 20, 1988, November 14, 1990, March 17, 1993 and January 14, 1994, subject to the following provision:

> Omaha Public Power District may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Amendment No. ~~50,70,118,155,~~
~~158,160,181,184, 199~~

TECHNICAL SPECIFICATIONS

DEFINITIONS

The following terms are defined for uniform interpretation of these Specifications.

REACTOR OPERATING CONDITIONS

Rated Power

A steady state reactor core output of 1524 MWt.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition (Operating Mode 1)

The reactor is in the power operation condition when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is 6.606×10^6 lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at RATED POWER[(3)] to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At RATED POWER, sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.[(4)] These analyses are based on a minimum of four-of-five operable main steam safety valves on each main steam header.

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 in^2.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References
(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

3.0 SURVEILLANCE REQUIREMENTS
3.5 Containment Tests (Continued)

Basis

The containment is designed for an accident pressure of 60 psig.[(2)] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, at RATED POWER, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident.[(3)] The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.

LIC-03-0122 Attachment 10

List of Regulatory Commitments

(No changes from LIC-03-0067 Attachment 10)

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by OPPD in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	Due Date/Event
1. Modifications associated with the MUR power uprate (Attachment 1, 3.0) will be completed prior to implementation. (This includes implementation of control room alarm functions.) (Attachment 2, VII.1)	1. Prior to MUR power uprate implementation.
2. Figure 2-1 of the Technical Specifications will be revised prior to the reactor vessel reaching 39.9 EFPYs of operation or adjusted when the NRC approves the FCS license amendment request for pressure and temperature limits report approval (Attachment 2, IV.1.1.2)	2. Prior to reactor vessel reaching 39.9 EFPYs of operation.
3. Both relief valves associated with feedwater heaters FW-16A, B will be replaced in the next refueling outage (Attachment 2, VI.2.6)	3. During 2003 RFO.