Omaha Public Power District

Fort Calhoun Station
P.O. Box 550, Highway 75
Fort Calhoun, NE 68023-0550

LIC-03-0067
July 18, 2003

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

REFERENCE: Docket No. 50-285

SUBJECT: Fort Calhoun Station Unit No. 1
License Amendment Request (LAR)
Measurement Uncertainty Recapture Power Uprate

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby requests the following amendment to the Fort Calhoun Station Unit No. 1 (FCS) Operating License and Technical Specifications:

> Revise Paragraph 3.A. in Operating License (OL) DPR-40 to authorize operation at a steady state reactor core power level not in excess of 1525 megawatts thermal (MWt).
>
> Revise the definition of RATED POWER in Technical Specification (TS) to reflect the increase from 1500 MWt to 1525 MWt.

Corresponding TS Bases changes are also requested:

> In the Basis to TS 2.1.6, pages 2-15a and 2-16, Pressurizer and Main Steam Safety Valves, change all instances of "1500 MWt" to "RATED POWER."
>
> In the Basis to TS 3.5, page 3-51, replace "a reactor power level of 1500 MWt," with "RATED POWER."

The proposed license amendment will increase licensed power level to 1525 MWt, or 1.67% greater than the current level of 1500 MWt. The requested increase in licensed rated power is the result of a measurement uncertainty recapture (MUR) power uprate. The information provided in support of this request is based on Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications", January 31, 2002.

The OPPD request is based on reduced uncertainty in the reactor thermal output measurement achieved by installation of a Westinghouse, LLC CROSSFLOW ultrasonic flow measurement system (CROSSFLOW system) and feedwater temperature resistance temperature detectors (RTDs). The reduced power measurement uncertainty allows for a power uprate that is equivalent to the 10 CFR 50, Appendix K criteria of 2% minus the bounded CROSSFLOW based power uncertainty of 0.33%. The NRC approved CENPD-397-P-A for referencing in power uprate license applications in a safety evaluation dated March 20, 2000.

A001

OPPD has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent released, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the categorical exclusion requirements of 10 CFR 51.22(c) (9) and an environmental impact appraisal need not be prepared.

OPPD requests approval of this proposed amendment by January 16, 2004. Upon NRC approval of this proposed change, OPPD requests that the amendment be effective on the date of issuance, but allow an implementation period of 60 days to provide sufficient time for associated administrative activities. The approval date was selected based on an approximate 6-month NRC review period. It should be noted that the plant does not require this amendment to allow continued safe, full power operation.

Pursuant to 10 CFR 2.790, OPPD requests that the proprietary information presented and discussed in Attachment 3 be withheld from public disclosure. This information concerning the methodology used in the calorimetric uncertainty evaluation is proprietary to PL Integrated Resources LLC, as justified in the supporting affidavit (Attachment 5). Although the methodology is considered proprietary, the evaluation was done under exclusive contract to OPPD, using the procedures and reviews covered by the OPPD Quality Assurance Program. Attachment 3 is the calorimetric uncertainty evaluation with proprietary information enclosed in brackets. Attachment 4 is the non-proprietary version of Attachment 3 with the bracketed information deleted.

This request is supported by the attachments summarized in the following table:

Attachment	Content Description
1	A description and assessment of the MUR power uprate including: description, background, proposed OL and TS changes, technical assessment, a no significant hazards consideration and environmental considerations.
2	Summary of the MUR power uprate evaluation following guidance provided in Regulatory Issue Summary 2002-03.
3	Calorimetric Uncertainty Evaluation Proprietary Version
4	Calorimetric Uncertainty Evaluation Non Proprietary Version
5	Affidavit for Calorimetric Uncertainty Evaluation
6	Non-Proprietary Framatome Evaluation
7	Reactor Internals Structural Evaluation
8	OL, TS, and TS bases pages marked up to show the proposed changes.
9	Revised (clean) OL, TS, and TS bases pages.
10	List of regulatory commitments associated with this proposed amendment.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated state of Nebraska official.

I declare under penalty of perjury under the laws of the United States of America that I am authorized by Omaha Public Power District to make this request and that the foregoing is true and correct.

If you have any questions or require information, please contact Mr. Tom Matthews at 402-533-6938.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

Attachments: See table above

c: Thomas P. Gwynn, Acting NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska
Winston & Strawn

LIC-03-0067 Attachment 1
Description of Change, Safety Evaluation, Significant Hazards Determination, and Statement of Environmental Considerations

1.0 Introduction

Omaha Public Power District (OPPD) proposes to amend the Facility Operating License (OL) DPR-40 and Technical Specifications (TS) to increase licensed rated power level for Fort Calhoun Station Unit No. 1 (FCS). FCS is currently licensed to operate at a maximum rated power of 1500 megawatts thermal (MWt). Approval is being requested to increase the licensed core rated power by 1.67% to 1525 MWt. This power increase will be accomplished by using a more accurate main feedwater flow and temperature measurement system to calculate the reactor thermal output of the unit. Increasing rated power by reducing measurement uncertainty is called a measurement uncertainty recapture (MUR) power uprate. OPPD has evaluated the impact of a 1.67% uprate to 1525 MWt for the applicable systems, structures, components, and safety analyses at FCS. The results of this evaluation and the new main feedwater flow measurement system are described in Attachment 2 of this letter, "Summary of Measurement Uncertainty Recapture Power Uprate Evaluation Following Guidance Provided in NRC Regulatory Issue Summary (RIS) 2002-03," (Reference 10.1).

2.0 Description of License and Technical Specification Changes

The proposed license amendment will revise the FCS OL and the TS to increase the licensed rated power by 1.67% from 1500 MWt to 1525 MWt. The proposed changes are described in detail below and are also indicated on the marked up and clean copy Operating License and TS pages in Attachments 8 and 9.

2.1 Revise paragraph 3.A of the operating license, DPR-40, to authorize operation at reactor core power levels not in excess of 1525 MWt.
2.2 Revise TS 1.0, RATED POWER, to reflect the increase from 1500 MWt to 1525 MWt.

Corresponding TS Bases changes are also requested:

2.3 In the Basis to TS 2.1.6, pages 2-15a and 2-16, Pressurizer and Main Steam Safety Valves, change all instances of "1500 MWt" to "RATED POWER."
2.4 In the Basis to TS 3.5, page 3-51, replace "a reactor power level of 1500 MWt," with "RATED POWER."

3.0 Background

The 1.67% power uprate for FCS is based on eliminating unnecessary analytical margin that is assumed in analyses to account for the measurement uncertainties associated with the calorimetric calculations. FCS current accident and transient analyses include a minimum 2% margin on rated power to account for power measurement uncertainty. This power measurement uncertainty was originally required by Title 10 of the Code of Federal Regulations,

Part 50 (10 CFR 50), Appendix K, "ECCS Evaluation Models." The rule required a 2% power margin between the licensed power level and the power level assumed for the emergency core cooling system (ECCS) evaluations. In June 2000, the NRC amended 10 CFR 50, Appendix K to provide licensees the option of maintaining the 2% power margin or applying a reduced margin. For the latter case, the new assumed power level had to account for measurement uncertainties in the power level measurement instrumentation. The revised Appendix K rule had an effective date of July 31, 2000.

Uncertainty in the main feedwater flow measurement is one of the most significant contributors to power measurement uncertainty. Based on this fact and on the above Appendix K rule change, OPPD proposes a reduced power measurement uncertainty of 0.33% and an increase in rated power of 1.67%. To accomplish this reduction in uncertainty and increase in power, OPPD will install a CROSSFLOW ultrasonic flow measurement system (CROSSFLOW system) for measuring the main feedwater flow at FCS. The CROSSFLOW system provides a more accurate measurement of feedwater flow than that assumed during the development of the original Appendix K requirements and that of the feedwater flow venturis currently used to calculate reactor power. The CROSSFLOW system will measure feedwater mass flow to within ± 0.3% for FCS. This bounding feedwater mass flow uncertainty was used to establish a bounding total power measurement uncertainty of ± 0.33%. Installation of new feedwater temperature RTD's provides more accurate temperature measurement than that assumed in the development of original Appendix K requirements. Based on this, FCS proposes to reduce the power measurement uncertainty required by Appendix K to 0.33%. The improved power measurement uncertainty obviates the need for the 2% power margin originally required by Appendix K, thereby allowing an increase in the rated power available for electrical generation by 1.67%.

In addition to the proposal to increase the rated power to 1525 MWt, OPPD also proposes continued use of the topical reports identified in the OPPD proposal to implement a Core Operating Limits Report (COLR) at FCS (Letter LIC-02-0109, dated October 8, 2002). The topical reports describe the NRC-approved analytical methodologies used to determine the core operating limits for FCS. This includes the small and large break loss of coolant accidents. In some of these topical reports, reference is made to the use of a 2% power measurement uncertainty being applied consistent with 10 CFR 50, Appendix K. OPPD requests that these topical reports be approved for use consistent with this MUR power uprate request (i.e., 0.33% power measurement uncertainty be assumed instead of 2%). The proposed change was described in section 2.0 of this attachment. Additionally, the reduction of the power measurement uncertainty does not constitute a significant change as defined in10 CFR 50.46 (a) (3) (I) regarding ECCS evaluation models.

3.1 Licensing Methodologies for Uprate

The proposed FCS MUR power uprate is consistent with topical report CENPD-397-P-A, "Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology." The NRC has approved this topical report for referencing in MUR power uprate submittals. OPPD is specifically applying this topical report, and the criteria listed in the NRC SER for the CENPD-397-P-A, for a requested 1.67% rated power increase.

In addition to the above methodology, OPPD has taken into account the specific guidance developed by the NRC for the content of MUR power uprate applications. This guidance was published on January 31, 2002, as NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," (Reference 10.1). Attachment 2 of this application provides an evaluation of the proposed MUR power uprate structured to be consistent with the NRC guidance. The NRC requests for additional information (RAI) for other licensee MUR power uprate requests were also reviewed and answers for applicable RAIs have been incorporated into the text of Attachment 2.

3.2 Licensing Approach to Plant Safety, Component and System Analyses

The reactor core power and the NSSS thermal power are used as inputs to most plant safety, component and system analyses. Generally, the FCS MUR power uprate analyses were evaluated as such:

- For safety analyses the power level was bounded at 1530MWt.[1]
- For component analyses, reviews were conducted to verify original design basis limits were still applicable.
- For systems analyses, reviews were conducted for overall system performance to uprate conditions. Some re-analyses were performed to ensure that parameters would be bound at the new power level.

No new analytical techniques have been used to support this power uprate request.

3.3 Conclusion

OPPD is requesting a 1.67% increase in core rated thermal power for FCS from 1500 MWt to 1525 MWt. This power increase will be accomplished by using a more accurate main feedwater flow measurement system to calculate the reactor power. This higher accuracy measurement will be achieved with the use of a CROSSFLOW system. This license amendment request has taken into account industry and NRC accepted methodologies and guidelines for power uprates.

This License Amendment Request (LAR) is made pursuant to 10 CFR 50.90 to modify the OL and the TS requirements associated with rated thermal power and the use of the power measurement uncertainty in safety analyses.

4.0 Regulatory Requirements & Guidance

OPPD has evaluated the impact of the proposed power uprate on safety analyses, NSSS systems and components, and balance of plant (BOP) systems. Attachment 2 summarizes the results of the comprehensive engineering review performed to evaluate the increase in the licensed core rated power. Results of this evaluation are provided in a format consistent with the regulatory guidance provided in NRC RIS 2002-03 (reference 10.1). The results of OPPD's evaluation demonstrate that applicable acceptance criteria will continue to be met following the implementation of the proposed 1.67% MUR power uprate.

[1] Note: some safety analyses are evaluated at zero percent power for most limiting conditions.

5.0 Technical Analysis

Attachment 2 provides the detailed technical analysis for this technical specification change. Attachment 2 summarizes the results of the comprehensive engineering review performed to evaluate the increase in the licensed core rated thermal power. Results of this evaluation are provided in a format consistent with the regulatory guidance provided in NRC RIS 2002-03 (reference 10.1).

6.0 Regulatory Analysis

Based on the detailed considerations discussed in Attachment 2 and the No Significant Hazards Determination, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be harmful to the common defense and security or to the health and safety of the public.

7.0 No Significant Hazards Determination

OPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*

   Response: No.
   There are no changes as a result of the MUR power uprate to the design or operation of the plant that could affect system, component, or accident functions. All systems and components function as designed and the performance requirements have been evaluated and found to be acceptable.

   The reduction in power measurement uncertainty allows for safety analyses to continue to be used without modification. This is because the safety analyses *dependent on power level* were performed or evaluated at 102% of 1500 MWt (1530 MWt) or higher. Analyses at these power levels support a core power level of 1525 MWt with a measurement uncertainty of 0.33%. Radiological consequences of USAR Chapter 14 accidents were assessed previously using the alternate source term methodology (Reference 10.2). These analyses were performed at 102% of 1500 MWt (1530 MWt) and continue to be bounding. Updated Safety Analysis Report (USAR) Chapter 14 analyses and accident analyses continue to demonstrate compliance with the relevant accident analyses' acceptance criteria. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

   The primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) were evaluated at an uprated core power level of 1525 MWt and continue to

comply with their applicable structural limits. These analyses also demonstrate the components will continue to perform their intended design functions. Changing the heatup and cooldown curves is based on uprated fluence values. This does not have a significant effect on the reactor vessel integrity. Thus, there is no significant increase in the probability of a structural failure of the primary loop components. The LBB analysis conclusions remain valid and the breaks previously exempted from structural consideration remain unchanged.

All of the NSSS systems will continue to perform their intended design functions during normal and accident conditions. The auxiliary systems and components continue to comply with the applicable structural limits and will continue to perform their intended functions. The NSSS/BOP interface systems were evaluated at 1525 MWt and will continue to perform their intended design functions. Plant electrical equipment was also evaluated and will continue to perform their intended functions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*

   Response: No.
   No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the uprated power level. The proposed change has no adverse effects on any safety related systems or component and does not challenge the performance or integrity of any safety related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Involve a significant reduction in a margin of safety.*

   Response: No.
   Operation at 1525 MWt core power does not involve a significant reduction in the margin of safety. The current accident analyses have been previously performed with a 2% power measurement uncertainty or at uprated core powers that exceed the MUR uprated core power. System and component analyses have been completed at the MUR uprated core power conditions. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been both reviewed and approved by the NRC, or are currently under review (the proposed Pressure-Temperature Limits Report). Therefore, the proposed change does not involve a significant reduction in margin of safety.

Conclusion: Operation of FCS in accordance with the proposed amendment will not result in a significant increase in the probability or consequences of any accident previously analyzed; will

not result in a new or different kind of accident from any accident previously analyzed; and does not result in a significant reduction in a margin of safety.
Based on the above, OPPD concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

8.0 Environmental Consideration

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22 (c) (9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

In accordance with RIS 2002-03, the environmental considerations pertaining to this license amendment request are addressed in detail in Attachment 2, Section VII, "Environmental Review."

9.0 Precedent

Between October 10, 2002 and January 31, 2003, NRC Safety Evaluation Reports (SERs) were issued to the following stations for increased power level due to measurement uncertainty recapture:

1. Grand Gulf Nuclear Station, Issuance of Amendment, 1.7% Increase in Licensed Power Level (TAC No. MB3972), October 10, 2002.
2. H. B. Robinson Steam Electric Plant, UNIT NO. 2 (HBRSEP2) Issuance of Amendment Regarding a 1.7 Percent Power Uprate (TAC No. MB5106), November 5, 2002.
3. Peach Bottom Atomic Power Station, Units 2 and 3 Issuance of Amendment 1.62% Increase in Licensed Power Level (TAC Nos. MB5192 and MB5193), November 22, 2002.
4. Indian Point Nuclear Generating Unit No. 3 Issuance of Amendment 1.4 Percent Power Uprate (TAC No. MB5297), November 26, 2002.
5. Point Beach Nuclear Plant, Units 1 and 2 Issuance of Amendments Measurement Uncertainty Recapture Power Uprate (TAC Nos. MB4956 and MB4957), November 29, 2002.
6. Donald C. Cook Nuclear Plant, Unit 1 Issuance of Amendment 273 Regarding Measurement Uncertainty Recapture Power Uprate (TAC No. MB5498), December 20, 2002.
7. River Bend Station, Issuance of Amendment 1.7 Percent Increase in Licensed Power Level (TAC No. MB5094) January 31, 2003.

Additional SERs have been issued to stations for measurement uncertainty recapture prior to October 2002.

10.0 References

10.1 NRC Regulatory Issue Summary 2002-03: "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002.

10.2 NRC approved the FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221)".

LIC-03-0067 Attachment 2
Summary of the MUR Power Uprate Evaluation Following Guidance Provided in Regulatory Issue Summary 2002-03

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LIST OF ACRONYMS

AC	alternating current
AFW	auxiliary feedwater
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
AOV	air operated valve
AOP	abnormal operating procedure
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BD	blowdown
BOP	balance of plant
CASS	cast austenitic stainless steel
CCW	component cooling water
CEA	control element assembly
CEDM	control element drive mechanism
CEOG	Combustion Engineering Owner's Group
CFR	Code of Federal Regulations
CS	containment spray system
CST	condensate storage tank
CVCS	chemical and volume control system
CW	circulating water
DBA	design basis accident
DC	direct current
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DP	differential pressure
ECCS	emergency core cooling system
EDG	emergency diesel generator
EEQ	electrical equipment qualification
EFPY	effective full-power year
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EQ	environmental qualification
ESF	engineered safety feature
ETAP	electrical transient analyzer program
FAC	flow accelerated corrosion
FANP	Framatome-ANP
FCS	Fort Calhoun Station
FTC	Fort Calhoun
gpm	gallons per minute
HEI	Heat Exchanger Institute
HELB	high energy line break
HFP	hot-full power
hp	horsepower
HVAC	heating, ventilating, and air conditioning
HZP	hot-zero power
I&C	instrumentation and control

In-k	inches-kips
ISI	in-service inspection
IST	in-service testing
K	kips
LBB	leak-before-break
LBLOCA	large-break loss-of-coolant accident
LCO	limiting condition for operation
LHR	linear heat rate
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LSSS	limiting safety system setpoint
LTOP	low temperature overpressure protection
MCO	moisture carryover
MDAFW	motor-driven auxiliary feedwater
MOV	motor operated valve
MP	mechanical plug
MSIV	main steam isolation valve
MS	main steam
msec	milli-second
MSSV	main steam safety valve
MTC	moderator temperature coefficient
MUR	measurement uncertainty recapture
MVA	mega-volt-amperes
MVAR	mega-volt-amperes reactive
MW	megawatts
MWe	megawatts electric
MWt	megawatts thermal
NEI	Nuclear Energy Institute
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
ODSCC	outer diameter stress corrosion cracking
OL	Operating License
OPPD	Omaha Public Power District
PORV	power-operated relief valve
PC	plant computer
PED-MEI	Production Engineering Division Mechanical Engineering Instructions
PRA	probabilistic risk assessment
psi	pounds per square inch
psia	pounds per square inch - atmospheric
psig	pounds per square inch - gauge
P-T	pressure-temperature
PTLR	pressure-temperature limits report
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RCP	reactor coolant pump

RCS	reactor coolant system
RETS	radiological effluent technical specifications
RG	Regulatory Guide
RIS	Regulatory Issue Summary
RPS	reactor protection system
RSG	replacement steam generator
RTD	resistance temperature detector
RTP	rated thermal power
SBLOCA	small-break loss-of-coolant accident
SBO	station blackout
SC	shutdown cooling
SCC	stress corrosion cracking
SER	Safety Evaluation Report
SFPC	spent fuel pool cooling
SGIS	steam generator isolation signal
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SI	safety injection
SIRWT	safety injection refueling water tank
SIT	Safety Injection Tank
SRSS	square root of the sum of the squares
ST	Surveillance Test
TDAFW	turbine driven auxiliary feedwater
T/H	thermal and hydraulic
TS	Technical Specifications
TSP	tube support plates
UAT	unit auxiliary transformer
UFM	Ultrasonic flow measurement
USAR	Updated Safety Analysis Report
USAS	USA Standards Institute
USE	upper shelf energy
VCT	volume control tank
VWO	valve wide open

Summary of Measurement Uncertainty Recapture Evaluation
Following Guidance Provided In Regulatory Issue Summary 2002-03

Introduction

Omaha Public Power District (OPPD) proposes to amend the Operating License (OL) DPR-40 and the Technical Specification (TS) for Fort Calhoun Station (FCS). FCS is presently licensed for a core power rating of 1500 MWt (Section 3.A). Through the use of more accurate feedwater flow measurement instrumentation, approval is sought to increase the licensed core power by 1.67%, to 1525 MWt. The proposed 1.67% power uprate is based on eliminating unnecessary analytical margin originally required of ECCS evaluation models developed in accordance with the requirements set forth in 10 CFR 50, Appendix K "ECCS Evaluation Models."

In June 2000, the NRC approved a change to the 10 CFR 50, Appendix K, requirements to provide licensees with the option of maintaining the 2% power margin between the licensed core power level and the assumed core power level for ECCS evaluations, or apply a reduced margin to the ECCS evaluations. The proposed alternative to recapture margin for ECCS evaluation has been demonstrated to account for uncertainties due to a reduction in core power level measurement instrumentation error. OPPD will be installing Westinghouse CROSSFLOW instrumentation and with a calculated power measurement uncertainty of 0.33%. Based on the implementation of the CROSSFLOW system with improved feedwater temperature instrumentation and FCS specific power calorimetric uncertainties, OPPD proposes to reduce the licensed core power uncertainty required by 10 CFR 50, Appendix K, and evaluating the increased core power level by 1.67% using NRC-approved methodologies.

OPPD has evaluated the impact of the proposed power uprate on NSSS systems and components, BOP systems, safety analyses, and programs. The results of OPPD's analyses and evaluations, which demonstrate that applicable acceptance criteria will continue to be met, are summarized in this assessment. RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," (Reference 1) was used to establish the appropriate scope, structure, and level of detail presented in this assessment.

Approach for Increasing the Plant Power Level

The FCS MUR Uprate Program has been developed consistent with the format established in (Reference 2). This format has been successfully used as the basis for power uprate projects for several PWR units, including DC Cook Units 1 and 2.

The referenced approach establishes the general approach and criteria for uprate projects, including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents, and nuclear fuel, as well as the interfaces between the NSSS and BOP systems. The methodology includes the use of well-defined analysis input assumptions/parameter values, use of currently-approved analytical techniques, and use of currently-applicable licensing criteria and standards.

Overview of this Attachment

A comprehensive engineering review program consistent with Reference 1 has been performed for FCS to evaluate the increase in the licensed core power from 1500 MWt to 1525 MWt. Section I of this attachment describes the Westinghouse CROSSFLOW system that will be implemented to

provide more accurate feedwater flow measurement. Section II provides the results of the accident and transient analyses for which the existing analyses of record bound plant operation at the uprated power level. Section III summarizes those accidents and transient analyses that required re-analysis to produce analytical results that bound the uprated power level. Table 1, "System and Program Review Summary," summarizes the results of the evaluations that were performed on the FCS NSSS and BOP systems and components and plant programs. Table 2, "MUR Power Uprate Impact on FCS Accident/Transient Analyses," summarizes the accident and transient analyses for FCS, and documents whether or not each analysis of record bounds plant operation at the uprated power level proposed by the MUR Uprate Program. Table 2 also indicates where the summary of the evaluation/analysis is addressed in Section II of this attachment.

Sections IV and V of this attachment address the impact of the power uprate on the structural integrity of major plant components and on electrical equipment. Section VI addresses the effect of the power uprate on major plant systems and Section VII addresses the identification and evaluation of impacts on the control room and simulator, operator actions, modifications, procedures, the environment, sampling system and programs resulting from the 1.67% power uprate.The results of the analyses and evaluations addressed in Sections II through VII demonstrate that all acceptance criteria continue to be met.

Section VIII discusses the required changes to the FCS TS.

Table 1 System and Program Review Summary

System/ Component/ Program	Parameters with MUR Power Uprate Potential Impact	Components Impacted	Bounded By Existing Design/ Analyses?	REFERENCES (Report Section Number)
STEAM/POWER SYSTEMS				
Condensate	Flowrate (Increase) System Pressure (Increase) System Temperature (Increase)	Condensate Pumps	Yes	VI.2.2, VII.3
		Piping	Yes	VI.2.2, VII.3
		Valves and Miscellaneous Components	Yes	VI.2.2, VII.3
		Low Pressure Feedwater Heaters	Yes	VI.2.2, VII.3
Feedwater	Flowrate (Increase) System Pressure (None) System Temperature (Increase)	Feedwater Pumps	Yes	VI.2.2, VII.3
		Feedwater Regulating Valves	Yes	VI.2.2, VII.3
		Feedwater Isolation Valves	Yes	VI.2.2, VII.3
		High Pressure Feedwater Heaters	Yes	VI.2.2, VII.3
		Feedwater Check Valves	Yes	VI.2.2, VII.3
		Piping	Yes	VI.2.2, VII.3

Table 1 System and Program Review Summary				
System/ Component/ Program	**Parameters with MUR Power Uprate Potential Impact**	**Components Impacted**	**Bounded By Existing Design/ Analyses?**	**REFERENCES (Report Section Number)**
AFW System and EFWST	Required flow to steam generators when normal feedwater not available	Turbine Driven Auxiliary Feedwater Pump	Yes	VI.2.3, VII.3
		Motor Driven Auxiliary Feedwater Pumps	Yes	VI.2.3, VII.3
		Emergency Feedwater Storage Tank	Yes	VI.2.3, VII.3
		Diesel Driven Auxiliary Feedwater Pump	Yes	VI.2.3, VII.3
Main Steam	Steam Flow (Increase) System Pressure (Decrease)	Steam Dump	Yes	VI.2.1, VII.3, VII.4
Main Steam		Main Steam Safety Valves	Yes	VI.2.1.A, VII.3
		Atmospheric Dump Valve	Yes	VI.2.1.B, VII.3
		Bypass Valves	Yes	VI.2.1.D, VII.3
		Auxiliary Feedwater Pump Turbine	Yes	VI.2.2, VII.3
		Main Turbine	Yes	VI.2.1.1, VII.3
		Main Turbine Stop Valves	Yes	VI.2.1.C, VII.3
		Main Turbine Control Valves	Yes	VI.2.1.C, VII.3
	Disc Impact Energy	Main Steam Isolation Valves	Yes	VI.2.1.E, VII.3
Feedwater Heater Drains	Steam and Feedwater Flow (Increase) System Temperature (Increase)	Feedwater Heaters	Yes	VI.2.2, VII.3
		Feedwater Heater Drains	Yes	VI.2.4, VII.3
		Feedwater Heater Level Control Valves	Yes	VI.2.4, VII.3
		Heater Drain Pumps	Yes	VI.2.4, VII.3
		Feedwater Heater Shell Side Relief	No	VI.2.6
		Heater Drain Tank	Yes	VI.2.4, VII.3

Table 1 System and Program Review Summary				
System/ Component/ Program	**Parameters with MUR Power Uprate Potential Impact**	**Components Impacted**	**Bounded By Existing Design/ Analyses?**	**REFERENCES (Report Section Number)**
SG Blowdown	Flowrate	Piping	Yes	VI.2.7
COOLING/SUPPORT SYSTEMS				
CCW	Cooldown Flow to SC Heat Exchangers (Increase)	SC System	Yes	VI.3.1, VII.3
CW	Condenser Operating Pressure (Increase)	Main Condenser	Yes	VI.3.3, VII.3
	Outlet Temperature (increase)	Piping/Condenser	Yes	VI.3.3, VII.3
TPCWS	Heat Load (increase)	None	Yes	VI.3.2
RW	Heat Load (increase)	None	Yes	VI.3.4
SFPC	Spent Fuel Pit Decay Heatload (Increase)	SFPC Pumps and Heat Exchangers	No	III.1.1
Auxiliary Building Ventilation System	Heat load (increase)	None	Yes	VI.4
Containment Air Cooling System	Containment peak pressure (Bounded)	Containment Air Cooling	Yes	VI.3.5
Condensers	Steam Flow (increase)	Condensers	Yes	VI.2.5
Extraction Steam	Steam Flow (increase)	Piping	Yes	VI.2.6
ELECTRICAL SYSTEMS				
Turbine/ Generator	Generator Output (MVA Increase)	Generator	Yes	V.1
Isolated phase Bus	Main Generator Current (Increase)	Isolated phase Bus	Yes	V.3
Main Transformer	Transformer Output (MVA Increase)	Transformers	Yes	V.2
Switchyard	Switchyard Current (Increase)	Circuit Breakers	Yes	V
Offsite Power Feeders	Tie Line Current (Increase)	Tie Line (Current Rating)	Yes	V
Grid Stability	Output Power Level	Main Generator Impedance	Yes	V.7
EDGs	No Changes	No Changes	Yes	V.8
Electrical Distribution System	Bus Current Increase	4160V Bus, Breakers, Cables, Transformers	Yes	V.4, V.5, V.6, V.9, V.10, VII.4

Table 1 System and Program Review Summary				
System/ Component/ Program	**Parameters with MUR Power Uprate Potential Impact**	**Components Impacted**	**Bounded By Existing Design/ Analyses?**	**REFERENCES (Report Section Number)**
NSSS SYSTEMS				
NSSS Fluid Systems	Temperature (increase)	RCS Components	Yes	IV.1 – IV.6, VI.1
NSSS Auxiliary Systems (CVCS, SI, CS, SC)	Temperature (increase)	Various	Yes	VI.1, IV.7
NSSS/BOP INTERFACE SYSTEMS				
NSSS Control Systems	None	Valves, heaters	Yes	VI.5
LTOP System	None	None	Yes	VI.5
Reactor Vessel	Fluence, temperature (increase)	Pressure vessel	Yes	IV.1
Reactor Internals	Thermal hydraulic, Temperature (increase)	Reactor internals	Yes	IV.1.2
Piping and Supports	Temperature (increase)	Piping and supports	Yes	IV.2
Control Element Drive Mechanisms	Temperature (increase)	Housings, drive mechanisms	Yes	IV.3
RCPs and Motors	Temperature (increase), amps	Pumps and motors	Yes	IV.4
SGs	Thermal-hydraulic, stress	Steam generators	Yes	IV.5
Pressurizer	Stress, fatigue	Pressurizer	Yes	IV.6
NSSS Auxiliary Equipment	Temperature, fatigue	Various	Yes	IV.7
Fuel	None	Fuel	Yes	IV.8
Containment	Mass and Energy Release (increase)	Containment and protection	Yes	II.2, VI.1.9
SFPC System	Temperature (increase), cooling	Various	No	III.1.1
Instrument Air	None	None	Yes	VI.6
PROGRAMS				
EEQ	Temperature and Source Term	None	Yes	VII.6.1
MOVs	Temperature Increase	None	Yes	VII.6.2
AOVs	Flow Increase, pressure increase	None	Yes	VII.6.3
FAC	Increased Wear	Piping	Re-evaluated	VII.6.4

Table 1 System and Program Review Summary				
System/ Component/ Program	**Parameters with MUR Power Uprate Potential Impact**	**Components Impacted**	**Bounded By Existing Design/ Analyses?**	**REFERENCES (Report Section Number)**
			Wear Rates	
HELB	None	None	Yes	VII.6.5
SBO	Decay Heat	None	Yes	V.11
Fire Protection and Appendix R/Safe Shutdown	Decay Heat Load increase	None	No	III.2.1
ISI	None	None	Yes	VII.6.6
IST	None	None	Yes	VII.6.7
Individual and Occupational Radiation Exposure	Core Inventory	None	Yes	VII.6.9
Radiological Environmental Assessments	Core Inventory	None	Yes	VII.6.8
EOPs and AOPs	Decay Heat		Yes	VII.4
Coatings	None	None	Yes	VII.6.10
Alloy 600	Temperature (increase)	None	Yes	VII.6.11
SG	Temperature (increase)	None	Yes	VII.6.12
Containment Leak Rate	None	None	Yes	VII.6.13
Zinc Injection	None	None	Yes	VII.6.14
Sampling system	Heat Load (increase)	Chillers	Yes	VII.6.15

Evaluation Approach for the MUR Power Uprate Program

The licensed core power and/or NSSS thermal power are used as inputs to most plant safety, component, and system analyses. The current NSSS analyses of record for FCS model the core and/or NSSS thermal power in one of four ways. The approach taken for the proposed 1.67% power uprate for each of the four modeling approaches is provided below.

1. One FCS analysis assumed a nominal power level (MSLB). This analysis was evaluated for a 1.67% increased power level. Results of this evaluation demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.67% uprate conditions. Evaluation of this analysis bounds the MUR power uprate and is addressed in Section II.

2. A majority of FCS analyses already assume a core power level in excess of the proposed 1525 MWt. These analyses were performed at a higher power level (typically 1530 MWt) as part of prior plant programs. For these analyses, a portion of the available margin may be applied to offset the 1.67% uprate. Consequently, these analyses have been evaluated to confirm that sufficient analysis margin exists to envelop the 1.67% uprate. These analyses bound this MUR power uprate, and are addressed in Section II.

3. Some FCS analyses apply a 2% increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been revised for the 1.67% uprate conditions because the sum of increased core power level (1.67%) and the reduced power measurement uncertainty (0.33%) fall within the previously analyzed conditions. These analyses bound the MUR power uprate, and are addressed in Section II.

4. Some FCS analyses are performed at zero% power conditions, or do not model the core power level. Consequently, these analyses have not been re-performed, since they are unaffected by the core power level. These analyses bound this MUR power uprate, and are addressed in Section II.

Table 2, "MUR Power Uprate Impact on FCS Accident/Transient Analyses," summarizes the accident and transient analyses for FCS, and demonstrates that the existing analysis of record bounds plant operation at the proposed uprated power level. Details of these evaluations are provided in subsequent sub-sections.

Table 2 MUR Power Uprate Impact on FCS Accident/Transient Analyses

Accident/Transient	FCS USAR Section	Impact of Uprate on Current USAR Analysis	Section of this Document
LOCA and Related Analyses			
LOCA Forces	14.15.7.4	Bounded	II.2.1
Large Break LOCA	14.15.4	Bounded	II.1.17
Small Break LOCA	14.15.5	Bounded	II.1.18
Long Term Core Cooling	14.15.6	Bounded	II.1.19
Generation of Hydrogen in Containment	14.17	Bounded	II.1.22
Non-Limiting/Bounded Events			
CEA Withdrawal	14.2	Bounded	II.1.1
Boron Dilution	14.3	Bounded	II.1.2
CEA Drop	14.4	Bounded	II.1.3
Mal-Positioning of the Non-Trippable CEAs	14.5	Bounded	II.1.4
Loss of Coolant Flow Event	14.6.1	Bounded	II.1.5
Seized Rotor Event	14.6.2	Bounded	II.1.6
Idle Loop Startup	14.7	Bounded	II.1.7
Turbine Generator Overspeed Incident	14.8	Bounded	II.1.8
Loss of Load to Both Steam Generators	14.9.1	Bounded	II.1.9
Loss of Load to One Steam Generator	14.9.2	Bounded	II.1.10
Loss of Feedwater Flow	14.10.1	Bounded	II.1.11
Loss of Feedwater Heating	14.10.2	Bounded	II.1.12
Excess Load	14.11	Bounded	II.1.13
Main Steam Line Break Accident	14.12	Dispositioned	II.1.14
CEA Ejection	14.13	Bounded	II.1.15
Steam Generator Tube Rupture Accident	14.14	Bounded	II.1.16
Fuel Handling Accident	14.18	Bounded	II.1.23
Gas Decay Tank Rupture	14.19	Bounded	II.1.24
Waste Liquid Incident	14.20	Bounded	II.1.25
Reactor Coolant System Depressurization	14.22	Bounded	II.1.27
Control of Heavy Loads	14.24	Bounded	II.1.28
Control Room Habitability	14.23	Bounded	II.1.29
Feedwater Line Break Analysis	N/A	Bounded	II.1.30
MUR Power Uprate Impact on FCS Accident/Transient Analyses Containment Pressure Analyses			
Containment Pressure Analysis for MSLB	14.16	Bounded	II.1.20
Containment Pressure Analysis for LOCA	14.16	Bounded	II.1.21

Accident/Transient	FCS USAR Section	Impact of Uprate on Current USAR Analysis	Section of this Document
Analyses Performed in Accordance with Specific Regulatory Requirements			
ATWS (10 CFR 50.62)	3.3.1.7	Bounded	II.1.31
SBO (10 CFR 50.63)	8.7	Bounded	V.11

Design Operating Parameters and Initial Conditions

The revised NSSS design thermal and hydraulic parameters that were changed as a result of the MUR Uprate Program serve as the basis for the NSSS analyses and evaluations

The NSSS design parameters are the fundamental parameters used as input in all of the NSSS analyses. These design parameters are the primary and secondary side system conditions (temperatures, pressures, and flow) that are used as the basis for the NSSS analyses and evaluations. These parameters are revised to accommodate the proposed 1.67% increase in licensed core power from 1500 MWt to 1525 MWt. The NSSS parameters were conservatively generated for a 2% core power uprating to 1530 MWt (based on an assumed initial core power of 1500 MWt) to bound the actual uprating. Furthermore, the evaluations have been performed to support a power uprate such that the sum of the uprate plus uncertainty is less than or equal to 2%. In support of the 1.67% power uprate, these parameters have been incorporated, as required, into the applicable safety analyses and NSSS system and component evaluations.

The revised NSSS design thermal and hydraulic parameters that were changed as a result of the MUR Uprate Program are presented in Table 3, "FCS MUR Power Uprate - NSSS Design Parameters."

Table 3 FCS Thermal Design Parameters

Parameter	Current Condition	MUR Uprate
Core Power (MWt)	1500	1525
T_{hot} (°F)	593.3	594.1
T_{cold} (°F)	543	543
T_{ave} (°F)	568.2	568.6
Steam Flow per SG (lb/hr)	3.322 E6	3.386 E6
Steam T (°F)	520	520
Feedwater Flow per SG (lb/hr)	3.322 E6	3.386 E6
Feedwater T (°F)	442	443.6

References (Introduction Section)

1. NRC Regulatory Issue Summary 2002-03: Guidance On The Content Of Measurement Uncertainty Recapture Power Uprate Applications, Nuclear Regulatory Commission Office Of Nuclear Reactor Regulation, January 31, 2002.

2. AEP:NRC:2902, "Donald C. Cook Nuclear Plant Unit 2 Docket No. 50-316 License Amendment Request for Appendix K Measurement Uncertainty Recapture – Power Uprate Request, November 15, 2002.

I. Feedwater Flow Measurement Technique and Power Measurement Uncertainty

Instrumentation

The feedwater flow measurement system being installed at FCS is the CROSSFLOW ultrasonic flow measurement system. The installation of this system conforms to the requirements of the topical report cited next.

A. The referenced topical report for the CROSSFLOW system is CENPD-397-P-A, Revision 1, Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology, May, 2000. Updated to include safety evaluation reference.

B. The NRC approved CENPD-397-P, Revision-01-P "Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology" for referencing in power uprate license applications in a safety evaluation dated March 20, 2000 (TAC No. MA6452).

C. CROSSFLOW XT System

The Westinghouse CROSSFLOW XT ultrasonic flow measurement (UFM) system is used in conjunction with the plant process computer, to support the increase in reactor power. Reactor power is calculated using plant supplied inputs for feedwater temperature, steam generator pressure, steam generator moisture carryover, blowdown flow and feedwater venturi flow that has been corrected to improve its accuracy using the CROSSFLOW™ system. The components and information flow paths are shown in Figure I-1, "Block Diagram of the FCS CROSSFLOW System."

Precision temperature instrumentation will be installed for feedwater temperature measurement. This instrumentation will replace the existing temperature instrumentation currently used by the calorimetric calculation, XC105 (T1396 and T1399). The instrumentation will consist of precision matched RTDs and transmitters manufactured by Rosemount. The new RTDs and transmitters will be installed in the same place as the existing equipment.

The CROSSFLOW XT system consists of two sets ultrasonic sensors that are permanently mounted on the main feedwater common header, cables, signal conditioning unit, multiplexer and a data processing computer. The CROSSFLOW meter is mounted upstream of where the auxiliary feedwater enters the pipe. The CROSSFLOW system provides two measurements of total feedwater flow. The two transducer sets (4 transducers in each set) are mounted on a single metal support frame that attaches, externally, to the feedwater pipe.

Signals are passed from the two sets of ultrasonic flow meters, through the multiplexer to the signal conditioning unit and data processing computers located in a non-harsh environment area. The functions of the signal conditioning unit and data processing computer are described in the Topical Report CENPD-397-P-A, Revision 1, Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology, May 2000. The multiplexer simply alternates the ultrasonic signal from the signal-conditioning unit between the two-transducer sets to provide the two independent flow measurements.

The data processing computer receives values of feedwater flow, temperature and pressure for each loop from the plant process computer. The data processing computer then calculates a feedwater density and compensates for any thermal growth of the feedwater pipe due to a change in feedwater temperature. It also calculates an instantaneous correction factor for the venturi by measuring the feedwater flow with the CROSSFLOW meter and then dividing it by

the corresponding sum of the venturi readings for the same time period. The instantaneous correction factor is next added to a moving average of correction factors in order to smooth the data. The data processing computer then verifies the accuracy of the correction factor and passes the smoothed correction factor back to the plant computer along with a quality flag indicating that the correction factor meets the required accuracy for the Appendix K power uprate.

D. Compliance with NRC SER

The installation of the CROSSFLOW flow measurement system at FCS will be consistent with topical report CENPD-397-P-A. In addition to the installation requirements, the NRC identified the following four criteria that must be addressed by licensees requesting a license amendment based on the Topical Report. FCS will be consistent with the four criteria described below.

Criterion 1
Discuss maintenance and calibration procedures that will be implemented with the incorporation of the CROSSFLOW system, including processes and contingencies for inoperable CROSSFLOW instrumentation and the effect on thermal power measurements and plant operation.

Response to Criterion 1
The first criterion is to develop maintenance and calibration procedures that will be implemented with the CROSSFLOW UFM installation, including the process and contingencies for an inoperable CROSSFLOW UFM and the effect on thermal power measurement and plant operation.

Installation, maintenance, and calibration will be performed using FCS maintenance and calibration procedures, which will be developed from vendor information and FCS specific experience, or will be performed by a combination of vendor and FCS procedures.

Verification of proper CROSSFLOW system operation is provided by onboard system diagnostics. CROSSFLOW operation will be monitored on a periodic basis using an internal time delay check. The onboard system diagnostics enable verification that the signal conditioning unit, computer, and software remain within the stated accuracy.

The effective flow measurement uncertainty for the CROSSFLOW system, when operating properly, is assumed to be 0.3% of full power flow (feedwater mass flow uncertainty, Attachment 3). This number is based on preliminary test data taken by Westinghouse. The final number will be determined after the system is installed and prior to an increase in plant power. The uncertainty analysis will be updated to reflect the final uncertainty number. Based on vendor experience and preliminary test data taken at FCS, it is not anticipated that the flow measurement uncertainty will be greater than 0.3% after system installation. The accuracies of all other process input parameters to the core thermal power calculation reflect either vendor or operating data uncertainties for one full fuel cycle. Hence, there is only one condition that requires an explanation for the proposed actions - an inoperable CROSSFLOW system. The following paragraphs describe the proposed actions for this condition.

CROSSFLOW UFM failure will be detected and transmitted to the Plant Computer and will cause an audible alarm in the control room. The CROSSFLOW system does not perform any safety function and is not used to directly control any plant systems. However, adjustments to

RPS power indication based on CROSSFLOW are considered important to safety. Operations will enter an operating procedure that will contain a step for CROSSFLOW UFM failure. Plant operations may remain at an RTP of 1525 MWt, while continuing to use the last valid CROSSFLOW UFM correction factor in the heat balance calculation. If the CROSSFLOW system is not returned to service within 24 hours, power will be reduced and maintained at the appropriate power levels until the CROSSFLOW UFMs are returned to service. The power level is described below.

Condition	Measurement Uncertainty	Power Level
With CROSSFLOW	RTP Uncertainty with UFM	1525 MWt
Without CROSSFLOW	RTP Uncertainty without UFM	1500 MWt

As part of the approved amendment implementation process, OPPD will revise appropriate operations procedures to reflect the above responses to the unavailability of the CROSSFLOW system, and will include this information in the operator training program.

Criterion 2
For a plant that currently has CROSSFLOW UFM installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the CROSSFLOW UFM system and bound the analysis and assumptions set forth in Topical Report CENPD-357-P-A.

Response to Criterion 2
The CROSSFLOW system when installed will satisfy the requirements of Topical Report CENPD-357-P-A and will be bounded by them. At FCS the location of the CROSSFLOW UFM is representative of the location requirements set forth in the Topical Report. The CROSSFLOW UFM will be installed approximately 54 pipe diameters downstream of the nearest elbow where the flow is fully developed. In the Westinghouse Response to NRC RAIs regarding WCAP-15689-P "Evaluation of Transit-Time and Cross Correlation Ultrasonic Flow Measurement Experience with Nuclear Plant Feedwater Flow Measurement, March 14, 2002, it was stated that based on high temperature laboratory tests run in the past that demonstrate plant operating conditions, the flow is fully developed for 15 or more diameters downstream of a 90° elbow.

Criterion 3
Confirm that the methodology used to calculate the uncertainty of the CROSSFLOW UFM in comparison to current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3
The methodology used to calculate the uncertainty of the CROSSFLOW UFM in comparison to the current feedwater flow instrumentation is based on accepted plant setpoint methodology, with regard to the development of instrument uncertainty in Regulatory Guide 1.105 and ISA S67.04, as described in the Topical Report. An alternative methodology is not used.

FCS has completed the uncertainty calculation with a mass flow accuracy of equal to or better than 0.3% of rated feedwater flow for the FCS specific installation. The FCS CROSSFLOW uncertainty calculations are consistent with the methodology described in the Topical Report.

Criterion 4
For plants where the ultrasonic meter was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original UFM installation and calibration assumptions.

Response to Criterion 4
For FCS there will be no site-specific piping configuration calibration because the installation is equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers.

The meter installation is located on long straight sections of piping and will be far enough from disturbance to conform to the proprietary installation requirements of the Topical Report.

E. The following table summarizes the core thermal power measurement uncertainty at FCS:

Table I-1 FCS Process Parameter Inputs to Reactor Thermal Power

Independent Variable	Term	Uncertainty	Sensitivity
Feedwater Flow	UW_{FW}	0.2896 %	1.0107
Feedwater Temperature	UT_{FW}	0.69 F	0.4903
SG Pressure	UP_{SG}	14.68 psia	0.0144
SG Moisture Carryover	UM_{CO} A/B	0.11%/0.05%	0.0011/0.0008
SG Blowdown Flow	UW_{BD}	1873 lbm/hr	0.0034
SG Blowdown Temperature	UT_{BD}	2.94 F	0.0038

Attachment 3 provides the detailed proprietary calorimetric uncertainty calculation performed for FCS.

F. The following information addresses specific aspects of calibration and maintenance procedures addressing the CROSSFLOW system.

i. Calibration and maintenance will be performed by OPPD I&C personnel using site procedures. The site procedures will be developed using the CROSSFLOW technical manuals. All work will be performed in accordance with site work control procedures.

Routine preventive maintenance activities will include physical inspections, and power supply checks.

I&C maintenance personnel will be trained in the operation of the equipment prior to performing any system calibration.

ii. The CROSSFLOW system is designed and manufactured in accordance with Westinghouse's quality assurance program (class 4, considered important to safety) and in accordance with the topical report CENPD-357-P-A.

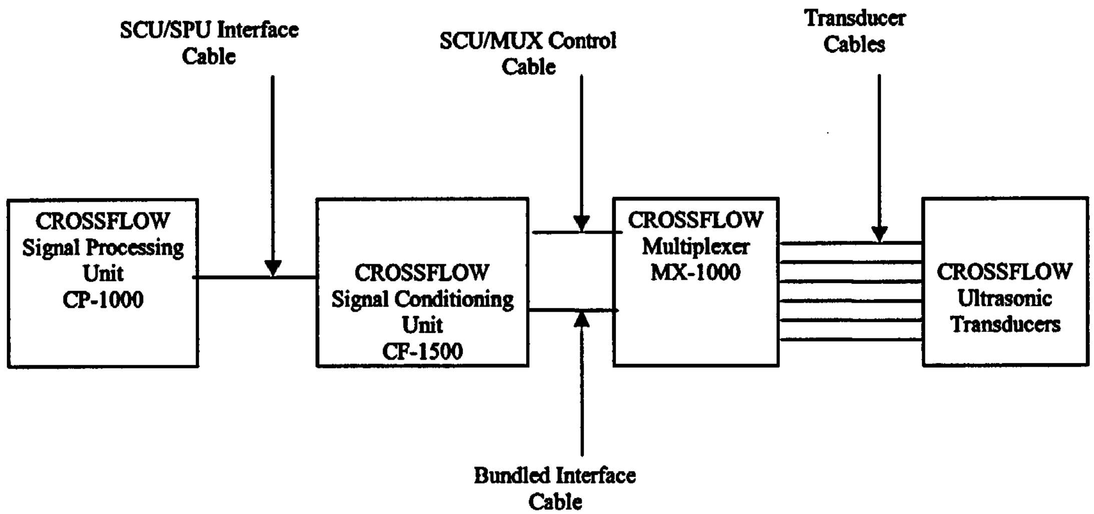
iii. Corrective actions involving maintenance will be performed by OPPD I&C maintenance personnel, qualified in accordance with FCS training program.

iv. Reliability of the CROSSFLOW system will be monitored by OPPD reliability engineering personnel. Equipment problems for all plant systems will be under site work control processes. Corrective Action procedures will be maintained that include instructions for notification of deficiencies and error reporting.

G/H. The proposed allowed outage time for operation at the uprated power level with the CROSSFLOW system out of service is 24 hours, provided steady state conditions exist.

Figure I-1 Block Diagram of the FCS CROSSFLOW System

Basic SCU connection layout



II. Accidents and Transients for which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level

II.1 Updated Safety Analysis Report

Table 2 summarizes the FCS accident and transient analyses that were determined to bound plant operation at the 1.67% power level proposed by the MUR Uprate Program. Details of these evaluations with specific references for analysis and NRC approval of methodology follow in each subsequent sub-section. All of the USAR Chapter 14 transients were reviewed for impact from a power increase of up to 1.67% from a MUR Power Uprate. This listing consists of those events represented in USAR Chapter 14, along with Feedwater Line Break, which is used for the basis for Auxiliary Feedwater Delivery.

II.1.1 USAR 14.2 - CEA Withdrawal - Run at 102% Power - not affected.

References: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.1.1 Page 27.

The AOR for this event was performed in Cycle 21 in F/ANP Calculation E-6088-595-10 "Fort Calhoun Cycle 21 HFP CEA Withdrawal Analysis", dated 4/24/02 for HFP and Calculation E-6088-595-9 "Fort Calhoun Cycle 21: HZP CEA Withdrawal Analysis", dated 4/24/02 for HZP and were performed with S-RELAP using methodology from EMF-2310(P)(A) Revision 0 "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", dated May 2001, Approved by the NRC in SER Project No. 702 from R. R. Landry and transmitted to F/ANP via letter from S. A. Richards (NRC) to J. F. Mallay (F/ANP) dated 5/11/01. This methodology was approved for use by OPPD in Technical Specification Amendment No. 203 transmitted to OPPD in Letter from A. B. Wang to R. T. Ridenoure "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT RE: ADDITION OF TOPICAL REPORT REFERENCES TO TS 5.9.5, "CORE OPERATING LIMITS REPORT" (TAC NO. MB3449)", dated 3/4/02.

II.1.2 USAR 14.3 - Boron Dilution - Only run at shutdown conditions, power operating bounded by CEA Withdrawal - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Page 31.

The AOR for this event was performed in Cycle 20 in EA-FC-00-024 "Cycle 20 Boron Dilution Analysis", which used methodology outlined in OPPD-NA-8303 Revision 4, "Transient and Accident Methods and Verification", and approved by the NRC in SER from L. Kopp dated 3/29/94 and transmitted to OPPD via letter from S. Bloom (NRC) to T. L. Peterson (OPPD) dated 3/29/94.

II.1.3 USAR 14.4 - CEA Drop - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.3.1 Page 36.

The AOR for this event was performed in Cycle 21 in F/ANP Calculation E-6088-595-11 "Fort Calhoun Station Cycle 21 Control Element Assembly Drop Analysis", dated 4/24/02 using methodology from EMF-2310(P)(A) Revision 0 "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" dated May 2001, Approved by the NRC in SER Project No. 702

from R. R. Landry and transmitted to F/ANP via letter from S. A. Richards (NRC) to J. F. Mallay (F/ANP) dated 5/11/01. This methodology was approved for use by OPPD in Technical Specification Amendment No. 203 transmitted to OPPD in Letter from A. B. Wang to R. T. Ridenoure "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT RE: ADDITION OF TOPICAL REPORT REFERENCES TO TS 5.9.5, "CORE OPERATING LIMITS REPORT" (TAC NO. MB3449)", dated 3/4/02.

II.1.4 USAR 14.5 - Mal-Positioning of the Non-Trippable CEAs - Not allowed by Technical Specifications - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Page 40.

II.1.5 USAR 14.6.1 - Loss of Coolant Flow Event - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.5.1 Page 42.

The AOR for this event was performed in Cycle 18 in EA-FC-97-029 "Cycle 18 Loss of Coolant Flow", which used methodology outlined in OPPD-NA-8303 Revision 4, "Transient and Accident Methods and Verification", and approved by the NRC in SER from L. Kopp dated 3/29/94 and transmitted to OPPD via letter from S. Bloom (NRC) to T. L. Peterson (OPPD) dated 3/29/94. The thermal-hydraulic evaluation of this event was redone (to address a lower RCS Technical Specification Flow) using X-COBRA-IIIC for Cycle 21 in F/ANP Calculation E-4750-595-2, using methodology EMF-92-1 53(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994. This methodology was approved by the NRC by SER transmitted to Siemens Power Corporation by letter A. C. Thadani (NRC) to R. A. Copeland (SPC) dated 12/28/93. This methodology was approved for use at OPPD in Technical Specification Amendment Number 196 transmitted in a letter from L. R. Wharton (NRC) to S. K. Gambhir (OPPD) "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT (TAC NO. MB0083)", dated 3/14/01.

II.1.6 USAR 14.6.2 - Seized Rotor Event - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.6.1 Page 46.

The AOR for this event was performed in Cycle 20 in E-4370-595-1 "Fort Calhoun Unit 1 RCP Rotor Seizure (SRP 15.3.3) Analysis" using methodology from ANF-89-151(P)(A), Revision 0 "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events", dated May 1992. This methodology was approved by the NRC in TER EGG-RTS-10032 from C. P. Fineman, dated January, 1992 and transmitted to Siemens Nuclear Power Corporation by letter A. C. Thadani (NRC) to R. A. Copeland (SNPC) dated 3/16/92. This methodology was approved for use at OPPD in Technical Specification Amendment Number 196 transmitted in a letter from L. R. Wharton (NRC) to S. K. Gambhir (OPPD) "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT (TAC NO. MB0083)", dated 3/14/01.

II.1.7 USAR 14.7 - Idle Loop Startup - Not allowed by Technical Specifications - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Page 48.

II.1.8 USAR 14.8 - Turbine Generator Overspeed Incident - Event unrelated to T/H conditions - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Page 49.

II.1.9 USAR 14.9.1 - Loss of Load to Both Steam Generators - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.9.1 Page 51.

The AOR for this event was performed in Cycle 17 in EA-FC-97-004 "Evaluation of the Effect of Increased Line Pressure Drop on MSSV and PSV Setpoints", Revision 1 which used methodology outlined in OPPD-NA-8303 Revision 4, "Transient and Accident Methods and Verification", and approved by the NRC in SER from L. Kopp dated 3/29/94 and transmitted to OPPD via letter from S. Bloom (NRC) to T. L. Peterson (OPPD) dated 3/29/94.

II.1.10 USAR 14.9.2 - Loss of Load to One Steam Generator - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.10.1 Page 55.

The AOR for this event was performed in Cycle 9 in OSAR 83-37 "Cycle 9 Reload Analysis, Loss of Load to One Steam Generator Event", Revision 0 which used methodology outlined in OPPD-NA-8303 Revision 0, which was approved by the NRC SER transmitted to OPPD in Letter from J. R. Miller (NRC) to W. C. Jones (OPPD) "RELOAD CORE ANALYSIS METHODOLOGY REPORTS", dated 5/11/84.

II.1.11 USAR 14.10.1 - Loss of Feedwater Flow - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.11.1 Page 58.

The AOR for this event was performed in Cycle 17 in EA-FC-97-004 "Evaluation of the Effect of Increased Line Pressure Drop on MSSV and PSV Setpoints", Revision 1 which used methodology outlined in OPPD-NA-8303 Revision 4, "Transient and Accident Methods and Verification", and approved by the NRC in SER from L. Kopp dated 3/29/94 and transmitted to OPPD via letter from S. Bloom (NRC) to T. L. Peterson (OPPD) dated 3/29/94.

II.1.12 USAR 14.10.2 - Loss of Feedwater Heating - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.12.1 Page 62.

The AOR for this event was performed in Cycle 6 by Exxon in XN-NF-79-77 "Fort Calhoun Nuclear Plant Cycle 6 Safety Analysis Report", dated October 1979 and reported in XN-NF-79-79 "Fort Calhoun Cycle 6 Reload Plant Transient Analysis Report", dated October 1979. This evaluation used the code PTSPWR, which is described in XN-74-5 "Description of the Exxon

Nuclear Plant Transient Simulation Model for Pressurizer Water Reactors (PTSPWR)" Revision 1 dated May 1975 as applied to a Combustion Engineering PWR in XN-NF-77-18 "Plant Transient Analysis of the Palisades Reactor for Operation at 2350 MWt".

II.1.13 USAR 14.11 - Excess Load - Run at 102% Power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.13.1 Page 65.

The AOR for this event was performed in Cycle 21 in Calculation E-6088-595-6 "Fort Calhoun Station Cycle 21 Excess Load Increase", dated May 9, 2002 using methodology from EMF-2310(P)(A) Revision 0 "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" dated May 2001, Approved by the NRC in SER Project No. 702 from R. R. Landry and transmitted to F/ANP via letter from S. A. Richards (NRC) to J. F. Mallay (F/ANP) dated 5/11/01. This methodology was approved for use by OPPD in Technical Specification Amendment No. 203 transmitted to OPPD in Letter from A. B. Wang to R. T. Ridenoure "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT RE: ADDITION OF TOPICAL REPORT REFERENCES TO TS 5.9.5, "CORE OPERATING LIMITS REPORT" (TAC NO. MB3449)", dated 3/4/02.

II.1.14 USAR 14.12 - Main Steam Line Break Accident - run at 100% power, and 102% feedwater flow.

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. It should be noted that the rated thermal power, which is increasing by 1.67% 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 result for the full power cases would be obtained if they were to be rerun with a 1.67% 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.

Reference: Framatome Calculation E-6088-595-7 "Fort Calhoun Cycle 21 Post Scram Steam Line Break Analysis – Cases Initiated from Full Power", B. A. Reeves, 4/24/02, Page 7-10. Previous analysis reference EA-FC-00-028 "Cycle 20 Transients Summary", dated 3/16/01, Table 5.14.1, Page 86. 102% Feedwater flow reference EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.14.1 Page 69. This event can be dispositioned by stating that the input power level for this event is not a primary contributor to this event. The moderator temperature coefficient, which is a primary contributor to the severity of this event, will continue to be bounded by the COLR limit and the higher average primary temperature (approximately 0.4°F) will not significantly change the cooldown during the event (which is more than 200°F) and will be bounded by the fact that the mixed core penalty of 2% DNBR margin, which was applied to the event in the analysis is no longer necessary for Cycle 22, which will be a full core of Framatome HTP fuel. Therefore, if the event was reanalyzed, additional margin would be available for Cycle 22. The AOR for this event was performed in Calculation E-6088-595-7 "Fort Calhoun Station Cycle 21 Post-Scram Steam Line Break Analysis – Cases Initiated from Full Power", dated 5/9/02 at HFP and in E-6088-595-8 "Fort Calhoun Station Cycle 21 Post-Scram Steam Line Break Analysis – Cases Initiated from Hot Zero Power", dated 5/9/02. These analyses were performed with S-RELAP using methodology from EMF-2310(P)(A) Revision 0 "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", dated May 2001, Approved by the NRC in SER Project No. 702 from R. R. Landry and transmitted to F/ANP via letter from S. A. Richards (NRC) to J. F. Mallay (F/ANP) dated 5/11/01. This methodology was approved for use by OPPD in Technical Specification Amendment No. 203 transmitted to OPPD in Letter from A. B. Wang to R. T. Ridenoure "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT RE: ADDITION OF TOPICAL REPORT REFERENCES TO TS 5.9.5, "CORE OPERATING LIMITS REPORT" (TAC NO. MB3449)", dated 3/4/02.

<u>II.1.15 USAR 14.13 - CEA Ejection - Run at 102% Power - not affected.</u>

Reference: Siemens Power Corporation Calculation E-4257-595-4 "Fort Calhoun Cycle 20 Control Rod Ejection Analysis", C. D. Fletcher, dated 11/28/00, Page 2-2.
Reference: EA-FC-02-007, "Cycle 21 CEA Ejection Verification – Westinghouse", dated 3/12/02.

The AOR for this event was performed in Cycle 20 for F/ANP in E-4257-595-4 "Ft. Calhoun Cycle 20 Control Rod Ejection Analysis", dated 11/28/00 and reported to OPPD in EMF-2488 "Fort Calhoun Control Element Assembly Ejection Analysis", dated December 2000 using methodology from ANF-89-151(P)(A), Revision 0 "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events", dated May 1992. This methodology was approved by the NRC in TER EGG-RTS-10032 from C. P. Fineman, dated January, 1992 and transmitted to Siemens Nuclear Power Corporation by letter A. C. Thadani (NRC) to R. A. Copeland (SNPC) dated 3/16/92. This methodology was approved for use at

OPPD in Technical Specification Amendment Number 196 transmitted in a letter from L. R. Wharton (NRC) to S. K. Gambhir (OPPD) "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT (TAC NO. MB0083)", dated 3/14/01.

The thermal-hydraulic evaluation of this event was redone (to address a lower RCS Technical Specification Flow) using X-COBRA-IIIC for Cycle 21 in F/ANP Calculation E-4750-595-2, using methodology EMF-92-1 53(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994. This methodology was approved by the NRC by SER transmitted to Siemens Power Corporation by letter A. C. Thadani (NRC) to R. A. Copeland (SPC) dated 12/28/93. This methodology was approved for use at OPPD in Technical Specification Amendment Number 196 transmitted in a letter from L. R. Wharton (NRC) to S. K. Gambhir (OPPD) "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT (TAC NO. MB0083)", dated 3/14/01. The analysis covering the Westinghouse fuel in the core was reperformed for Cycle 19 and the results transmitted to OPPD in Letter 99CF-G-0024, CAB-99-320 from M. F. Muenks (Westinghouse) to T. A. Heng (OPPD) "OMAHA PUBLIC POWER DISTRICT FORT CALHOUN Cycle 19 CEA Ejection Re-Analysis Results", dated 8/23/99.

This analysis used methodology approved by the NRC in WCAP-7588, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", Revision 1-A, January 1975. This methodology was approved for Fort Calhoun Station in License Amendment 144, which approved the inclusion of OPPD-NA-8303 Revision 3, which was transmitted to OPPD in Letter from D. L. Wigginton to W. G. Gates "TOPICAL REPORT OPPD-NA-8303, "RELOAD CORE ANALYSIS METHODOLOGY, TRANSIENT AND ACCIDENT METHODS AND VERIFICATION," REVISION 3 (TAC NO. M80436)" on 3/2/92. There will be no Westinghouse fuel in Cycle 22, so this AOR will no longer be applicable.

<u>II.1.16 USAR 14.14 – Steam Generator Tube Rupture Accident – only evaluated as a Thermal Hydraulic Event to determine steaming rates for atmospheric releases. Steaming rates determined based on 102% power level. See Section II.2.4 for details on dose consequences which were evaluated for core inventories, and power level of 1530 MWt, with applicable 102% power level steaming rates.</u>

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Page 74.

<u>II.1.17 USAR 14.15.4 – Large Break Loss of Coolant Accident – Run at maximum allowable peak linear heat generation rate (PLHGR) of 15.5 kw/ft, will limit maximum allowable F_q some calculations run at 102% power – not affected.</u>

Reference: Framatome ANP Calculation E-4750-868-1 "Fort Calhoun Cycle 21 LBLOCA Analysis with Reduced RCS Flowrate", R. C. Gorman, 4/5/02, Page 1-1 for kw/ft limit, Pages 6-22 and 6-84 for 102% power (1530 MWt).

The AOR for this event was originally performed in E-4257-868-1 Revision 1, "Ft. Calhoun Cycle 20 LBLOCA Analysis", dated 12/15/00 and summarized in EMF-2506 Revision 0, "Fort Calhoun Large Break LOCA/ECCS Analysis", dated December 2000. It was updated for Cycle 21 in E-4750-868-1 Revision 0, "Fort Calhoun Cycle 21 LBLOCA Analysis with Reduced RCS Flowrate", dated 4/5/02 and summarized in EMF-2734 Revision 0, "Fort Calhoun Cycle 21 Large Break LOCA/ECCS Analysis with reduced RCS Flow Rate", dated April 2002. The methodology for this evaluation was approved by the NRC in Siemens report EMF-2087(P)(A),

Revision 0, "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications", dated June 1999. This methodology was approved for use by OPPD in Technical Specification Amendment Number 196 transmitted in a letter from L. R. Wharton (NRC) to S. K. Gambhir (OPPD) "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT (TAC NO. MB0083)", dated 3/14/01.

<u>II.1.18 USAR 14.15.5 – Small Break Loss of Coolant Accident – Run at 102% power – not affected.</u>

Reference: Siemens Power Corporation Calculation E-4257-598-2 "Fort Calhoun Cycle 20 SBLOCA Analysis", G. S. Uyeda, dated 11/28/00, Page 2-3.

The AOR for this event was performed in Framatome ANP Calculation E-4257-598-2 Revision 0, "Fort Calhoun Cycle 20: SBLOCA Analysis", dated 11/22/00 and summarized in EMF-2482 Revision 0, "Fort Calhoun Small Break LOCA Analysis", dated December 2000. The methodology for this event was approved by the NRC in Exxon Nuclear Report XN-NF-82-49(P)(A) Revision 1, Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model", dated December 1994. This methodology was approved for use by OPPD in Technical Specification Amendment Number 196 transmitted in a letter from L. R. Wharton (NRC) to S. K. Gambhir (OPPD) "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT (TAC NO. MB0083)", dated 3/14/01.

<u>II.1.19 USAR 14.15.6 – Long Term Core Cooling/Hot Leg Switchover Analysis – Run at 102% power – not affected.</u>

Reference: Westinghouse Calculation SEC-SAII-3821-C1 "Fort Calhoun Unit 1 Hot Leg Switchover Analysis", D. J. Fink, Dated 6/21/96, Page 17.

The AOR for this analysis is Westinghouse Calculation SEC-SAII-3821-C1 "Fort Calhoun Unit 1 Hot Leg Switchover Analysis", D. J. Fink, Dated 6/21/96. The HLSO code used for this code is described in WAF B-1411, and the methodology used was performed in SEC-SAII-3821-C0, "Containment Sump Boron and Hot Leg Switchover Calculations for Fort Calhoun Unit 1", dated December, 1991. In EA-FC-90-004 Revision 0 "Mixed Vendor Core Data List for Fort Calhoun Unit 1", page 3001, it states "The Westinghouse commitment for assurance of Long Term Core Cooling is identified in WCAP-8339".

<u>II.1.20 USAR 14.16 – Containment Pressure Analysis for MSLB run at 102% power – not affected.</u>

Reference licensing basis for Mass and Energy Release data transmitted to OPPD in LTR-OA-01-13 Rev. 0 "Transmittal of Containment Licensing Basis LOCA and MSLB Mass and Energy Release Data for Fort Calhoun Station", September 14, 2001 Page 7 of Attachment 1 (Page 107 of EA-FC-02-001). Current AOR EA-FC-93-022, Revision 0, "MFIV Stroke Time Evaluation/Containment Response".

The Equipment Environmental Qualification (EEQ) case of Record for MSLB (and bounds the LOCA EEQ case) is documented in EA-FC-97-038 Page 20 (Page 14 of 25 of ABB calculation 002-ST97-C-028) which uses the model base case of Case 6 of ABB Calculation 002-AS93-C-005 Page 11 of 73 (Contained in EA-FC-93-022 as Page 27). The particular inputs for power are listed on Page 27 of 73 in ABB Calculation 002-AS93-C-005 (Page 43 of EA-FC-93-022)

PWMT (Core Thermal Power) of 1505.6 (1500 MWt + 5.6 MW pump heat) which is multiplied by the Initial Power Fraction, CCT(2) of 1.02 (102%). It is more succinctly shown on page 26 of EA-FC-93-002 (page 10 of 73 of ABB calculation 002-AS93-C-005) which describes Case 6 as "MSLB from 102% initial power, the Feedwater Regulating Valve to the ruptured SG fails as is, EEQ models used." The code used for producing the mass and energy releases for the MSLB containment analysis is SGN-III which was approved by the NRC for use in NUREG-75/112, "Safety Evaluation Report related to the preliminary design of the Standard Reference System CESSAR System 80," December 1975. The code used for performing the containment pressure analysis is CONTRANS, which was approved by the NRC for use in NRC Letter O. D. Parr (NRC) to A. E. Scherer (CE) which details approval of CENPD-140-A "Description of the CONTRANS Digital computer Code for Containment Pressure and Temperature Analysis".

II.1.21 USAR 14.16 – Containment Pressure Analysis for LOCA run at 104% power – not affected.

Reference licensing basis for Mass and Energy Release data transmitted to OPPD in LTR-OA-01-13 Rev. 0 "Transmittal of Containment Licensing Basis LOCA and MSLB Mass and Energy Release Data for Fort Calhoun Station", September 14, 2001 Page 29 of Attachment 1 (Page 129 of EA-FC-02-001).

Current AOR EA-FC-97-038, Revision 0, "Evaluation of Containment Spray Flowrate with Measurement Uncertainty for FCS Containment Analysis". The LOCA EEQ case has not been run in some time as it was found to be bounded by the MSLB EEQ case, which is documented above. The short term mass and energy releases for LOCA were developed with CEFLASH-4A, which was approved for use by the NRC in CENPD-133 Supplement 5-A, "CEFLASH-4A A Fortran77 Digital Computer Program for Reactor Blowdown Analysis", dated June 1985. The code used for performing the containment pressure analysis is CONTRANS, which was approved by the NRC for use in NRC Letter O. D. Parr (NRC) to A. E. Scherer (CE) which details approval of CENPD-140-A "Description of the CONTRANS Digital computer Code for Containment Pressure and Temperature Analysis",

II.1.22 USAR 14.17 - Generation of Hydrogen in Containment – This analysis is not directly related to power level thus, not affected by MUR power uprate. The radiolysis portion of the hydrogen generation was based on a 1% Zirconium cladding as a conservative input.

Reference: EA-FC-98-061 and OSAR 87-48.

II.1.23 USAR 14.18 - Fuel Handling Accident - Run at 102% power for core inventory - not affected

Reference: USAR Section 14.18 Table 14.18-1 Page 5 of 8.

II.1.24 USAR 14.19 - Gas Decay Tank Rupture - Run at 102% power for core inventory - not affected

Reference: USAR Section 14.19 Page 2 of 4.

II.1.25 USAR 14.20 - Waste Liquid Incident - core inventory power level 102% evaluated- not affected

Reference: USAR Section 14.20

II.1.26 USAR 14.21 - Maximum Hypothetical Accident - Same as USAR 14.15, not affected

Reference: See USAR 14.15

II.1.27 USAR 14.22 - Reactor Coolant System Depressurization - Run at 102% power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.21.1 Page 82.

The AOR for this analysis was performed in Cycle 19 in EA-FC-98-051 Revision 0, "Cycle 19 RCS Depressurization Analysis" using methodology outlined in OPPD-NA-8303 Revision 4, "Transient and Accident Methods and Verification", and approved by the NRC in SER from L. Kopp dated 3/29/94 and transmitted to OPPD via letter from S. Bloom (NRC) to T. L. Peterson (OPPD) dated 3/29/94. The thermal-hydraulic evaluation of this event was redone (to address a lower RCS Technical Specification Flow) using X-COBRA-IIIC for Cycle 21 in F/ANP Calculation E-4750-595-2, using methodology EMF-92-1 53(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994. This methodology was approved by the NRC by SER transmitted to Siemens Power Corporation by letter A. C. Thadani (NRC) to R. A. Copeland (SPC) dated 12/28/93. This methodology was approved for use at OPPD in Technical Specification Amendment Number 196 transmitted in a letter from L. R. Wharton (NRC) to S. K. Gambhir (OPPD) "FORT CALHOUN STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT (TAC NO. MB0083)", dated 3/14/01.

II.1.28 USAR 14.23 Control Room Habitability – Not Affected.

Toxic gas releases evaluated are not power dependent. Control room dose as a result of radiological consequences has been reanalyzed for AST and now bounds MUR power uprate see Section II.2.5.

II.1.29 USAR 14.24 Control of Heavy Loads Heavy Load Drop Analysis – Source Term Developed for 1530 MWt core – not affected.

Reference: USAR Chapter 14 reflects consideration of radiological consequences of accidents with the 1530 MWt core inventory. Application for Amendment of Operating License, "(Alternate Source Term)", LIC-01-0010, February 7, 2001, OPPD to USNRC Document Control Desk. "Implementation of Alternative Source Terms Site Boundary & Control Room Dose Analyses for Fort Calhoun Station", January 2001, Stone & Webster. NRC approved the FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221)". Section 2.2.1 of the SER addresses that FCS consistently used the 102% core power per R. G. 1.49, and as such approved of the application.

II.1.30 Feedwater Line Break Analysis (used as a basis for required AFW delivery) - Run at 102% power - not affected.

Reference: EA-FC-02-016, "Cycle 21 Transients Summary", dated 6/19/02, Table 5.20.1 Page 79.

The AOR for this event is EA-FC-97-012 Revision 0, "Evaluation of Reduced Auxiliary Feedwater Flow". This is not a Chapter 14 event for FCS, so no NRC approved methodology exists. This analysis was, however, reviewed by CE and it is assumed that the event was run in accordance with their standard methodology at that time.

II.1.31 USAR 7.2.11 Anticipated Transient Without Scram-Diverse Scram System

The DSS has been designed and installed to meet the requirements of 10 CFR 50.62, the ATWS rule. The DSS augments the protective function or the Reactor Protection System, by providing an independent means of initiating a Reactor Trip. The DSS uses components that are diverse, independent and separate from the Reactor Protection System to initiate a Reactor Trip for anticipated operational occurrences which result in an overpressurization of the RCS. The DSS introduces diversity into the Reactor Protective System, thereby reducing the probability of a Reactor Coolant Overpressurization from an ATWS event.

The DSS has a setpoint higher than the RPS high pressurizer pressure trip setpoint and less than the primary safety valve relief pressure setpoint. The setpoint is greater that the RPS high pressurizer pressure trip setpoint to avoid unnecessary scrams, to ensure that the DSS doesn't scram the reactor below the RPS trip setpoint. With respect to the upper limit on the setpoint, previous ATWS analyses have shown that the RCS pressure will temporarily plateau near the primary safety valve setpoint once the valves open. To prevent delay in generating the DSS trip signal the setpoint must be less than the safety valve setpoint. The MUR power uprate does not affect the current primary safety valve relief pressure setpoint or the high pressurizer pressure reactor trip setpoint. The DSS analysis for the FCS NSSS ATWS response is based on 1565 MWt. Thus, the conclusions regarding maintaining RCS pressure remain bounding for the MUR power uprate. Reference CE NPSD-354, "Functional Design Specification for the Diverse Scram System for Compliance with the ATWS Rule 10 CFR 50.62," dated May, 1986.

II.2 Loss of Coolant Accident and Loss of Coolant Accident-Related Events (Including Steam Generator Tube Rupture)

II.2.1 Loss of Coolant Accident Forces

The consequences of the LOCA forces on the reactor internals including lower structure, reactor core and upper guide structure have been analyzed for reactor coolant system breaks up to a double-ended rupture of a 32 inch pipe at zero and 102% of the current full (1500 MW) reactor thermal power. The maximum calculated stresses and deflections during blowdown for critical reactor components occur at zero power conditions that generate the highest pressure difference across the internal components (Reference II.2.1). These stresses and deflections were found to be below the critical stresses. Since the zero power conditions during the MUR power uprate will not change, it is concluded that the impact of the LOCA forces on the reactor internals is not affected by the MUR power uprate. Additionally, LOCA forces on the reactor internals at the MUR power uprate conditions of 101.67% of the current 1500 MW reactor thermal power are bounded by the current analysis of record performed at 102% of 1500 MW reactor thermal power.

II.2.2 Large Break Loss of Coolant Accident and Small Break Loss of Coolant Accident

The current licensing basis LBLOCA and SBLOCA analyses for FCS use approved methodology (See Section II, USAR 14.15.4 and 14.15.5 discussion). Both analyses used a nominal core power of 1500 MWt plus an additional 2% calorimetric power measurement uncertainty (yielding an assumed core power of 1530 MWt). FCS has proposed to reduce the power measurement uncertainty to 0.33% for FCS, and increase the nominal core power 1.67% to 1525 MWt. The analyses are conservative with respect to this uprate; thus, the uprate has no impact on the LBLOCA and SBLOCA analyses.

II.2.3 Post-Loss of Coolant Accident Analyses

Long Term Core Cooling (LTCC) containment sump boron concentration calculations are not directly affected by changes in core power. Changes in RCS temperatures can have a minor effect on the RCS mass contribution to the containment sump inventory used in the LTCC and HLSO calculations since the RCS mass is determined at some assumed conditions. The proposed FCS Appendix K power uprate is acceptable with respect to LOCA LTCC as the existing LTCC and HLSO analyses remain valid for uprate conditions.

II.2.3.1 Post-Loss of Coolant Accident Long-Term Core Cooling Evaluation

As reported above, LTCC containment sump boron concentration calculations are not directly affected by changes in core power. The changes in RCS conditions for the proposed uprate would have a negligible effect on the RCS mass assumed in the LTCC calculations since RCS fluid density would change only slightly for the proposed uprate conditions and the RCS is a relatively small contributor to the containment sump total mass.

II.2.3.2 Hot Leg Switchover Evaluation

For the uprate, the core power increase falls within the 2% power uncertainty used in the current analysis basis. Therefore, the core power increase does not impact the HLSO calculations. The changes in RCS conditions would have a negligible effect on the RCS mass assumed in the HLSO calculations since density varies only slightly for the proposed uprate conditions. The RCS is a relatively small contributor to the containment sump total mass. The proposed FCS Appendix K power uprate is acceptable with respect to LOCA LTCC as the existing LTCC and HLSO analyses remain valid for uprate conditions.

II.2.4 Steam Generator Tube Rupture

The analysis for the SGTR event, as documented in Section 14.14 of the USAR, is performed to demonstrate that the off-site radiological consequences remain below the guideline values. As input to the radiological consequences analysis, an SGTR T/H analysis was performed. The T/H analysis calculated the primary-to-secondary break flow and steam released to the environment. The SGTR analysis for on-site radiological consequences was approved by the NRC (Reference II.2.2). The SGTR T/H analysis considers core powers up to 1530 MWt. Therefore, the increase in core power to 1525 MWt is bounded by the analysis.

Therefore, the current analyses for the steam generator tube rupture event will bound the MUR Uprate Program.

II.2.5 USAR Chapter 14 Radiological Consequences LOCA and Non-LOCA events

Site Boundary & Control Room Dose Analyses for FCS

The Chapter 14 radiological consequences calculations were recently updated to reflect implementation of Alternate Source Term methodology (R. G. 1.183). As such, FCS re-analyzed all accident radiological consequences to the new methodology and was found to meet the criteria as specified in 10CFR50.67 and GDC 19 as stated in R. G. 1.183. The core inventory that was developed and approved for the radiological assessments was developed based on the following information and referenced details provided below:

Core Inventory-The inventory of fission products in the FCS reactor core was based on maximum full power operation of the core at a power level equal to the current licensed rated thermal power including a 2% instrument error per Regulatory Guide 1.49, and current licensed values of fuel enrichment and burnup. The equilibrium core inventory was calculated based on plant operation at 102% of the power level (i.e., at 1530 MWt), and assuming an 18-month fuel cycle. The core inventory developed by ORIGEN-S used a power level of 1530 MWt for LOCA, Fuel Handling Accident Heavy Load Drop, Seized Rotor, Control Rod Ejection, Main Steam Line Break, Steam Generator Tube Rupture, Waste Gas Decay Tank and Liquid Waste Tank radiological dose consequence assessments.

OPPD Calculation FC 06800, Rev. 0, Bounding Composite Equilibrium Core Inventory with Initial U-235 Enrichments of 3.5 w/o to 5.0 w/o (28 day outage, 520 days power operation).

Dose Consequences for the accident scenarios evaluated above were evaluated for the site boundary and control room in accordance with R. G. 1.183. (Implementation of Alternative Source Terms, Site Boundary and Control Room Dose Analyses for Fort Calhoun Station, January 2001, Stone and Webster).

USAR Chapter 14 reflects consideration of radiological consequences of accidents with the 1530 MWt core inventory. Application for Amendment of Operating License, "(Alternate Source Term)", LIC-01-0010, February 7, 2001, OPPD to USNRC Document Control Desk. "Implementation of Alternative Source Terms Site Boundary & Control Room Dose Analyses for Fort Calhoun Station", January 2001, Stone & Webster.

NRC approved the FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221)". Section 2.2.1 of the SER addresses that FCS consistently used the 1.02% core power per R. G. 1.49, and as such approved of the application.

References (Section II.2)

II.2.1 NRC approved the FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221)".
II.2.2 USAR Section 14.15.7.4

II.3 Containment Analyses

II.3.1 Steam Line Break Mass and Energy Releases

The licensing basis safety analysis related to the steam line break mass and energy releases for a containment response analysis were evaluated to determine the effect of a 1.67% power uprate. The evaluation determined that the analysis used 102% of 1505.6 MWt which included heat addition from the reactor coolant pumps as shown in Section 2, cards 48 and 50 of Reference II.3.1. (Note: the Westinghouse proprietary computer code SGN-III was used to perform the steam line break mass and energy releases. It was approved by the NRC in Reference II.3.2). Therefore, the power uprate is bounded by the current safety analysis.

II.3.2 Short-Term Loss of Coolant Accident Mass and Energy Transfer Rate Analysis

The licensing basis safety analysis related to the short-term mass and energy transfer rates for a containment response analysis were evaluated to determine the effect of a 1.67% power uprate. The short-term mass and energy transfer rates are defined as the blowdown phase of a loss-of-coolant accident in which the volume of water in the reactor coolant system is exiting as a steam and water mixture. The evaluation determined that the analysis was performed using a core power of 1560 MWt as shown in Section 3 of Reference II.3.1. (Note: the Westinghouse proprietary computer code CEFLASH-4A was used to perform the short-term steam line LOCA mass and energy releases. It was approved by the NRC in Reference II.3.2). Therefore, the power uprate is bounded by the current safety analysis. The analysis that performed this calculation was an input into the long-term mass and energy calculations for a containment response analysis.

II.3.3 Long-Term Loss of Coolant Accident Mass and Energy Transfer Rate Analyses

The licensing basis safety analysis related to the long-term mass and energy transfer rates for a containment response analysis were evaluated to determine the effect of a 1.67% power uprate. The long-term mass and energy transfer rates are defined as the time period after the initial blowdown of the reactor coolant system for a loss-of-coolant accident. The evaluation determined that the analysis was performed using a core power of 1560 MWt as shown in Section 3, card 82 of Reference II.3.1. (Note: The Westinghouse proprietary computer code CONTRANS was used to perform the long-term LOCA mass and energy releases. It was approved by the NRC in Reference II.3.3). Therefore, the power uprate is bounded by the current safety analyses.

II.3.4 Containment Pressure Analysis

The licensing basis containment response safety analyses for a MSLB and LOCA are not directly affected by the MUR power uprate. This is due to the containment response analyses being driven by the short and long term mass and energy transfer rates. These transfer rates were evaluated to determine the effect of a 1.67% power uprate and are bounded by the current analyses. Therefore, the containment pressure analyses are acceptable.

II.3.5 Post-Loss of Coolant Accident Containment Hydrogen Generation

The FCS post-LOCA containment hydrogen generation analysis of record was discussed in section II. It was documented in USAR 14.17 – "Generation of Hydrogen in Containment" that

this analysis is not directly related to power level and is therefore not affected by MUR power uprate.

References (Section II.3):

II.3.1 Letter LTR-OA-01-13, Revision 0, from WEC (M. J. Gancarz) to OPPD (J. Jensen), "Transmittal of Containment Licensing Basis LOCA and MSLB Mass and Energy Release Data for Fort Calhoun Station", dated September 14, 2001.
II.3.2 NUREG-75/112, "Safety Evaluation Report related to the preliminary design of the Standard Reference System CESSAR System 80," December 1975.
II.3.3 Letter from NRC (O. D. Parr) to CE (A. E. Scherer). This letter details the NRC approval of CENPD-140-A, "Description of the CONTRANS Digital computer Code for Containment Pressure and Temperature Analysis."

II.4 Non LOCA Analyses and Flooding

II.4.1 Non-LOCA Analyses

The non-LOCA design basis events are documented in USAR Sections 14.1 through 14.14, and 14.16 through 14.24. None of these events, required re-analysis to demonstrate that the acceptance criteria will still be met at the 1.67% uprated power conditions. See Section II for all non-LOCA Analyses.

II.4.2 Flooding

Protection from flooding is maintained by specific features for FCS that are not affected by changes associated with a MUR uprate program. These features include how the plant is sited in relationship to the Missouri River, component pump and piping location, and intake structure location References II.3.1, II.3.2 and Figure 9.8-1 of II.3.2.

The MUR power uprate does not affect features associated with leakage detection and isolation, or the frequency of natural events such as seiche. The MUR power uprate does not affect any of the analysis that have been performed with respect to flooding impacts and frequency (References II.3.1 and II.3.2). Current service water, component cooling water, fire protection, circulating water, raw water piping and pump configurations do not need modification in order to implement MUR power uprate. Therefore, the leakage conditions with the maximum flood potential that have been evaluated are not impacted by the proposed power uprate. Therefore, the changes associated with the MUR uprate program do not impact flooding analyses previously evaluated.

References (Section II.4):

II.4.1 USAR 2.7
II.4.2 USAR 9.8

II.5 Design Transients

II.5.1 Nuclear Steam Supply System Design Transients

The design transients and associated frequencies presented in the component specifications are used to calculate the thermal fatigue usage factors. Thermal fatigue is a function of temperature and pressure changes on the component. The design specifications for the Reactor Vessel, Steam Generator, Reactor Coolant Pumps, Pressurizer and Reactor Coolant System Piping are the components specifically addressed by the design transient evaluation. These same structural design transients are used in the design specification for the Control Element Drive Mechanism (CEDM). The continued applicability of the structural design transients for the CEDMs was also evaluated.

The structural design transients establish pressure and temperature criteria for the design specifications of plant components and appurtenances as described in Section III of the ASME Boiler and pressure Vessel Code (Reference II.4.1.2, NA-3250). The following transients are listed in the Reactor Vessel Engineering Specification (Reference II.4.1.3), Control Element Drive Mechanism Design Specification (Reference II.4.1.6), RCS Piping Engineering Specification (Reference II.4.1.7) and Pressurizer Engineering Specification (Reference II.4.1.8):

- Heatup, 100° F/hr
- Cooldown, 100° F/hr
- Loading, 10%/min
- Unloading, 10%/min
- Step Load Increase, 10%
- Step Load Decrease, 10%
- Reactor Trip
- Hydrostatic test
- Leak test
- Normal Plant Variations
- Loss of Flow (abnormal condition, does not form bases for design)
- Loss of Load (abnormal condition, does not form bases for design)
- Loss of Secondary Pressure (abnormal condition, does not form bases for design)

The Reactor Coolant Pump (RCP) Engineering Specification (Reference II.4.1.5) includes transients listed above and starting and stopping RCPs.

The Steam Generator Engineering Specification (Reference II.4.1.4) includes transients listed above and:

- Starting and Stopping RCPs
- Secondary Side Hydrostatic Test
- Secondary Side Leak Test
- Cold Feedwater Following Hot Standby

Loss of Feed Flow (abnormal condition, does not form bases for design of the vessel)

The transients that are potentially affected by power uprate were rerun for MUR power uprate conditions. The results confirm that the current transients in the component specifications remain valid and sufficiently conservative to insure the original OPPD plant life at the higher power level associated with the MUR power uprate.

The expected changes to important plant parameters are based on MUR power uprate conditions. The list of revised plant parameters infers many plant parameters are not expected to change. The evaluation assumed that the following plant parameters would not be affected by the Appendix K uprate (this assumption is valid since pressure, T_c temperature and no load temperature condition as well as flow rate remain the same for MUR power uprate):

- No-load RCS temperature (532 °F).
- The liquid volume in the pressurizer at full power (500 ft^3).
- The flow rate in the primary system (102,308 gpm per loop).
- Normal Primary System Pressure (2100 psia).
- Full power Cold leg temperature (543 °F).

The following MUR power uprate full power values were revised:

- Reactor power will be 101.67% of the current reactor power (1525 MWt).
- Feedwater temperature (443.6 °F).
- Hot leg Temperature (594.1 °F).

Certain transients are not dependent on reactor power. Due to the fact that the no-load plant conditions have not changed, transients related to subcritical and no-load operations are not affected by the power uprate. Specifications based on the following transients remain applicable for the MUR power uprate and no further evaluation is required.

- Heatup,100 °F/hr
- Cooldown, 100 °F/hr
- Hydrostatic Test
- Leak Test
- Normal Plant Variations* (±100 psi, ± 6 °F assumed values unrelated to power, References II.4.1.3-8)
- Loss of Secondary Pressure (Limiting case is based on no-load conditions, see Reference II.4.1.9)
- Starting and Stopping RCPs
- Secondary Side Hydrostatic test
- Secondary Side Leak Test
- Cold Feedwater Following Hot Standby

*Note that the pressurizer assumes ± 7 °F for Normal Plant Variations and the SG secondary side assumes ± 40 psi for Normal Plant Variations.

The original plant specifications are based on a core licensed for 1420 MWt. Following the MUR power uprate the licensed thermal power will be 1525 MWt. The evaluation includes the stretch power to 1500 MWt and the 1.67% increase to the MUR power uprate conditions. Even

after accounting for a power increase of nearly 7.5%, the transients generated with the LTC code are in very good agreement with the transient results presented in the component specifications. This is not an unexpected result. The increase in reactor power will have an effect on the stored energy in the core and the RCS ΔT ($T_{hot} - T_{cold}$). However, the initial design transients conservatively assumed higher full load RCS temperatures that continue to bound the projected MUR power uprate values. As a result, the increase in reactor power does not present a significant change to the design transients. The core decay heat will also be slightly higher after a reactor trip. However this is only a small percentage of the power increase and the Turbine Bypass System can easily compensate for increases in decay heat.

The changes in the transient responses are more related to operational changes than increased power. The current practice is to operate with all rods out. When the original transients were run rods were slightly inserted in the core so any motion provided instant feedback. Operating with all rods out introduces a delay before rod motion effects power and temperature changes. Another change directly related to operating with all rods out is the rate of power change during controlled evolutions. The 10% per minute or 10% step power changes relate more to control system design. Operationally, power changes are limited to the rate of changing the RCS boron concentration. This means that the temperature and pressure deviations associated with actual power changes are smaller than observed in the transient analysis.

References (Section II.5.1):

II.5.1.1 "Nuclear Services Policies & Procedures," WP-4.5 Revision 4, "Design Analysis," effective 10/01/01
II.5.1.2 ASME Boiler and Pressure Vessel Code Section III, 1971 Edition.
II.5.1.3 Specification No. 750S-23-1, Revision 2 through Mod. 7, "Engineering Specification for a Reactor Vessel Assembly for Omaha Public Power District Fort Calhoun Station", 12/5/72.
II.5.1.4 Specification No. 750S-23-2, Revision 2 through Mod. 6, "Engineering Specification for a Steam Generator Assemblies for Omaha Public Power District Fort Calhoun Station", Dec. 1972.
II.5.1.5 Specification No. 750S-23-3, Revision 2 through Mod. 6, "Engineering Specification for a Reactor Coolant Pumps for Omaha Public Power District Fort Calhoun Station", 12/31/69.
II.5.1.6 Specification No. 23866-486-311, Revision 3, "Design Specification for a Control Drive Mechanism for Omaha Public Power District Fort Calhoun Station", 8/30/71.
II.5.1.7 Specification No. 750S-23-5, Revision 5, "Project Specification for Primary Coolant Pipe and Fittings for Omaha Public Power District Fort Calhoun Station", 2/15/99.
II.5.1.8 Specification No. 750S-23-4, Revision 3, "Engineering Specification for a Pressurizer Assembly for Omaha Public Power District Fort Calhoun Station", 2/21/94.
II.5.1.9 Letter O-SE-127 to M. Puchir (OPPD) from E. A. Montanaro, "Fort Calhoun Thermal Transients" dated February 16, 1968.

III. Accidents and Transients for which the Existing Analyses of Record do not Bound Plant Operation at the Proposed Uprated Power Level

All existing analyses of record for transients and accidents identified in USAR Chapter 14 bound (or are dispositioned for) the proposed MUR power uprate. There were however, a few systems analyses and a cooldown analyses for Appendix R program that were required to be re-analyzed for MUR power uprate conditions. A discussion of the re-analysis is provided next.

III.1 Systems Reanalyzed

III.1.1 Spent Fuel Pool Cooling System (SFPCS)

The only potential impact to the SFPCS resulting from the MUR power uprate program is the amount of additional decay heat resulting from operating at higher power. Existing analyses assumed 1500 MWt for the decay heat calculations (Reference III.1.1.1). The NRC approved the license amendment for spent fuel rack expansion based on the existing spent fuel pool cooling system (Reference III.1.1.2). The revised analysis used the same methodology for pool cooling evaluations as previously approved by the NRC in Reference III.1.1.2. There were no changes in methodology for pool cooling evaluations than what was previously approved in Reference III.1.1.2. The only change was the input power level.

The spent fuel pool cooling analyses were revised to accommodate the MUR uprate program. The results of the revised analyses which were performed at 1530 MWt indicated only a slight increase (approximately 3° F) in bulk pool temperature for a full core discharge condition (end of plant life). Therefore, acceptable pool temperatures are maintained at the 1.67% power uprate conditions. The peak pool temperature after full core discharge offload (end of plant life) is anticipated to be approximately 138 °F, which is still below the 140 °F limit in the USAR and below the SRP Section 9.1.3 guidance for normal and abnormal discharges to the SFP. Therefore, the guidance of the SRP is met with regard to providing adequate cooling for the postulated spent fuel inventory under normal or abnormal operating conditions.
The USAR states it would take 7.2 hours for the SFP to heat from 134.9 °F to boiling (at 1500 MWt rated power). To conservatively determine the SFP heat up times for the 1.67% power uprated operation, the calculated maximum SFP heat loads for the current and 1.67% power uprate conditions are compared to the USAR heat load and heat-up rate values to determine the expected SFP heat up rate and the time to reach boiling.

The minimum time to SFP boil is calculated to be 6.5 hours for the 1530 MWt full core discharge case. This decrease in time is not significant compared to the time available to take action. In addition, the heat loads to develop this time are very conservative and did not credit any decay of heat load (Reference III.1.1.1). Therefore, minimum times for the SFP to boil are acceptable for the 1.67% power uprate operating conditions.

FC-5988, Rev. 1, states that the local temperatures in a fuel cell were calculated based on the power level of 0.3847×10^8 Btu/hr (1500MWt/133 assemblies). For non-blocked and blocked fuel cell locations the maximum local water temperature was 207.9 °F and 221.2 °F respectively. The maximum local fuel cladding temperature for non-blocked and blocked cell locations was calculated to be 254.9 °F and 264.9 °F respectively. The local temperature cases are based on the original analysis fuel rod radial peaking factor, maximum pin peak, and maximum core axial peak (total peaking factor of 3.067, which includes a 1.15 safety factor). For the MUR power

uprate scenario it will be assumed that the total peaking factor and all other inputs remain unchanged. As such, the local temperatures would strictly be based on the peak heat load discharged to the pool. The design basis delta temperature would then increase by a factor of 1.02 for 1530 MWt power configuration.

The resulting maximum local water temperatures for a 1530 MWt power configuration for blocked and non-blocked fuel cell locations would be 212.1 °F and 225.6 °F respectively. The maximum local fuel cladding temperature for non-blocked and blocked cell locations would be 260 °F and 270.2 °F. No nucleate boiling is indicated at any location. No excessive stress at the clad surface will occur as a result of the MUR power uprate configuration.

References (Section III.1.1):

III.1.1.1 Letter to NRC Document Control Desk, LIC-92-0340A, "Application for Amendment of Operating License", December 7, 1992.

III.1.1.2 NRC Safety Evaluation Report Docket No. 50-285, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 155 to Facility Operating License No. DPR-40 Omaha Public Power District Fort Calhoun Station, Unit No. 1", August 12, 1993.

III.2 Programs Analysis Reanalyzed

III.2.1 Fire Protection/Appendix R Programs

The Fire Protection system provides means for detecting, alarming, isolating, and suppressing plant fires. The system is divided into the following subsystems:

- The Fire Detection and Alarm System
- The Fire Suppression System including automatic sprinklers, deluge systems, portable fire extinguisher, automatic carbon dioxide and halon systems, standpipe hose systems, and outside fire hydrants.

The Fire Protection system is not affected by the 1.67% MUR power uprate and will perform its design function at the uprated power conditions.

The FCS Appendix R program meets all the requirements of 10 CFR 50 Appendix R by the successful performance of the following safe shutdown functions:

- Reactivity Control
- Reactor Coolant System Inventory Control
- Reactor Coolant System Pressure Control
- Reactor Heat Removal / Secondary Side Integrity (hot shutdown, cold shutdown)
- Essential Electrical Support
- Essential Mechanical Support
- Plant Monitoring Instrumentation

The review of the FCS Appendix R Safe Shutdown analysis and supporting calculations concluded that the FCS safe shutdown capability documented in the Appendix R program will not be affected by the 1.67% MUR power uprate. The requirement to reach shutdown

conditions within 72 hours is met with the increased decay heat resulting from the higher thermal power (1525 MWt). Additionally, the installation of the feedwater flow measuring instrumentation that affects the MUR power uprate will be implemented in accordance with the station's design process and will not adversely impact the Appendix R safe shutdown analysis.

References (Section III.2.1):

III.2.1.1 USAR Section 9.11
III.2.1.2 EA-FC-89-055 "10 CFR 50 Appendix R Safe Shutdown Analysis"
III.2.1.3 Calculation FC06355 "10 CFR 50 Appendix R Functional Requirements and Component Selection"
III.2.1.4 Calculation FC06669 "Heat Removal Process Paths to Maintain RCS Temperature Below 300°F for the Fort Calhoun Station"

IV. Mechanical/Structural/Material Component Integrity and Design

IV.1 Reactor Vessel Structural Evaluation

The FCS reactor vessel was evaluated for impact due to the MUR uprate program. This section will address the evaluations that were conducted for assessment of reactor vessel at MUR power uprate conditions.

IV.1.1 Reactor Vessel Integrity

IV.1.1.1 Reactor Vessel Integrity - Neutron Irradiation

Reactor Vessel Fluence Assessment for Measurement Uncertainty Recovery (MUR) Uprate

10CFR 50.61 (the PTS Rule) requires that for the reactor vessel beltline region that the end-of life (EOL) value of RT_{PTS} be less than 270°F for plates, forgings, and axial weld materials and 300°F for circumferential weld materials.

WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel", dated July 2000 (Reference IV.1.1) represents the fast neutron fluence analysis for FCS and provides projections to 48 EFPY. This analysis assumes an ongoing capacity factor of 85% and documents the projected radial fluences. CEN-636, Revision 02, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials", dated July 19, 2000 (Reference IV.1.2) documents the chemistry factors for each plate and weld and concludes that the 3-410 weld comprised of weld wire heats 13253/12008 is the limiting material for FCS. This material is the most limiting, because unlike other weld combinations, no surveillance data exists for it and Position 2.1 of Regulatory Guide 1.99, Rev.2 can not be applied to reduce the Margin Term from 65.5°F to 44°F.

An assessment was conducted, to increase the fast neutron flux by 1.67% to account for the MUR Power Uprate with the same fuel management currently in use, the future Refueling Outage schedule used/projected, and the capacity factor increased to 94%. The results of this assessment show that the RT_{PTS} for the limiting reactor vessel material (269.80°F) increases by approximately 0.66°F when compared to the previous value (269.14°F) at 48 EFPY (with an

85% capacity factor), but remains below 270°F, thus demonstrating that the increased fluence rate for the MUR power uprate and increased capacity factor are acceptable and bound MUR power uprate.

IV.1.1.2 Reactor Vessel Pressure-Temperature Limits

Per 10 CFR 50 Appendix G, "Fracture Toughness Requirements," the P-T limit curve must be updated prior to the end of the applicable fluence period. The current P-T limit curves are valid to 40 EFPY (Reference IV.1.3) of reactor operation, which is based on the following:

Neutron fluence accumulation of 2.15×10^{19} n/cm^2.
Limiting weld is the 3-410 axial weld consisting of weld wire heat number 12008/13253.

The fluence value used in the P-T limit curve was based on a "best estimate" fluence analysis (Reference IV.1.4) that must be adjusted by the increase in fast neutron flux by 1.67% which is the increase due to the MUR power uprate. This adjustment has been completed by the evaluation "Reactor Vessel Fluence Assessment for Measurement Uncertainty Recovery Uprate" in Reference IV.1.5. Based on this evaluation, the fluence value of 2.15×10^{19} n/cm^2 for the limiting 3-410 weld is expected to occur at 39.9 EFPY, which reduces the P-T limit curve applicability to 39.9 EFPY versus 40.0 EFPY. 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 (Reference IV.1.6).

IV.1.1.3 Upper Shelf Energy (USE)

Per 10 CFR 50 Appendix G, "Fracture Toughness Requirements," the USE for the reactor vessel beltline components in the transverse direction must be maintained greater than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Reference IV.1.7, provides information about the limiting reactor vessel beltline material. The 1.67% MUR power uprate causes an insignificant increase to the fluence at the 1/4 T location for this weld. The predicted USE decrease per Figure 2 of Reference IV.1.8 remains essentially unchanged; therefore, the USE remains above the regulatory limit of 50 ft-lbs.

IV.1.1.4 Surveillance Capsule Withdrawal Schedule

10CFR50 Appendix H (Reference IV.1.9) defines the reactor vessel surveillance program that is to be used by the licensee to monitor the neutron radiation induced changes in fracture toughness of the vessel during the life of the plant. It includes requirements to establish a surveillance capsule withdrawal schedule. The schedule was updated (Reference IV.1.10) to reflect changes to the program. Therefore, the updated surveillance capsule withdrawal schedule is also applicable under conditions including the Appendix-K power uprate.

IV.1.2 Reactor Internals

IV.1.2.1 Reactor Internals Structural Evaluations

It was determined that of the major RVI components only the Core Shroud would potentially be adversely affected by the increased thermal loadings associated with power uprate. A structural evaluation of the Core Shroud was performed for MUR power uprate conditions. This evaluation documented in Attachment 7 concludes that for the Core Shroud that the stresses were determined to be acceptable for MUR power uprate conditions. A fatigue evaluation was also performed for the evaluated components and determined to be acceptable. See Attachment 7 for detailed internals structural evaluation.

IV.1.2.2 Reactor Vessel Internals Materials Evaluation

During service, RVI components are exposed to a high temperature aqueous environment, fast neutron irradiation and applied loads. Prolonged exposures to these conditions will result in changes to the mechanical and corrosion properties of the RVI component materials and may result in age related degradation of the materials. The planned power uprate will result in increases in lifetime neutron fluences received by the RVI components, primary coolant temperature and metal temperatures of components nearest the core as a result of increased gamma heating. These condition changes may result in additional degradation of the RVI component materials from the various age related degradation mechanisms (ARDMs) that potentially could affect the materials. An evaluation was conducted to evaluate potential changes in the FCS RVI materials as a result of the MUR power uprate condition.

The component groups evaluated were:

- Upper internals assembly
- CEA shroud assemblies
- Core support barrel assembly
- Core shroud assembly
- Lower internals assembly
- In core instrumentation system

The RVIs were fabricated from various wrought austenitic stainless steels, cast austenitic stainless steels and other stainless steels. The predominant RVI material was Type 304 SS. Other grades included type 304L and type 316 were also used. Most of the internals bolting applications, such as the core shroud panel to former plate bolts, were fabricated from annealed type 316 SS. The CEA shroud bolts were fabricated from A-286, an austenitic, precipitation hardened stainless steel. The hold down ring was type 403, a martensitic stainless steel.

ARDMs for RVI components such as the following were evaluated with respect to MUR power uprate conditions:

- Irradiation embrittlement
- Fatigue
- Corrosion (general, intergranular, pitting, crevice)
- Stress corrosion cracking (SCC)
- Irradiation assisted stress corrosion cracking (IASCC)

- Flow assisted corrosion (erosion corrosion)
- Thermal aging
- Creep and stress relaxation
- Mechanical Wear

The evaluations concluded that the FCS MUR power uprate would have no effect or a negligible effect on the following ARDMs:

- Fatigue
- Corrosion (general, intergranular, pitting, crevice)
- Flow assisted corrosion (erosion corrosion)
- Creep and stress relaxation
- Mechanical Wear

The evaluations also concluded irradiation induced embrittlement will not increase significantly as a result of the uprate. The peak estimated end of life fluence will increase by only 2×10^{20} n/cm^2) in the core shroud. The core shroud is the most limiting component in terms of maximum fluence.

Susceptibility of RVI bolts to IASCC due to MUR power uprate was also evaluated. It was concluded that IASCC could not be eliminated as an ARDM for the core shroud panel to former plate bolts, but there is not a significant increase in susceptibility to IASCC arising from MUR power uprate.

The potential for SCC of FCS RVI components is low and MUR power uprate will have a negligible effect on the potential for this ARDM. There will be only a minor increase in coolant and metal temperatures (temperature is a key parameter for SCC). FCS used annealed austenitic SS with controls imposed during the original fabrication, primary coolant chemistry controls, coolant flow patterns and low operational stress levels that continue to insure that SCC remains a low potential ARDM for all austenitic SS components. The FCS CEA shroud bolts are fabricated of A-286 material, but have differences in material with respect to less severe cold working prior to forging, rolling of threads after final heat treatment and inducing compressive stresses. Due to these measures SCC of A-286 CEA shroud bolts in FCS remain a low potential event, even after MUR power uprate.

Stress relaxation of RVI bolting components will not increase significantly as a result of power uprate. Stress relaxation from thermal and neutron irradiation will continue and thus, will be monitored through current inspections methods in order to manage this ARDM.

Embrittlement of CASS components as a result of thermal aging and neutron irradiation will not be significantly affected by the power uprate because changes to the temperature, neutron fluence and applied loads will be small. To date, there have not been any effects of thermal or irradiation Embrittlement of CASS RVI components detected in any CE plants or in other PWRs.

In summary the MUR power uprate will not have a significant impact on ARDMs that could potentially affect the FCS reactor vessel internals.

IV.1.2.3 Thermal-Hydraulic Systems Evaluations

CEA Drop Test Time Assessment for Measurement Uncertainty Recovery (MUR) Uprate

Technical Specification 2.10.2(8) requires that the CEA drop time be less then or equal to 2.5 seconds from the time the clutch coil is deenergized until the CEA reaches 90% insertion. Per OP-ST-CEA-0006, measurements of CEA drop time are made prior to criticality at the beginning of each fuel cycle, with T_{cold} greater or equal to 215 °F and all reactor coolant pumps operating. Step 9.1 of this procedure also requires that the CEA drop times be less than or equal to 2.5 seconds. The Accident Analyses assume a CEA drop time of 3.1 seconds, thus ensuring additional margin between the analysis value and the Technical Specification/Surveillance Test limit.

Surveillance test (ST) results for FCS have shown the average drop time to be approximately 1.5 seconds (see CE-NPSD-1049-P, Rev.01, "Potential for Delayed CEA Insertion Times at CE Designed Plants", June 1996), thus providing significant margins to the TS and ST acceptance criterion of less than or equal to 2.5 seconds and the Accident Analysis value of 3.1 seconds.

Previous experience with the power uprate from 1420 MWt to 1500 MWt in 1980 showed no noticeable change in CEA drop times, which supports a conclusion that a MUR power uprate of up to 2% from 1500 to 1530 MWt will also have no noticeable adverse effect on CEA drop times.

Finally, with T_{cold} remaining constant, small increases in T_{hot} and T_{avg} will result from the thermal power increase. The resultant effect is a slight decrease in coolant density. This slight decrease in coolant density would be expected to beneficially have a slightly lower resistance to a dropped CEA, although the change in drop time would probably not be measurable (consistent with the uprate from 1420 MWt to 1500 MWt). This further supports the conclusion that there will be no noticeable adverse effect on CEA drop times for the MUR power uprate and the CEA drop times will continue to be consistentconsistent with TS 2.10.2(8).

References (Section IV.1)

IV.1.1 WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel", dated July 2000.
IV.1.2. CEN-636, Revision 02, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials", dated July 19, 2000.
IV.1.3 EA-FC-01-022, Rev. 0, "Pressure and Temperature Limit Curve for 40 EFPY."
IV.1.4 WCAP-15443, Rev. 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel."
IV.1.5 EA-FC-02-028, Rev. 0, "Appendix K Power Uprate Evaluation."
IV.1.6 Letter from OPPD (D. J. Bannister) to NRC (Document Control Desk), "Fort Calhoun Station Unit No. 1 License Amendment Request, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)"," dated October 8, 2002. This letter is contained in LIC-02-0109.
IV.1.7 EA-FC-01-024, Rev. 0, "Upper Shelf Energy (USE) Evaluation for 2033 Operation."
IV.1.8 Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials."
IV.1.9 Appendix H to 10 CFR Part 50, "Reactor Vessel Materials Surveillance Program Requirements", Federal Register, Volume 60, No. 243, dated December 19, 1995.

IV.1.10 "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications, Fort Calhoun Station, Omaha Public Power District," Westinghouse Report WCAP-15741, dated September 2001.

IV.2 Piping and Supports

IV.2.1 Nuclear Steam Supply System Piping

The Reactor Coolant System (RCS) is designed to remove heat from the core and internals and transfer it to the secondary side of the steam generators. The RCS also serves as a barrier to the release of radioactive material to the containment building.

The RCS consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains one steam generator, two reactor coolant pumps, connecting piping, valves and instrumentation. A pressurizer is connected to one of the reactor vessel hot leg pipes by a surge line and the pressurizer relief and safety valves discharge to the quench tank.

The RCS system components adhere to the following Boiler and Pressure Vessel Codes:

Reactor Vessel	ASME Section III, Class A
Steam Generator Primary Side	ASME Section III, Class A
Steam Generator Secondary Side	ASME Section III, Class A
Pressurizer	ASME Section III, Class A
Coolant Pumps	ASME Section III, Class A
Quench Tank	ASME Section III, Class C
Pressurizer Safety and Relief Valves	ASME Section III
Piping	ASME Section III, and USAS B31.1.

At the MUR power uprate conditions the reactor thermal power will be changed from 1500 MWt to 1525 MWt, reactor coolant pressure and inlet temperature (T_c) remain unchanged at 2100 psia and 543°F, respectively. The hot reactor coolant temperature (T_h) changes from 593.3°F to 594.1°F and the average temperature (T_a) changes from 568.2°F to 568.6°F.

At the increased thermal power the RCS flow remains unchanged since the cold leg temperature remains unchanged. The RCS design temperature and pressure of 650°F and 2500 psia remain unchanged and the pressurizer design temperature and pressure of 700°F and 2500 psia remain unchanged. The pressurizer relief requirements increase slightly due the additional decay heat. The change, however, is bounded within the relieving capacity of the pressurizer safety valves for the design transient conditions.

The increased thermal power, T_h and T_a temperatures affect the cumulative fatigue usage of the following RCS components:

- Reactor Vessel Nozzles
- Steam Generator Primary and Secondary Sides Nozzles
- Pressurizer Nozzle
- Coolant Pumps
- Piping

The above RCS components were designed so that the cumulative fatigue usage for all transients over the life of the plant does not exceed 1.0. The review of the cumulative fatigue usage evaluation has concluded that the 1.67% increase in thermal power and the 0.8°F increase in T_h at the MUR power uprate conditions will increase fatigue usage by an insignificant amount. With this insignificant increase the cumulative fatigue usage shall not exceed 1.0 for the life of the unit.
The RCS was designed to load combination criteria that assure the integrity of all components in the system. A review of the evaluations of the design loads, expansion stresses, allowable stresses on RCS components during normal as well as accident conditions has concluded that the changes in the design loads and stress allowables, at the MUR power uprate conditions, will be insignificant.

The effect of the 0.8°F increase in T_h on loads, stresses and allowable stresses on reactor vessel nozzles, reactor vessel supports, and reactor core support structures were evaluated and found to be insignificant. Since the RCS system flow remains unchanged at the uprate conditions, flow induced vibrations are not affected. Additionally, the power uprate will not change the existing RCS inspection and testing programs.

With regard to stratification issues in the pressurizer surge line, an increase in hot leg temperature will reduce the temperature difference between the pressurizer and the hot leg fluid thereby reducing stratification stresses. Therefore, fatigue usage due to stratification will decrease as a result of the MUR power uprate (Reference IV.2.3).

At the MUR power uprate conditions the primary side inlet conditions of the Steam Generators will experience a 0.8°F increase in temperature. This increase will affect the Steam Generator's cumulative fatigue usage, normal and transient loads, stresses and allowable stresses. The evaluation of the impact of the increased temperature concluded that it is insignificant and is bounded by the analysis of record.
The steam flow and feedwater flow in the secondary side of the Steam Generator will be changed at the MUR power uprate conditions:

Steam Flow from 3.322 E6 to 3.386 E6 lb/hr per Steam Generator
Feedwater Flow from 3.322 E6 to 3.386 E6 lb/hr per Steam Generator

The impact of the change of the above parameters on the Steam Generator and steam line piping, on normal, as well as transient loads, are discussed in the Steam Generator section.

The Steam Generator safety valves that provide overpressure protection for the shell side of the steam generators and the main steam line piping have been designed with adequate relief capacity to accommodate the MUR power uprate conditions.

At the MUR power uprate conditions, the reactor coolant system average temperature T_a will increase from 568.2°F to 568.6°F. The review of the pressurizer level control program has concluded that the average coolant temperature at the MUR power uprate conditions is within the bounds of the reactor coolant average temperature and therefore, does not affect the pressurizer level control program. It is therefore concluded that, the reactor coolant volume control function is not affected by the uprate.

The capacity of the power operated relief valves is sufficient to ensure that the pressurizer safety valves do not open during the MUR power uprate conditions since their relief capacity

was originally designed to preclude the opening of the safety valves during a loss-of-load incident from 102.4% power.

The capacity of the pressurizer safety valves is sufficient to limit the reactor coolant pressure to 110% of design pressure following a complete loss of turbine generator load. This analysis has been performed at 102.4% power. Therefore, the effect of the 1.67% MUR power uprate on the pressurizer safety valves is bounded by the existing analysis.References (Section IV.2):

IV.2.1 USAR Section 4
IV.2.2 Design Basis Document SDBD-RC-128 "Reactor Coolant"
IV.2.3 Calculation FC06896 "Evaluation of Appendix K Uprate on Structural Integrity of the Reactor Coolant System"
IV.2.4 Stone & Webster "Power Recovery and Power Upgrade Evaluation Report", 1993
IV.2.5 Calculation FC06627 "Inlet Pressure Drop to Main Steam Safety Valves"

IV.2.2 Reactor Coolant Loop Support System

The steam generator, reactor coolant pump, reactor vessel, and pressurizer supports have been qualified for piping and component loads. Since the MUR Uprate Program does not change the loads exerted upon the support structures, the supports will continue to be qualified for the 1.67% power uprate condition.

IV.2.3 Leak-Before-Break Analysis

By references IV.2.3.1 and IV.2.3.2, the NRC approved FCS's use of the LBB methodology. The LBB analyses justified the elimination of large primary loop pipe rupture from the structural design basis for FCS Unit 1. To demonstrate the continued acceptability of the elimination of RCS primary loop pipe rupture from the structural design basis for the MUR power uprate program, the following objectives must be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate.
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.
- Demonstrate margin on applied load.
- Demonstrate that fatigue crack growth is negligible.
- These objectives were met by the analyses discussed in References IV.2.3.1 through IV.2.3.3.

There is no change in loads on the primary loop piping due to the uprating parameters. The effect of material properties due to the changes in temperature were bound by the Westinghouse analysis which was conducted at higher RCS loop temperatures and pressures, thus, the change in temperature will have a negligible impact on the existing LBB analysis margins. Reference IV.2.3.1 was based on enveloped design loads of axial tension-1,800k, bending moment 45,600 in.-k, RCS pressure of 2,250 psi, and RCS temperature of 600 °F. The enveloped nozzle loads for FCS were reported as axial load-1,650k, and bending moment 9,800 in.-k. FCS operates at a much lower RCS temperature and pressure, thus, even with an increase in RCS temperature of 0.8°F, the LBB evaluated enveloped design loads still provide adequate margin in regards to crack stability conclusions. The thermal expansion stresses

increase slightly because of the slight temperature increase; however the generic loads will bound the uprate loads.

The previous LBB leak detection capability for radiation monitoring was based on a 1500 MWt core inventory, and TID source term. With implementation of AST source term at 1530 MWt an assessment was made to ensure that leak detection margins were still met. The conclusions of that assessment indicated that LBB leak detection capability was not impacted by the power uprate condition.
Therefore, the existing LBB analyses and revised radiation monitoring analysis conclusions remain applicable for FCS MUR Uprate program.

References (Section IV.2.3):

IV.2.3.1 WCAP-9558, Revision 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack", Westinghouse Energy Systems, May 1981.
IV.2.3.2 Letter, USNRC (Steve Bloom) to OPPD (T. L. Patterson), "Fort Calhoun Station, Unit No. 1 – Amendment No. 165 to Facility Operating License No. DPR-40 (Tac No. M85848), NRC-94-246, August 25, 1994.
IV.2.3.3 Calculation "FC-05462 Rev. 8, Response Time of Containment Air Monitoring System", May 9, 2003.

IV.3 Control Element Drive Mechanisms

The CEDMs are subjected to hot leg temperatures and RCS pressures. There is no change to the maximum operating reactor coolant pressure of 2250 psia (which bounds operation at 2100 psia). These are the only NSSS design parameters considered in the CEDM evaluation.

Higher temperatures are more limiting for the CEDM structural design qualification because it results in a decrease in the margin to the allowable design stress limits. The maximum T_{hot} from the MUR Uprate Program NSSS design parameters (Table 3) for any case is 594.1 °F. Furthermore, the possible RCS operating pressure values continue to remain at 2100 psia for the MUR Uprate Program.

CEDM Grease Hardening

The FCS CRDs have operated as designed and have not experienced any failures. CEDM inspections have not observed any grease hardening. This is attributed to the fact that CEDM mechanical seal leak off average temperatures have been relatively low less than approximately 106 °F. Plants operating with seal leak off temperatures greater than 130 °F have observed grease hardening in the CEDMs.

At the MUR power uprate conditions, T_{hot} will increase by approximately 0.8°F. This temperature increase is expected to have an insignificant impact on leakoff temperatures and the operation of CEDMs (Reference IV.3.1).

Reference (Section IV.3):

IV.3.1 INPO OE 10612-Control Rod Failure to Trip (Palisades)

IV.4 Reactor Coolant Pumps and Motors

The RCPs and RCP motors were evaluated to determine the impact of the revised RCS conditions to demonstrate that the RCP structural integrity is not adversely affected.

Reactor Coolant Pumps

The RCPs are located between the steam generator outlet and reactor vessel inlet in the reactor coolant loop. From Table 3 as noted the RCS cold leg temperatures will not be affected by the MUR power uprate. Thus, the temperature of the fluid in the cold leg in the RCP will remain the same as current conditions. The pressure of the coolant will also be the same. The RCS flow will remain the same through the RCP. Thus, the overall temperature, pressure and flow of RCS through the RCPs remains unchanged, and the therefore, the MUR power uprate conditions will not impact the existing RCPs.

Reactor Coolant Pump Motors

The FCS has four reactor coolant pumps that are powered by four synchronous 1200 rpm 3650 horsepower motors. Based on the motor's efficiency and the current operating conditions 100% power reactor coolant pump KW load data, the reactor coolant pump motors are operating at approximately 69% of their capacity. At the 1.67% MUR power uprate conditions, the reactor coolant flow and inlet temperature will not change significantly and the motor horsepower loading will not change significantly. Therefore, the reactor coolant pump motors will perform their design function with an approximately 31% margin in the horsepower loading (References IV.4.2 and IV.4.3).

References (Section IV.4):

IV.4.1 Section 4.3.5
IV.4.2 General Electric Motor Curves for motors RC-3A, 3C, 3D
IV.4.3 ABB Motor Curves for motor RC-3B
IV.4.4 Typical Reactor Coolant Pump KW Load Data measured by ERF for 100% power operation

IV.5 Steam Generators

IV.5.1 Thermal-Hydraulic Evaluation

Westinghouse evaluated two (2) sets of plant conditions using the standard Westinghouse steam generator (SG) performance codes. Evaluations were performed at the current operating conditions (1500 MWt) and the uprated plant conditions (1525 MWt).

The magnitude and importance of the changes in the secondary side thermal-hydraulic performance characteristics at the Appendix K power uprate conditions were assessed primarily in terms of circulation ratio/bundle liquid flow, damping factor, and SG pressure drop.

The Ft. Calhoun SGs were evaluated to ensure that their thermal-hydraulic performance following the Appendix K power uprate would be acceptable as compared to pre-uprate licensed design operating conditions. The magnitude and importance of the changes in the overall

secondary side thermal-hydraulic performance characteristics at the uprate conditions were assessed primarily in terms of circulation ratio, flow rates through the tube bundle, SG pressure drop, etc. A summary of the SG performance characteristics is shown in Table IV-1.

The bench mark calculations for the Ft. Calhoun SGs (designated RC-2A and RC-2B) required "fouling resistance" values of 0.0000153 and 0.0000222 hr-ft^2-°F/BTU to match the calculated steam pressures with the plant data. These values are less than 3% of the average overall resistance of 0.00072 hr-ft^2-°F/BTU. The fouling resistance represents both the effects of tube surface deposits and differences between the actual and the analytical heat transfer coefficients. The small adjustments to these values indicate a reasonably accurate simulation of the Ft. Calhoun SGs.

As a result of the 1.67% Appendix K power uprate and 6.5% SG tube plugging, the calculated SG thermal-hydraulic characteristics change as follows:

1. The SG primary inlet temperature (T_{HOT}) increases from 593.8°F to 594.6°F. This is with the assumed T_{COLD} of 543°F. Currently the measured T_{HOT} and T_{COLD} for SG RC-2A are 592.5°F and 542°F, respectively. For SG RC-2B, the measured T_{HOT} and T_{COLD} are 593°F and 542°F, respectively.
2. The primary side pressure drop is essentially the same, 35.66 psi vs. 35.64 psi.
3. The steam pressure decreases from 822.2 psia to 819.9 psia. Currently, the measured steam pressures for SG RC-2A and 2B are 818.75 psia and 810.156 psia, respectively.
4. The steam flow rate increases from 3.311 to 3.364 Mlb/hr. The current steam flow rates in SG RC-2A and 2B are 3.277 and 3.336 Mlb/hr.
5. The circulation ratio decreases from 3.87 to 3.80.
6. The secondary fluid mass inventory decreases from 83,124 lb to 82,763 lb.
7. The secondary side pressure drop increases from 37.9 psi to 39.1 psi.
8. The average heat flux increases from 57,617 to 58,593 BTU/hr-ft^2. Heat flux at the SG inlet also increases from 108,130 to 110,016 BTU/hr-ft^2.

These changes are well within the design envelope of the Ft. Calhoun SGs and demonstrate that the Appendix K power increase will not adversely affect SG performance. Accordingly, operation at 1.67% of originally licensed power is acceptable.

IV.5.2 Steam Generator Tube Vibration

<u>Fluid Elastic Instability</u>
All tubes have stability ratio values less than the allowable of 1.0, which represents the onset of unstable vibrations. Therefore, no large diverging oscillations will occur causing violent impacting between tubes as a result of fluid-elastic instability. The maximum stability ratio for the first forty mode shapes of the four critical tube locations is shown in Table IV-2.

Random Turbulent Excitation
The mid-span Root Mean Square (RMS) displacements from subcritical fluid flow as defined in Article N-1343.2 of the ASME Code are shown in Table IV-3 for the first forty mode shapes of the four critical tube locations. Also, the peak displacements are included in the table. These values are less than the design objective of 10 mils for the maximum mid-span RMS displacement and one-half the allowable gap of 125 mils for the peak displacement.

Steam Generator Tube Stabilizers
Reference IV.5.1 is the current evaluation of tube stabilizers for the Ft. Calhoun SG. For this evaluation, tube row 103 was chosen as the bounding condition since it has the longest unsupported spans of all tubes which pass through the upper support plate. The maximum stability ratio was 0.546 at a mode frequency of 62.7 Hz. This calculation was performed with very conservative cross flow gap velocities of 383.7 in/sec and 152.9 in/sec on the hot and cold side, respectively. In the thermal-hydraulic analysis of the Appendix K – 1.67% power uprate program (Reference IV.5.3), these cross flow gap velocities are more realistic at 202.6 in/sec and 95.5 in/sec on the hot and cold side, respectively.

Therefore, the maximum stability ratio of 0.546 remains bounding for the Appendix K – 1.67% power uprate. Also, from Reference IV.5.1, the maximum mid-span RMS displacement of the staked and severed tube due to subcritical fluid flow is 4.0 mils and the maximum peak displacement is 16 mils. These values are less than the design objective of 10 mils for the maximum mid-span RMS displacement and one-half the allowable gap of 125 mils for the peak displacement.

Blowdown Pipe
Reference IV.5.2, Pages A322 through A323, contains the original evaluation of bottom blowdown pipe. With the Appendix K – 1.67% power uprate, the natural frequency for the straight span and elbow bend will remain the same at 142.9 Hz and 261 Hz, respectively. Therefore, the pipe is very flexible, and the thermal load differentials will not change and would still produce only a small stress, as stated in Reference IV.5.2.

Feedwater Sparger and Sparger Supports
Reference IV.5.2, Pages A324 through A331, contains the original evaluation of the feedwater sparger and sparger supports. The temperatures in the thermal analysis of the support bracket of 270°F for body 2 and 532°F for body 1 do not change in the Appendix K – 1.67% power uprate. Also, the most extreme thermal transient of the intermittent introduction of 70°F feedwater at 600 gpm into the steam generator at 532°F in Reference IV.5.2 is still valid for the Appendix K – 1.67% power uprate. Therefore, the results in Reference IV.5.2 continue to bound those for the power uprate. Namely, the maximum primary plus secondary stress intensity range at the "L" shaped bar to the feedwater nozzle weld is 48.1 ksi, which is less than the allowable 3 S_m value of 58.8 ksi and the maximum cumulative usage factor is 0.5, which is less than the allowable of 1.0. The lowest natural frequency for the feedwater ring portion of the sparger does not change and continues to have a magnitude of 68.5 Hz. This value is well above the frequency of 19 – 20 Hz, introduced due to an imbalance of the pump impeller.

Separator Deck and Separators
The structural evaluation of the steam separator support deck and associated members was originally performed on Pages A332 through A345 of Reference IV.5.2. Three cases were considered in the analysis, which included, 1) stresses due to dead weight, 2) stresses due to steam flow at steady state, and 3) stresses due to the casualty condition defined in the

Westinghouse Specification No. 750S-23-2, Revision 2. In Reference IV.5.2, Page B126, the steam flow rate at 100% power is 3.112 x 10^6 lb./hr. For the Appendix K – 1.67% power uprate, the steam flow rate at 100% power is 3.364 x 10^6 lb./hr (Reference IV.5.3), for an increase of 8.1%. Reference IV.5.4, Table 2-9, contains the steam flow rate for the casualty flow load condition that is described in Reference IV.5.2. For the first ten (10) seconds in Table 2-9, the average steam flow rate for the casualty flow load is 11.93 x 10^6 lb./hr. This flow load is only 3.55 times the normal flow load during the uprate program as compared to the bounding case of four times the normal flow load in Reference IV.5.2 for case 3.

With the equations on Pages A334 through A340 of Reference IV.5.2 and the uprate program flow rate values, the maximum membrane stress in the can deck due to casualty flow load is 10.0 ksi as compared to the 11.0 ksi value in Reference IV.5.2. Similarly, the maximum bending stress in the can deck due to the casualty flow load is 49.2 ksi as compared to 53.7 ksi in Reference IV.5.2. Also, the maximum stress in the horizontal support pipe from the casualty flow load of 11.7 ksi in Reference IV.5.2 bounds the same maximum stress value for the uprate program. The bearing stress of 8.8 ksi in the shroud due to casualty flow load in Reference IV.5.2 also bounds the same stress value for the power uprate.

Therefore, the maximum stress values in Reference IV.5.2 for the separator deck and separators remain bounding for the Appendix K – 1.67% power uprate.

Dryer Deck

Pages A346 through A348 of Reference IV.5.2 detailed the structural evaluation of the dryer deck including the support beam and wall bracket. The same three cases in 5 above were also used for the dryer deck components. The maximum bending stress in the support beam of 49.6 ksi and the maximum stress intensity due to bending in the wall bracket weld of 58.7 ksi in Reference IV.5.2 remain bounding for the Appendix K – 1.67% power uprate program Both values continue to be below the allowable of 64.4 ksi.

The lowest natural frequency continues to be unchanged for the center support beam with a magnitude of 21.8 Hz. This value is above the frequency of 19 – 20 Hz, introduced due to an imbalance of the pump impeller.

Shroud and Shroud Supports

The shroud and shroud supports were originally evaluated on Pages A351 through A370 of Reference IV.5.2. The main loading considered is dead weight while the vessel is rested in the horizontal position. This position is required for shipping and during installation. During normal operation, when the vessel is in the vertical position, most of the weight of the tubes and the water contained therein is supported by the tubesheet. Therefore, the shroud supports experience lower stresses during normal operation than during shipment. Thus, the results in Reference IV.5.2 bound those for the power uprate.

The maximum bending stress in Support Nos. 1, 2, and 3 while the vessel is in a horizontal position are 8.8 ksi, 7.5 ksi and 9.8 ksi, respectively, which are less than the allowable of 29.4 ksi.

Eggcrate Supports

Pages A371 through A372 of Reference IV.5.2 detailed the structural evaluation of the eggcrate supports. The maximum stress condition occurs in eggcrate number 5 and is based on the ΔP across the area of the eggcrate. Since overall flow through the tube bundle does not change

(i.e., steam flow increases but recirculation flow decreases) these values will not change for the power uprate. Thus, the stress results in Reference IV.5.2 remain bounding. The maximum shear stress remains at 0.25 ksi, which is less than the 11.8 ksi allowable value. The maximum radial stress remains at 2.81 ksi, which is less than the 40.0 ksi allowable value.

Tie Rods

Pages A372 through A373 of Reference IV.5.2 detailed the structural evaluation of the tie rods. The stress in the tie rods is based on the deflection in the eggcrate described above. Since these values will not change for the power uprate, the stress results in Reference IV.5.2 also bound those for the power uprate program. The maximum shear stress remains at 0.46 ksi, which is less than the 11.8 ksi allowable value.

Moisture Carryover

Carryover calculation is performed using the "Excel" spreadsheet which utilizes the same methodology as the power uprate evaluations for the Waterford 3 SGs (Reference IV.5.5). Excel provides adequate information for the verifiers to independently verify the equations and results. The steam carryover is a function of water and steam flows through a separator, water level outside the separator, and secondary side pressure. An algorithm of the separator carryover performance is developed using the data from Reference IV.5.6 and the multi-linear regression technique. The algorithm was verified by comparing the input and output data.

Moisture Carryover Predictions

The moisture carryover calculations are performed using the same methodology as Reference IV.5.5. The methodology is based on the empirical correlation developed from data taken from the curves in Reference IV.5.6. Carryover is calculated from the ATHOS3 calculated steam and water flow rates per separator at the Appendix K – power level of 1531 MWt. The bounding water levels of 2 and 16 inches are considered for this evaluation.

The carryover prediction estimates the amount of water contained in the steam flow at the steam generator outlet nozzle and it is defined as:

$\%CO = 100 * M_{CO}/(M_S + M_{CO})$

Where:

M_S = Mass flow rate of steam exiting the steam generator, and
M_{CO} = Mass flow rate of liquid entrapped with steam

The carryover performance of the separators is calculated using the algorithm described in Reference IV.5.6, Section 6.1.1. Table IV-4 (Reference IV.5.6) provides a summary of flow rates at the separator deck and calculated carryover values for the Appendix K –1.67% power uprate conditions. The calculated carryover, with 16-inch uniform water level, is 0.196% and with 2-inch water level it is 0.110%. The maximum calculated MCO value of 0.196% for the 16-inch water level is comparable to the guaranteed maximum steam moisture content of 0.2%.

References (Section IV.5):

IV.5.1 Westinghouse Report No. A-OPPD-9416-1200, Rev. 00, "Evaluation of an ABB/CE "Full Length" Tube Stake for Application in Omaha Public Power District – Fort Calhoun Station Steam Generators", May 1998.

IV.5.2 Westinghouse CENP Report No. CENC-1138, "Analytical Report for Omaha Public Power District Steam Generator", June 1970.

IV.5.3 Westinghouse Calculation Note. CN-SGDA-03-20, "Thermal Hydraulic Analysis of OPPD Fort Calhoun Station Steam Generators for Appendix K - 1.7% Power Uprate Program", March 2003.

IV.5.4 Westinghouse Letter No. LTR-QA-01-013, Rev. 0, Subject: "Transmittal of Containment Licensing Basis LOCA and MSLB Mass and Energy Release Data for Fort Calhoun Station", September 14, 2001.

IV.5.5 Westinghouse Calculation Note. CN-SGDA-03-25, "Thermal Hydraulic Analysis of Waterford-3 Steam Generators at 3716 Mwt Power Uprate Conditions", May 2003.

IV.5.6 Westinghouse (Combustion Engineering) Report, "Steam Separator and Dryer Test Program Summary", February 1984.

IV.5.7 Westinghouse Calculation Note CN-SGDA-03-58, Revision 0, "Vibration and Structural Analysis of OPPD Fort Calhoun Station Steam Generators for Appendix K – 1.7% Power Uprate Program", June 2003.

Table IV-1 Thermal Hydraulic Characteristics of Fort Calhoun Steam Generators

Parameter	Case 1 (Pre-Appendix K Uprate)	Case 2 (Appendix K Uprate)[4]
Power, %	100	101.7
NSSS Power (including RCP heat), MWt	1505.6	1531.1
Tube Plugging, % of Total Tubes	6.5	6.5
Steam Flow Rate per SG, 10^6 lb/hr	3.311	3.364
Steam Pressure, psia	822.2	819.9
Steam Temperature, °F	521.4	521.1
Circulation Ratio	3.87	3.80
Downcomer Fluid Velocity, ft/sec	12.10	12.07
Secondary Side Pressure Drop, psi	37.90	39.10
Secondary Side Liquid Mass, lb	77,638	77,282
Secondary Side Liquid Volume, ft^3	1612.7	1604.6
Secondary Side Vapor Mass, lb	5,486	5,478
Secondary Side Vapor Volume, ft^3	3014.8	3022.9
Total Secondary Fluid Mass, lb	83,124	82,763
Heat Flux at Inlet, BTU/hr-ft^2	108,130	110,016
Average Heat Flux, BTU/hr-ft^2	57,617	58,593
Heat Flux at Outlet, BTU/hr-ft^2	43,216	43,402
Prim. Fluid Flow Rate, 10^6 lb/hr	38.722	38.722
Primary Side Pressure Drop, psi	35.64	35.66
SG Prim. Inlet (T_{HOT}) Temperature, °F	593.8	594.6
SG Prim. Average Temperature, °F	568.4	568.8
SG Prim. Outlet (T_{COLD}) Temperature, °F	543.0	543.0

[4] Note: Westinghouse evaluations for SG performance were performed to a higher power level (i.e., 1.7%, 1526 MWt) versus the requested license application value of 1.67%, 1525 MWt.

Table IV-2 Tube Vibration

Tube Location	Critical Mode No.	Natural Frequency f_n (Hz)	Maximum Stability Ratio
R1C124	29	140.5	0.336
R48C117	32	130.7	0.497
R73C106	28	117.4	0.309
R103C64	28	82.0	0.278

Table IV-3 Tube Peak Displacements

Tube Location	Critical Mode No.	Natural Frequency f_n (Hz)	Mid-span RMS Displacement (mils)	Peak Displacement (mils)
R1C124	29	140.5	0.9	3.6
R48C117	32	130.7	0.7	3.0
R73C106	30	127.8	0.1	0.6
R103C64	28	82.0	0.4	1.4

Table IV-4 Ft. Calhoun Steam Generator Appendix K – 1.67% Power Uprate Separator Loading and Carryover Performance

	Appendix K Power Uprate (1531 MWt)[5]
1. Number of Separators	98
2. Steam Flow Rate at Separator Deck, lb/hr	$3,447.2 \times 10^3$
3. Water Flow Rate at Separator Deck, lb/hr	$10,578.4 \times 10^3$
4. Total Flow Rate at Separator Deck, lb/hr	$14,025.6 \times 10^3$
5. Avg. Steam Flow Rate per Separator, lb/hr	35.17×10^3
6. Max. Steam Flow Rate per Separator, lb/hr	49.97×10^3
7. Min. Steam Flow Rate per Separator, lb/hr	22.29×10^3
8. Avg. Water Flow Rate per Separator, lb/hr	107.94×10^3
9. Max. Water Flow Rate Separator, lb/hr	161.83×10^3
10. Min. Water Flow Rate Separator, lb/hr	68.91×10^3
11. Calculated Carryover Flow Rate, lb/hr (Water Level = 16 Inches)	6.77×10^3
12. Calculated Carryover, % (Water Level = 16 Inches)	0.196
13. 13. Calculated Carryover Flow Rate, lb/hr (Water Level = 2 Inches)	3.78×10^3
14. Calculated Carryover, % (Water Level = 2 Inches)	0.110

[5] Note: Westinghouse evaluations for SG performance were done at a higher power level than requested (i.e. at 1.7% power uprate).

IV.6 Pressurizer

A review of the revised temperature parameters show that any changes in T_{hot} and T_{cold} are very small (0.8°F for T_{hot}), and are bounded by the existing pressurizer stress analysis performed for FCS Unit 1 (WCAP-15889, Rev. 00, Table 8.1-4). The design loading pressure for the stress analysis was 2500 psia. Since there is no increase in operating pressure there is no reduction in margin of stress allowable for the pressurizer components. The changes made to the design transients that affect the pressurizer are insignificant relative to the pressurizer components analysis. For this reason, it is concluded that the revised parameters will not have any impact on the pressurizer stress analysis. It is concluded that the pressurizer components meet the stress analysis requirements for plant operation at the MUR power uprate conditions.

Reference (Section IV.6):

IV.6.1 WCAP-15889, Rev. 00, Table 8.1-4

IV.7 Nuclear Steam Supply System Auxiliary Equipment

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks. An evaluation was performed to determine the potential effect that the revised design conditions will have on the equipment and is discussed in detail in later sections related to safety injection, chemical and volume control.

None of the transients associated with the SITs are impacted by the MUR Uprate Program; therefore, these tanks are not affected by the MUR Uprate Program. Additionally, the MUR Uprate Program has no effect on the pressurizer quench tank or the VCT.

The revised design conditions have been evaluated with respect to the impact on the auxiliary heat exchangers, valves, pumps, and tanks. The results of this review concluded that the auxiliary equipment continues to meet the design pressure and temperature requirements, as well as the fatigue usage factors and allowable limits for which the equipment is designed.

IV.8 Fuel Evaluation

This section summarizes the evaluations performed to determine the effect of MUR uprate program on the nuclear fuel. The nuclear fuel review for the MUR uprate program evaluated the nuclear design, mechanical rod design, and fuel structural integrity.

Reload specific evaluations that confirm loading patterns and associated fuel types utilized in future reload designs will be performed. Reload safety evaluations will be conducted each cycle to ensure that the core design bounds the uprated conditions.

IV.8.1 Nuclear Design

The MUR Power Uprate requires additional energy from the normal 18-month cycle to be able to meet the desired objective of operating capacity factor. The requirement for the additional energy is manifested in increased enrichment or an increased number of feed assemblies. The increased energy needs of the power uprate are similar to those required by an increased capacity factor, which has occurred in stages over the last several cycles from 85% to 90% to

95%. The MUR power uprate represents a smaller increase than either of the 5% capacity factor increases, which have been handled within the normal core design processes.

The Cycle 22 core design (Reference IV.8.1.1) was created to address a 1.5% increase in core average power over the course of the cycle. Increases in the core power level to 1.67% (maximum potential uprate) will not change the acceptability of the peaking factors or downstream analyses, as the combination of core power level plus power measurement uncertainty will continue to be bound by the 102% power used in all the current transient analyses (with the exception of the Main Steam Line Break event, which has a 2% DNBR mixed core penalty that is no longer needed and serves the same purpose as the previous 2% measurement uncertainty allowance).

The Cycle 22 core design proves that the increased energy requirements of the uprated power are easily handled by the OPPD core design process. The core peaking factors are still within previously established limits, and the assembly enrichments, which are currently limited to 4.5 weight percent (w/o) by Reference IV.8.1.2 are not limiting factors. The flexibility in adding additional assemblies to meet increased energy needs will increase the total fuel cost for the station, but will not create a safety or capacity issue. The 44 fresh assembly feeds used in Cycle 22 are within previous reload numbers, so no unexpected issues relating from this feed batch size, such as capacity of the new fuel racks are expected. The information relating to the reload batch size under the uprated conditions have been transferred to the dry cask storage project, to ensure sufficient pool space will exist prior to the implementation of a dry cask storage facility at FCS.

References (Section IV.8.1):

IV.8.1.1 EA-FC-02-030 "Cycle 22 Design Depletions"
IV.8.1.2 FCS Technical Specifications, Section 4.4.2 "Spent Fuel Storage"

IV.8.2 PWR Fuel Design Criteria
The mechanical analyses and evaluations of previous analyses confirm that both the new fuel for cycles 20 and 21 Types 6 and 7 fuel (FTC-6 and FTC-7) continue to meet the approved design criteria for Cycle 21 with the MUR power uprate.

IV.8.3 Fuel Mechanical Evaluation
Mechanical design analyses of the FCS MUR power uprate have been performed using NRC approved mechanical design analysis methodology (References IV.8.3.1 and IV.8.3.2). The analyses address the Framatome-ANP (FANP) PWR generic design criteria (Reference IV.8.3.3).

References (Section IV.8.3):

IV.8.3.1 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.
IV.8.3.2 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.
IV.8.3.3 EMF-92-116(P)(A) Revision 0, Generic Mechanical Design Criteria for PWR Fuel Designs, Siemens Power Corporation, February 1999.

The analyses demonstrate that the mechanical design criteria for the fuel rod and fuel assembly design are satisfied for the MUR power uprate. The evaluation was performed to a peak assembly average exposure of 58,000 MWd/MTU and a peak rod average exposure of 62,000 MWd/MTU when the fuel is operated within the peaking limits given in the TS. 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 IV-5 provides a summary of the reactor information that was used for the mechanical design evaluations and compares that information with the current reactor information.

IV.8.4 Core Thermal-Hydraulic Design

Core T/H analysis and evaluations were performed at a 1.7% uprated core level of 1526 MWt which bounds the proposed 1.67% uprate. The DNBR design limits and safety limits were kept unchanged from the values used in the current design basis analyses. These analyses are reanalyzed each reload. For further details refer to Framatome report EMF-2904 (NP) provided in Attachment 6.

Table IV-5 Comparison of Reactor Operating Conditions

for MUR Power Uprate Mechanical Evaluations

Parameter	Current Value	MUR Power Uprate Value
Core Thermal Power, MWt[6]	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	543
Core Outlet Temperature,°F[7]	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	2.57	2.53
Maximum Rod Peaking Factor, F_R	1.853	1.853
Peak Assembly Burnup, GWd/MTU	58.0	58.0
Peak Rod Burnup, GWd/MTU	62.0	62.0

[6] A higher more bounding core thermal power was used for the fuel design evaluation than that noted in Table 3.

[7] A higher more bounding core outlet temperature was used for the fuel design evaluation than that noted in Table 3.

V. Electrical Equipment Design

The plant presently operates with approximate electrical outputs of 502.2 MWe respectively. The generator rated maximum output is 590.8 MVA. The impact of increasing the output power by 1.67% will result in the generator maximum output increasing to approximately 509.7 MWe. The increased output is accomplished by opening the turbine control valves further, admitting more steam to the turbine.

The increase in steam flow and generator electrical output will result in increased loading of other plant equipment. Components that deliver the electrical output to the grid will be subjected to an increase in current flow. Also, certain generator auxiliary equipment will have increased electrical power requirements as will the motors for certain mechanical equipment necessary to support the increased steam and feedwater flow requirements.

The output of the generator is fed, via isolated phase bus, to the station service transformers. The auxiliary power system consists of the 4160V, 480V, vital 120V and 125VDC systems. Certain power train pumps fed from the auxiliary power system will have increased brake horsepower requirements due to the MUR power uprate. The motors for these pumps have been evaluated and it has been determined that the required motor horsepower will remain below the rated horsepower of the motors, including service factor. Since all of the motor loads are below their horsepower ratings, the current evaluations for the motors, cables, and busses in question will remain bounding following MUR power uprate. Further, there are no significant changes to the loading of the Unit Auxiliary Transformers or House Service Transformers.

The following table summarizes the increased loading of various electrical equipment and identifies whether or not there is sufficient margin to accommodate the 1.67% power increase.

Table V-1 Impact of Power Uprate on Electrical Equipment

Component	100% Power[8]		1.67% Uprate		Design Rating	
Generator (Note 1)	MW	MVA	MW	MVA	MW	MVA
	502.2	590.8	509.7	590.8	NA	590.8
Isophase Bus	590.8 MVA max.		590.8 MVA		620.35 MVA @22,000 V	
Switchyard Breakers	1657 amps		1668 amps		2000 Amps	
Grid Stability	N/A		Unaffected		N/A	
Switchyard Protection	N/A		Unaffected		N/A	
Plant Electrical Distribution	Acceptable		Insignificant		Various	

In each case, the current design of these components and systems continue to bound the 1.67% power uprate conditions. Major electrical components and impacts of the uprate on these components and associated analyses are discussed in further detail below.

[8] Heat balance "benchmark" case lists the generator electric power output at 511 MW, and the heat balance for the uprated output at 519.5 MW. The generator can deliver any amount of electric power within its MVA rating.

V.1 Turbine-Generator

The FCS Turbine is an 1800 rpm, tandem-compound, no reheat with one high pressure (HP) and two double flow low pressure (LP) cylinders. The Turbine was evaluated by the turbine manufacturer for operation at the 1.67% MUR power uprate conditions. The evaluation concluded the following:

- The HP and LP turbines have adequate capacity to pass the increased flow of the MUR power uprate conditions
- The HP and LP turbines and support systems are adequate without modification for operation at the MUR power uprate conditions
- The main turbine mechanical overspeed trip setpoints are satisfactory for the MUR power uprate without a change
- No changes to the EHC pressure control or to the EHC control system are required
- The moisture separator relief valves and piping are adequate since the backpressure in the tailpipes is less than 90 psig.
- The current licensed thermal power turbine missile analysis bounds the MUR power uprate conditions

The FCS Main Generator is rated 590.8 MVA at a 0.85 Power Factor and 45 psig hydrogen pressure. The Generator was evaluated by the generator manufacturer for operation at the MUR power uprate conditions. The evaluation concluded that the Generator will accommodate the MUR power uprate at the same 590.8 MVA rating, 45 psig hydrogen pressure and an approximate power factor of 0.87.

The increase in the electric power output of the main generator to 509.7 MWe does not require an increase in the maximum current flow from the generator through the generator circuit breaker and the main transformers. The MVAR output of the generator can be adjusted, when necessary, so that the total MVA output does not exceed the generator rating of 590.8 MVA when the generator is delivering its maximum power output of 509.7 MWe. The resulting minor reduction in FCS MVAR output to the OPPD 345-kV system under this condition can be easily accommodated, if necessary, by minor adjustment (automatic or manual) in the MVAR output of the other generators in the area. Therefore, the minor reduction in MVAR output capability from FCS while delivering maximum uprated power output is not expected to have any tangible adverse impact on system operation or reliability.

References (Section V.1):

V.1.1 USAR Section 10.2
V.1.2 GE Report "Fort Calhoun Unit # 1, Turbine 170X417 Thermal Power Optimization-Turbine
V.1.3 "Generator Performance Evaluation", April 2002.
V.1.4 S&L Plant Output Distribution System Evaluation, Rev. 0, MUR Power Uprate, Project No. 07751-106

V.2 Main Transformer

The Main Transformer is designed to carry the maximum main generator output and transform the generator output voltage to the transmission system voltage. The main transformer is rated at 648.3 MVA at 65°C and 578.8 MVA at 55°C. The maximum main generator output at 1500 MWt is approximately 502.2 MWe gross generation. At the 1.67% MUR power uprate the generator gross electrical output will be approximately 509.7 MWe. The maximum MVA capability of the main generator remains at 590.8 MVA which is within the rating of the main transformer.

References (Section V.2):

V.2.1 USAR Section 8
V.2.2 Design Basis Document SDBD-EE-201 "AC Distribution"
V.2.3 S&L Plant Output Distribution System Evaluation, Rev. 0, MUR Power Uprate, Project No. 07751-106

V.3 Isolated Phase Bus

The Isolated Phase Bus connects the main generator to the primary windings of the main transformer and the unit auxiliary transformer. The Isolated Phase Bus is rated at 22 kV, 16,280 amperes or 620.35 MVA, with forced cooled temperature rise of 65°C. The maximum main generator output at 1500 MWt is approximately 502.2 MWe gross generation. At the 1.67% MUR power uprate the generator gross electrical output will be approximately 509.7 MWe. At a power factor of .85, this generator gross electrical output would require an isolated phase bus rating of 599.6 MVA which is within the rating of the isolated phase bus. Therefore, the design of the FCS Isolated Phase bus bounds the changes due to the MUR power uprate.

References (Section V.3):

V.3.1 USAR Section 8
V.3.2 Design Basis Document SDBD-EE-201 "AC Distribution"

V.4 4160/480 Volts Distribution System

The 4160/480 volts distribution system is designed to supply electrical power during normal plant operation, including startup and shutdown, and during accident conditions. Following the 1.67% MUR power uprate and during accident conditions the 4160/480 volts distribution system will not experience any additional loads other than or beyond those it has been designed to support. The 4160/480 volts system has been analyzed for the current ESF pump and fan performance characteristics. These characteristics bound the necessary performance requirements for a design basis accident at 102% power. The cables and protective relaying are based on the nominal rating of the motors plus design margins. Therefore, the existing analysis bounds the system's needs during accident conditions following the MUR power uprate.

At the MUR power uprate conditions and during normal plant operation the AC distribution system loads will experience an insignificant increase in power demand since some flows will

increase by approximately 1.67%. The FCS Degraded Voltage Protection Analysis concludes that the 4160/480 distribution system has adequate margin to support the incremental needs during the 1.67% MUR power uprate conditions.

References (Section V.4):

V.4.1 USAR Section 8.3
V.4.2 Design Basis Document SDBD-EE-201 "AC Distribution"
V.4.3 Calculation EA-FC-00-002 "FCS Degraded Voltage Protection Analysis ETS-2.08N-L1, VD1, BL1, MS1"

V.5 Motor Loads and Power Cables

The Auxiliary Power system consists of 4160 V, 480 V, vital 120 V, and 125 VDC systems. Certain power train pumps fed from the Auxiliary Power system will have increased brake horsepower requirements due to the MUR power uprate. The motors for these pumps have been evaluated and it has been determined that the required motor horsepower will remain below the rated horsepower of the motors, including service factor. Since all of the motor loads are below their horsepower ratings, the current evaluations for the motors, cables, and busses in question will remain bounding following MUR power uprate.

References (Section V.5):

V.5.1 EA-FC-90-076
V.5.2 S&L Report Onsite Power System Evaluation, Rev. 0, MUR Power Uprate, Project No. 07751-106

V.6 DC Distribution System

The DC distribution system is designed to supply non-interruptible power during normal, shutdown, accident and post accident conditions to plant inverters, DC control and instrumentation circuits as well as supply the same with non-interruptible power for a minimum of 8 hours upon loss of all ac power. Additionally, it supplies non-interruptible power to non safety related inverters, DC control and instrumentation circuits during startup, shutdown and normal operation.

The 1.67% MUR power uprate in and of itself does not affect the DC system. If modifications to the main feedwater flow instrumentation are required, the instrument AC system load may change which in turn affects the DC system load. The FCS design change process will be followed to ensure the DC system design requirements are met.

References (Section V.6):

V.6.1 USAR Section 8.3
V.6.2 Design Basis Document SDBD-EE-202 "DC Distribution"
V.6.3 Calculation FC-05690 "Battery Load Profile and Capacity Calculation"

V.7 Grid Stability

FCS generator output is fed through a 648 MVA, 22-kV/345-kV main power transformer to a bay in the FCS 345-kV substation (substation 3451) located in FCS switchyard. The substation is directly connected to the 345-kV transmission network via three lines:

- 345-kV line to Omaha via OPPD substation 3459
- 345-kV line to the Lincoln Electric System Wagener Substation via OPPD Substation 3454
- 345-kV line to Sioux City via the Mid American Energy, Inc. Raun Substation

In addition, the 345-kV system is connected to the 161-kV system through two 345-kV/161-kV, 500 MVA autotransformers in the FCS switchyard. The FCS 161-kV substation is connected to the OPPD 161-kV network via three 161-kV transmission lines.

The FCS 345-kV and 161-kV substations are arranged as a breaker and a half scheme and include high speed relaying for line and bus protection. Two independent offsite electric power sources are available for the safety systems. The first is the dedicated offsite 161-kV systems brought in via two 161kV/4.16kV transformers. The second offsite source is brought in from the 345-kV system by opening the motor operated main generator disconnect switch and back feeding the plant through the main power transformer and the unit auxiliary transformers.

A review of the results of the current basis for FCS grid stability performance indicates that the FCS power uprate is not expected to have any adverse impact on the stability of FCS or any of the other neighboring generating units in the network. This conclusion is based on the substantial stability margin that FCS and neighboring generators have as reflected in applicable stability simulation studies. It is also based on the fact that the FCS power uprate reflects an increase of only 1.67% in electric power output which is too small to have a perceptible impact on the system stability characteristics.

Reference (Section V.7):

V.7.1 S&L Report Grid Stability, Rev. 0, MUR Power Uprate, Project No. 07751-106.

V.8 Emergency Diesel Generators

The Emergency Diesel-Generators are designed to furnish reliable ac power for safe plant shutdown and for operation of engineered safeguards, when no power is available from the 345 or 161 kV systems. The capacity of each Emergency Diesel-Generator is adequate to support the operation of required engineered safeguards under the most restrictive design basis accident from initiation through long term post accident cooling.

The electrical loads of each Emergency Diesel Generator consist of:

- Non-load shed continuous loads
- Non-load shed intermittent loads
- Sequential loads

The review of the ESF loads concluded that they have been conservatively determined for the most restrictive design basis accident (LOCA) from 102% power. These loads are not affected by the 1.67% MUR power uprate and no new loads have been identified. Therefore, the existing analyses that document the adequate capacity of the Emergency Diesel-Generators, and fuel oil storage requirements bounds the design basis accident conditions following the MUR power uprate.

References (Section V.8):

V.8.1 USAR Section 8.4
V.8.2 Calculation FC-03382 "Diesel Generator LOCA Loads ETS-2.08N-L1"

V.9 Unit Auxiliary and House Service Transformers

The Unit Auxiliary (UAT) and House Service Transformers (HST) are rated 17.9 MVA with a 65°C rise. Normally, a UAT or HST feeds one bus. During startup, shutdown or when the 161kV transmission system is lost, one transformer can feed two busses; either 1A1 and 1A3 or 1A2 and 1A4. The MUR power uprate will increase the loading of the UATs and/or HSTs by a maximum of 10.8A or 77kVA, if the plant was in one of these conditions. Per the data in calculation EA-FC-00-002, the largest UAT/HST load would be 14.65 MW on UAT T1A-2 or HST T1A-4, if it were feeding busses 1A2 and 1A4. After the MUR power uprate, this maximum load would be increased to 14.73 MW, which is below the rating of either transformer, so it is acceptable.

References (Section V.9):

V.9.1 S&L report Onsite Power System Evaluation, Rev. 0, MUR Power Uprate, Project No. 07751-106.
V.9.2 EA-FC-00-002. "FCS Degraded Voltage Protection Analysis ETS-2.08N-L1, VD1, BL1, MS1"

V.10 Switchgear Busses

The 4160 volt switchgear busses are rated at 2,000 amps. Per the ETAP runs included in calculation Reference V.10.2 the worst switchgear bus current at 100% voltage is 1657A, which can occur when switchgear 1A2 and 1A4 are fed from HST T1A-4. Since the total load increase on each switchgear bus is 5.4A, only 10.8A will be added to bus 1A4, which is less than the current margin of 343A. Thus, the load increase on the switchgear is acceptable based on 25 MW house power load.

Reference (Section V.10)

V.10.1 S&L Report Onsite Power System Evaluation, Rev. 0, MUR Power Uprate, Project Number 07751-106.
V.10.2 EA-FC-00-002. "FCS Degraded Voltage Protection Analysis ETS-2.08N-L1, VD1, BL1, MS1".

V.11 Station Blackout

A Station Blackout (SBO) is defined as the complete loss of alternating current electric power to the essential and non essential switchgear buses. The FCS meets all the SBO Rule requirements and is capable of coping for 4 hours under SBO conditions. The analysis concludes that:

- The coping duration of 4 hours is met
- The diesel generator reliability meets the required guidelines
- Sufficient core coolant inventory is maintained to prevent core uncovery
- There is enough water in the emergency feedwater storage tank to supply the steam generator for the removal of decay heat
- There is sufficient DC battery capacity
- The loss of HVAC will not affect the operability of station blackout equipment
- Containment integrity is ensured by the containment isolation valves

References V.11.1 through V.11.5 were reviewed to determine the impact of the additional 1.67% power produced by the plant under MUR conditions, and the effect on these analyses/calculations. This review has determined that the proposed power uprate will not invalidate any assumptions or alter the conclusions of these analyses/calculations in response to a station blackout event. The coping duration of 4 hours continues to be satisfied, the diesel-generator reliability meets the required guidelines, the DC batteries capacity is sufficient to handle a station blackout event, containment integrity is assured by the containment isolation valves, and the loss of HVAC will not affect the operability of station blackout equipment, additionally;

Reference V.11.6 was examined to determine if the predicted time to core uncovery would be reduced below the 4 hour coping time by the additional 1.67% power produced by the plant under MUR conditions. Examining the core levels at the end of the 4 hour timeframe showed a mixture level in the vessel 6 feet higher than the top of the core and maintaining a near steady state condition. The collapsed liquid level, which is actually cooling the core, remains at near the same level for the preceding hour. The additional decay heat caused by the higher initial power level would not change the acceptability of the core cooling at the 4 hour timeframe. The large uncertainty in the decay heat would also bound the effect of the MUR power uprate power increase of 1.67%
Reference V.11.7 was examined to determine if anything in the analysis for CEDM leakage, which was a critical determinant in the time to core uncovery, would be changed for the MUR power uprate, and would affect the core uncovery timeframe. The testing for the CEDM leakage was not very sensitive to small temperature changes that would be introduced by a 1.67% increase in the core power level. Therefore CEDM leakage would not be expected to change significantly at the higher power or temperature level, and core uncovery would continue to be shown to be beyond the 4 hour coping timeframe.

Reference V.11.8 was reviewed to see if this evaluation, which was performed to determine if the containment pressure and temperature for SBO conditions were more severe than that for MSLB or LOCA. Because the pressure at these SBO conditions was only slightly less than half of the value for the MSLB and LOCA, the increase in the slight increase in the containment temperature and pressure for the MUR would not challenge the conclusion that the MSLB and LOCA events are still more limiting.

Reference V.11.9, which was performed at 100% core power (1500 MWt), was reviewed to determine if the adequacy of the Emergency Feedwater Storage Tank would continue to be met. This calculation's condensate requirements (for 4 hour SBO coping) are bounded by the 8 hour requirements of Reference V.11.10, which was performed at a 102% power (which includes a 2% power uncertainty). This power level bounds the MUR power level with uncertainties.

This review of the FCS SBO coping assessment and supporting calculations and analyses therefore concluded that the 1.67% MUR power uprate will not change the conclusions listed above. NRC issued SER for Station Blackout see Reference V.11.11.

References (Section V.11):

V.11.1 EA-FC-89-054, Rev. 3 "Station Blackout Coping Assessment"
V.11.2 Calculation FC06173 "Diesel Generator Reliability for Station Blackout"
V.11.3 Calculation FC06174 "Required Coping Duration for Station Blackout"
V.11.4 Calculation FC05690 "Battery Load Profile and Capacity Calculation"
V.11.5 Calculation FC06176 "Room Heatup due to Loss of HVAC during Station Blackout"
V.11.6 EA-FC-93-091 "SBO RCS Inventory Analysis"
V.11.7 EA-FC-89-18 "SBO Coping Evaluation and CEDM Leak Testing"
V.11.8 Calculation FC05695 "Verification of SBO Containment Maximum Pressure/Temperature"
V.11.9 Calculation FC06175 "Emergency Feedwater Storage Tank Adequacy for Station Blackout"
V.11.10 Calculation FC06148 "Auxiliary Feedwater Storage Requirements"
V.11.11 NRC Supplemental Safety Evaluation Report, "Station Blackout Rule (10CFR50.63)", TAC No. M68547, April 13, 1992.

VI. System Design

VI.1 Nuclear Steam Supply System Interface Systems

This section discusses the evaluations performed on the NSSS fluid systems using the revised design parameters presented in Table 3, "FCS MUR Uprate - NSSS Design Parameters." For this evaluation, calculations were evaluated to determine whether the NSSS would be impacted by the MUR power uprate.

VI.1.1 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) is designed to perform the following functions:

- Support the SI system by providing a path for hot leg injection in post LOCA long term cooling
- Maintain reactor coolant chemistry and purity
- Maintain reactor coolant volume and provide makeup water to compensate for volume changes resulting from heatup, cooldown and power level changes
- Provide means for adding or removing boron

- Provide a system for mixing and storing of concentrated boric acid
- Provide auxiliary spray to the pressurizer

The CVCS system was designed and constructed in accordance with the following codes and standards:

- ASME Section II, Material Specifications
- ASME Section III, Nuclear Vessel and ASME Nuclear Code Case Interpretations
- ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels
- ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications
- USAS B16.25, B16.11, B31.1 1967, B31.7 Draft 1968

The CVCS system operates under the following modes of operation:

- Normal power
- Startup
- Shutdown
- Emergency
- Post LOCA long term cooling
- Mixing and storing boric acid
- Leak testing
- Resin transfer

From the above modes of operation, the normal power operating mode is affected by the MUR power uprate. In this operating mode the CVCS system is supporting full power operation with the reactor coolant at normal operating pressure and temperatures. At the MUR power uprate conditions the reactor coolant pressure and inlet temperature (T_c) remain unchanged at 2100 psia and 543°F respectively. The hot reactor coolant temperature (T_h), however, changes from 593.3°F to 594.1° F and the average temperature (T_a) changes from 568.2°F to 568.6°F. The increased T_h and T_a temperatures affect the thermal expansion stress and cumulative fatigue usage of the following CVCS components:

- Charging nozzles
- Regenerative heat exchanger
- Letdown heat exchanger
- Letdown piping

The above CVCS components were designed so that the cumulative fatigue usage for all transients over the life of the plant does not exceed 1.0. The review of the cumulative fatigue usage evaluation has concluded that the 0.8°F increase in T_h at the MUR power uprate conditions will increase the cumulative fatigue usage by an insignificant amount and that the cumulative fatigue usage shall not exceed 1.0. The review of the effect of the 0.8°F increase in T_h on the expansion stresses, allowable stresses and loads on supports concluded that:

- Expansion stresses will not increase significantly
- Allowable stresses will not decrease significantly
- Loads on supports will not increase significantly

The CVCS automatically adjusts the volume of water in the reactor coolant system using a signal from the level instrumentation located on the pressurizer. The system adjusts the amount of water that must be transferred between the reactor coolant system and the CVCS during power changes by employing a programmed pressurizer level setpoint which varies with reactor power.

At the MUR power uprate conditions, the reactor coolant system average temperature (T_a) will increase from 568.2°F to 568.6°F. The review of the pressurizer level control program has concluded that the average coolant temperature at the MUR power uprate conditions is within the bounds of the reactor coolant average temperature and therefore, does not affect the pressurizer level control program. It is therefore concluded that, the CVCS will perform the reactor coolant volume control function at the MUR power uprate conditions.

The CVCS design for hot leg nominal temperature is 596.5°F, which bounds the MUR power uprate T_h of 594.1°F. Additionally, the small increase in N-16 activity resulting from the 1.67% uprate will not significantly alter the dose rates at the CVCS charging pump, demineralizer and volume control tank and therefore the existing concrete shielding is adequate.

At the MUR power uprate conditions the boric acid tanks will have sufficient boron to bring the reactor to cold shutdown conditions based on the level and boron concentration requirements that will be established in the core operating limit report. Additionally, at the MUR power uprate conditions, the letdown flow through the system's demineralizers will be bounded by the system's maximum design flow and will have an insignificant impact on demineralizer loadings.

Based on the above, it is concluded that the CVCS will perform its design functions during the MUR power uprate conditions.

References (Section VI.1.1):

VI.1.1.1 USAR Section 9.2 "Chemical and Volume Control System"
VI.1.1.2 Design Basis Document SDBD-CH-108 "Chemical and Volume Control System"
VI.1.1.3 USAR Figures 4.3-10 and 4.3-11 "Pressurizer Level Setpoint"
VI.1.1.4 Evaluation of the impact of the MUR power uprate on CVCS cumulative fatigue usage

VI.1.2 Shutdown Cooling System

The shutdown cooling system is designed to reduce the temperature of the reactor coolant at a controlled rate from 300°F to normal refueling temperature. The system also functions to maintain the proper reactor coolant temperature during refueling and it can be used for reactor coolant purification purposes. While in plant shutdown, the shutdown cooling system provides emergency backup for the spent fuel pool cooling system.

The shutdown cooling system uses portions of other systems i.e., reactor coolant system, and engineered safeguards systems. In the shutdown cooling system, reactor coolant is circulated using the low-pressure injection pumps though the shutdown cooling heat exchangers where the reactor coolant heat is transferred to the component cooling water system.

A review of the SAR and the Design Basis Document for the shutdown cooling system identified only general safety related design performance criteria. These requirements are that the system provide heat removal, coolant circulation and cool down and maintain the reactor coolant system at designated refueling temperature of 130 °F. No specific performance based time criteria were identified. However, a Non-safety related requirement to achieve 130 °F in 27-½ hrs post shutdown is stated (Reference VI.1.2.2).

With this in mind, Several SDC system performance analyses were performed to provide a comparison that relates single and two train performance at core power values for 1500 MWt and 1525 MWt (1.67% uprate).

Table VI-1 shows performance of a two train SDC system cooldown for FCS RCS. Key system functional parameters are that the total flow of the system is 3000 gpm (1500 per heat exchanger); initial RCS temperature is 300 °F and a constant component cooling water temperature of 93 °F is supplied to the shell side of the heat exchanger. The heat exchangers are conservatively assumed as fouled to design limits. In each of the cases, the only parameter that varied is core power effecting decay heat levels. The 1979 ANS decay heat curve is the basis for the decay heat fractions.

Table VI-1 Shutdown Cooling MUR Impact Dual Train Performance

Core Power	Time to RCS Temperature				
	200 °F	180 °F	140 °F	130 °F	125 °F
1500 MWt	4.95 hr	5.3 hr	7.4 hr	13.6 hr	23.0 hr
1525 MWt	4.95 hr	5.3 hr	7.6 hr	14.4 hr	26 hr

Also provided are data that demonstrate single train performance (Table VI-2). In these cases, the same criteria as above were applied, however only 1500 gpm and one SDC heat exchanger are assumed. Decay heat, fouling and component cooling water temperature values are the same as in the two train cases.

Table VI-2 Shutdown Cooling MUR Impact Single Train Performance

Core Power	Time to RCS Temperature		
	200 °F	180 °F	140 °F
1500 MWt	6.95 hr	9.45 hr	67.9 hr
1525 MWt	7.05 hr	9.75 hr	80 hr

From these tables it is apparent that the effect of the decay heat is generally minor for most of the cooldown. However, toward the very end of the cooldown, as RCS temperature approached shell side temperature, the cooldown slows.

Based upon the results of study it is apparent that the SDCS can achieve the design basis criteria of removing core decay heat and achieving refueling temperature. As the core power increases, the cooldown duration increases. This effect is more significant at lower RCS temperature values. It is also important to note that the system has great capacity to cool the RCS to cold shutdown, nominally 200 degrees. Core power increase has much less effect on this level of performance.

References (Section VI.1.2):

VI.1.2.1 USAR Section 9.3
VI.1.2.2 Design Basis Document SDBD-SI-130 "Shutdown Cooling"
VI.1.2.3 Calculation FC05694 "Calculation of Minimum Reactor Coolant Cooldown Time Using Shutdown Cooling System"
VI.1.2.4 Westinghouse Corporation OPPD Fort Calhoun Station MUR Power Uprate Review, Shutdown Cooling System, Engineering Evaluation Shutdown Cooling System MUR Power Uprate Review, April 2003.

VI.1.3 Safety Injection System

The safety injection (SI) system is designed to prevent fuel and cladding damage by supplying adequate core cooling following a loss-of-coolant accident. Additionally, the SI system is designed to provide rapid injection of large quantities of borated water during rapid cooldown of the reactor coolant system caused by a rupture of a main steam line.

The SI system has been designed and constructed to the standards of ASME Section III, Class A and C for heat exchangers, ASME Section III, Class B for vessels and USAS B31.7 and B31.1 for piping. The system consists of passive and active components as follows:

The active components of the SI system consist of the high pressure safety injection (HPSI) pumps, the low pressure safety injection (LPSI) pumps, the safety injection and refueling water tank (SIRWT), the associated piping, valves and instruments. The HPSI and LPSI pumps take suction from the SIRWT and discharge into the reactor coolant system through the four safety injection nozzles.

The passive portion of the SI system consists of four pressurized safety injection tanks. The safety injection tanks are connected to one of the four safety injection nozzles, one in each of the reactor coolant cold legs. The driving head for water injection is provided by nitrogen cover gas at a minimum pressure of 240 psig.
During normal plant operation the SI system is not performing any design function. The SI system, however, has been designed to supply adequate core cooling to prevent fuel and cladding damage following a LOCA from 102% reactor thermal power. The results of the FCS LOCA and MSLB analyses verify that the design HPSI, LPSI pump flows, NPSH, the safety injection tanks borated water design capacity, the piping and instrumentation are adequate to prevent fuel and cladding damage following a LOCA or MSLB from 102% reactor thermal power. Therefore, it is concluded that the SI system will perform its design function at the MUR power uprate.

References (Section VI.1.3):

VI.1.3.1 USAR Section 6.2
VI.1.3.2 Design Basis Document SDBD-SI-HP-132 "High Pressure Safety Injection"
VI.1.3.3 Design Basis Document SDBD-SI-LP-133 "Low Pressure Safety Injection"

VI.1.4 Containment Spray System

The containment spray system is designed to limit the containment pressure rise and reduce the leakage of airborne radioactivity from the containment by providing the means of cooling the containment following a LOCA. This is accomplished by:

- The spraying of cool borated water into the containment atmosphere and by recirculating the cooling water through the shutdown cooling heat exchangers.
- The removal of radioactive particulates that become attached to the water droplets and are carried into the containment sump.

The containment spray system consists of the safety injection and refueling water tank (SIRWT), three pumps, two heat exchangers, 2 sumps, associated piping, valves and instruments. The system was designed and constructed to the standards of ASME Sections III, VIII, XI and USAS B31.7 1968, B31.1 1967, B16.25, and B16.11.

During normal plant operation the containment spray system is not in service. The containment spray system operates upon receipt of the containment spray actuation signal. The system is designed to maintain the peak containment pressure below 60 psig following a LOCA. The LOCA containment response analysis was performed at 102% thermal power and bounds the containment response of a LOCA at the MUR power uprate. Therefore, it is concluded that the containment spray system will perform its design function at the MUR power uprate conditions.

References (Section VI.1.4):

VI.1.4.1 USAR Section 6.3
VI.1.4.2Design Basis Document SDBD-SI-CS-131 "Containment Spray System"

VI.1.5 Regulating Systems

The regulating systems provide the means for monitoring and maintaining control over process variables over the life of the plant and for conditions that can be reasonably anticipated to cause variations in the process variables. The regulating systems instrumentation and control systems include the following:

- Reactor coolant pressure regulating system
- Pressurizer level regulating system
- Feedwater regulating system
- Steam dump and bypass system
- Turbine runback
- Turbine generator control system
- Reactor regulating system

The reactor coolant pressure regulating system maintains pressure within specified limits by the use of pressurizer heaters and spray valves. This system is not affected by the MUR power uprate since the reactor pressure will not change at the uprated conditions.

The pressurizer level regulating system maintains the level by the action of the CVCS system. The level setpoint is a function of the reactor coolant average temperature that will increase

from 568.2°F to 568.6°F following the 1.67% MUR power uprate. The review of the pressurizer level control program has concluded that the average coolant temperature at the MUR power uprate conditions is within the bounds of the reactor coolant average temperature and therefore, does not affect the pressurizer level control program. It is therefore concluded that, the pressurizer level regulating system will perform the reactor coolant volume control function at the MUR power uprate conditions.

The feedwater regulating system maintains steam generator downcomer level within acceptable limits by positioning the feedwater regulating valves. Steam flow, feedwater flow and downcomer level are used in a three-element controller to maintain a preset level at each steam generator during steady state and transient operation. The review of feedwater flow and steam flow instrumentation concluded the following:

- The instrument ranges for feedwater flow is 0 to 400 inches of water and steam flow is 0 to 4 E6 lb/hr
- The expected feedwater and steam flows at the 1.67% MUR power uprate conditions are 380 inches of water and 3.85 E6 lb/hr respectively
- The instrument ranges bound the expected feedwater and steam flows and therefore, this uprate does not affect the feedwater regulating system.

The steam dump and bypass system is designed to establish and maintain hot zero power conditions following a turbine trip or during the unit's start-up. The system senses the average reactor coolant temperature and generates signals that are delivered to the positioners of the dump and bypass valves. At the MUR power uprate conditions the reactor coolant average temperature increases from 568.2°F to 568.6°F and remains within the bounds of the regulating system. Therefore, this uprate does not affect the steam dump and bypass system.

The automatic turbine runback function has been discontinued and manual control is now used to reduce turbine power in the event of a CEA drop. Therefore this regulating system is not affected by the MUR power uprate.

The turbine generator control system is the means by which the turbine generator is made to meet the electrical load demand placed upon it. The turbine first stage pressure, turbine speed and electrical load are used as the control devices. The evaluation of the turbine generator control system by the turbine generator vendor concluded that no changes to the EHC control system are required for operation at the MUR power uprate conditions.
The automatic reactor regulating system function has been discontinued and control rods are manually inserted or withdrawn. Therefore this regulating system is not affected by the MUR power uprate.

The FCS regulating systems controls and instrumentation are not affected by the MUR power uprate.

References (Section VI.1.5):

VI.1.5.1 USAR Section 7.4
VI.1.5.2 Design Basis Document PLDBD-IC-32 "Instrumentation and Control Systems"

VI.1.6 Engineered Safeguards Controls and Instrumentation System

The engineered safeguards controls and instrumentation system was designed to actuate safeguards and essential support systems automatically. The system includes control devices and circuits for automatic initiation, control, supervision and manual test of the engineered safeguards systems. Two independent and redundant initiating systems continuously monitor the status of various systems and initiate protective actions in the event of an accident. The control system is a Class 1 protection system designed to satisfy the criteria of IEEE 279, August 1968.

The engineered safeguards controls and instrumentation system includes the following:

- The safety injection actuation signal
- Autostart of diesel generators
- Sequential starting of engineered safeguards equipment
- Containment spray actuation signal
- Containment isolation actuation signal
- Ventilation isolation actuation signal
- Recirculation actuation signal
- Auxiliary feedwater system controls
- Offsite power low signal (Automatic transfer and load shedding controls)
- Steam generator isolation signal

The FCS engineered safeguards controls and instrumentation system actuates based on a combination of initiating signals each of which is derived from a departure from the normal operating range of one of the critical parameters such as reactor coolant pressure, containment pressure, containment radionuclide content, borated water tank level, etc. The initiating signal setpoints have been established to mitigate the consequences of a design basis accident from 102% power. These initiating signal setpoints bound the MUR thermal uprate conditions and therefore, the engineered safeguards controls and instrumentation system is not affected by the MUR power uprate.

References (Section VI.1.6):

VI.1.15 USAR Section 7.3
VI.1.16 Design Basis Document PLDBD-IC-32 "Instrumentation and Control Systems"

VI.1.7 Instrumentation Systems

The FCS instrumentation systems includes the following:

- Process Instrumentation
- Nuclear Instrumentation
- CEA Position Instrumentation
- Incore Instrumentation

The process instrumentation includes temperature, pressure, level, and flow measurements. At the MUR power uprate conditions some of the process parameters will change. The changes,

however, are small (T_h will increase by 0.8°F while the RCS pressure will remain unchanged) and the existing instrumentation ranges are broad enough to bound the MUR power uprate.

The operating capability of the nuclear and incore instrumentation, per the FCS USAR, is more than adequate to monitor the reactor power from shutdown through startup to 200% power. Therefore, the nuclear and incore instrumentation remains unaffected by the MUR power uprate.

The CEA position indication instrumentation provides a function that is independent of the reactor thermal power; therefore, this instrumentation is not affected by the MUR power uprate.

References (Section VI.1.7):

VI.1.17 USAR Section 7.5
VI.1.18 Design Basis Document PLDBD-IC-32 "Instrumentation and Control Systems"

VI.1.8 Refueling Systems

VI.1.8.1. Spent Fuel Pool

a. Criticality
The design bases for the spent fuel pool storage system is provided in Reference VI.1.19. The spent fuel pool system was designed to store unirradiated fuel assemblies in Region 1 with enrichments that are less than or equal to 4.5 w/o U^{235} (Reference VI.1.19). Spent fuel can be stored in Region 2 if minimum exposure requirements specified in Figure 2-10 of the Technical Specifications are met (graph of enrichment versus exposure up to 4.5 w/o U^{235}). Fuel enrichments for unirradiated fuel are not required that would exceed 4.5 w/o U^{235} for the core designs to support MUR power uprate energy requirements. Thus, the spent fuel storage system is considered acceptable for fuel assemblies that will be used in MUR core designs.

b. Radiation Shielding of Spent Fuel
Adequate shielding for radiation protection of personnel has been provided by handling of irradiated fuel under not less than 10 ft of water. Mechanical stops are provided on the refueling system to maintain the low levels of radiation when handling fuel. The system is also designed such that water cannot drain by gravity out of the storage pool below the top of the stored fuel. The MUR power uprate has no impact on the shielding design features or practices.

c. Spent Fuel Handling Machine
The refueling machine will not be impacted by the MUR power uprate. The fuel assembly design will remain the same in regard to physical handling properties.

VI.1.8.2. Refueling Cavity and Major Handling Equipment

The MUR power uprate will not have an impact on the design basis or system performance of the components that comprise the reactor cavity or handing equipment. The fuel design will remain the same in regards to physical handling properties.

VI.1.8.3. New Fuel Storage

The new fuel storage racks which store unirradiated fuel are designed to store fuel with enrichments up to 5.0 w/o U235 (Reference VI.1.19), and thus are considered acceptable for storage of assemblies designed for use in MUR core designs.

None of the refueling system handling setpoints will be required to be changed as a result of the MUR power uprate.

Reference (Section VI.1.8):

VI.1.19 USAR 9.5

VI.1.9 Containment Systems

No change to the containment structure or containment isolation systems are being made as part of the MUR power uprate. The systems are periodically tested for containment design integrity. There are no changes in the test programs based on a 1.67% power uprate. The containment response for a MSLB and LOCA were performed with 2% uncertainty. Both of these analyses bound operation at the MUR power uprated power of 1525 MWt (See section II.2). Therefore, the 1.67% MUR power uprate does not affect these systems.

VI.2 Power/Steam Systems

As part of the FCS MUR Uprate Program, the following BOP fluid systems were reviewed to assess compliance with the NSSS/BOP interface guidelines:

- Main Steam System
- Condensate and Feedwater System
- Auxiliary Feedwater System
- Steam Generator Blowdown System
- Condensers
- Extraction Steam System
- Shell Side Safety Valves

The review was performed based on the range of NSSS design parameters presented in Table 3, "FCS MUR Uprate - NSSS Design Parameters." The various interface systems were reviewed to provide interface information that could be used in the BOP analyses.

Evaluation of the interface systems, delineated below, indicates that, the design of these systems bounds operation at the uprated core power level, 1525 MWt.

At the MUR power uprate conditions the process temperatures in the BOP piping are expected to increase by approximately 2° F. The expansion stresses resulting from the process temperature increase will have an insignificant impact on the BOP piping and supports. FCS uses spring or rod hangers to support BOP piping. These hangers are designed to support dead loads and accommodate thermal growth of piping thereby minimizing thermal expansion stresses and support loads due to thermal expansion. Therefore, the 2°F increase will have insignificant effect on piping and supports.

VI.2.1 Main Steam System and Steam Dump System

A. MAIN STEAM SAFETY VALVES

The MSSVs must provide overpressure protection for the shell side of the steam generators and the main steam line piping up to the main steam isolation valves. The MSSVs must maintain the SG pressure below the pressure safety limit of 1100 psia (110% of the design pressure).

The steam flow capacity of the valves is as follows:
8 valves (MS-275 through MS-282)
794,062 lbm/hr @ 1035 psig + 3% = 1066 psig

2 valves (MS-291 and MS-292):
126,299 lbm/hr @ 985 psig + 3% = 1015 psig

The total relief capacity with all 10 valves relieving at 1066 psig is = 6,617,605 lbm/hr

The adequacy of the MSSV capacities to provide overpressure protection under MUR conditions is demonstrated by USAR transient analyses.

The MSSVs were evaluated for opening setpoints to ensure that overpressure protection under MUR conditions were demonstrated. The valves were found to have adequate setpoints for overpressure protection. The MSSVs were also evaluated to ensure they would operate under MUR power uprate pressure and temperatures. This evaluation concluded that the MSSVs will continue to operate within design conditions with margin. The MSSVs were determined to provide adequate overpressure protection to the SG at MUR power uprate conditions for AOO events. The MSSV Valves MS-291 and 292 were evaluated for heat removal capability for plant cooldown and RCS temperature control. It was determined that with the MUR power uprate heat load that the MSSVs would be capable of meeting the requirement for heat removal ability.

Following is a listing of the design requirements for the MSSVs which were determined not to be impacted by MUR power uprate:

- Mechanical requirements related to design, construction, materials and testing
- Electrical requirements
- Seismic requirements
- Environmental requirements
- Handling, Storage and shipping requirements
- Testing requirements

Backpressure was evaluated under MUR conditions, and determined that the relief capacity would not be expected to be impacted since the backpressure is a function of set pressure and line resistances which are not changed under MUR power uprate.

Conclusions from MSSV calculations have determined that the valves will not chatter during blowdown, those conclusions are not affected by MUR.

Reference (Section VI.2.1.A):

VI.2.1.1 S&L Report Evaluation 2003-00400, Rev. 0, MUR Power Uprate-Main Steam, Project Number 07751-106

B. ATMOSPHERIC STEAM DUMP VALVE

One ADV is installed in the main steam header ahead of the point where the two main steam lines merged. The function of the ADV is to allow the control of system pressure and temperature by discharging steam to the atmosphere when the main condenser is not available. The valve is manually controlled from the control room.

The valve is capable of a steam flow of 15,000 lbm/hr. The adequacy of the ADV capacity was evaluated and found to still be able to perform its function for MUR power uprate. The ADV was determined to still function adequately under MUR power uprate with pressure and temperatures that are within design conditions with margin. The ADV is non-safety related equipment and it is not credited in any of the Chapter 14 safety analysis. The ADV is credited for plant cooldown to 300°F in natural circulation with two other MSSVs. After MUR power uprate the ADV can still perform this function. Evaluations concluded that the ADV could meet functional and mechanical design requirements.

Reference (Section VI.2.1.B):

VI.2.1.1 S&L Report Evaluation 2003-00400, Rev. 0, MUR Power Uprate-Main Steam, Project Number 07751-106

C. TURBINE STOP, CONTROL AND COMBINED INTERMEDIATE VALVES

The FCS turbine stop, control and combined intermediate valves will be required to pass 1.9% higher flow at the MUR power uprate conditions. These valves were evaluated by the turbine vendor on a total of 17% increase in steam flow from the current design at the VWO point. The evaluation concluded that no changes are required for the stop, control and combined intermediate valves. Therefore, the stop, control and combined intermediate valves are not adversely affected by the 1.67% MUR power uprate and they are expected to adequately perform their design function.

Reference (Section VI.2.1.C):

VI.2.1.2 Turbine – Generator Power Uprate Feasibility Study, General Electric Co., December 11, 2001.

D. STEAM DUMP AND BYPASS VALVES

The steam dump and bypass valves are sized to prevent lifting of the safety valves following a turbine and reactor trip at full load, and for subsequent removal and dissipation of reactor heat. The valves meet the requirements of USAS B31.1. Per SDBD the steam dump and bypass valves must be capable of passing the steam flow equivalent to 40% of full load.

The performance of the steam dump and bypass valves at the 1.67% MUR power uprate conditions has been evaluated. The evaluation concluded that the inlet pressures are higher than the inlet pressure on which the steam dump and bypass valve design flow capacities were based. Therefore, at the MUR power uprate conditions the combined flow capacities of the steam dump and bypass valves meet the functional requirement of passing 40% of full load

steam flow and they will prevent the lifting of the safety valves following a turbine and reactor trip at full load.

References (Section VI.2.1.D):

VI.2.1.3 USAR Section 10
VI.2.1.4 S&L Report Evaluation 2003-0500, Rev. 0
VI.2.1.5 Stone & Webster Evaluation, "Fort Calhoun Station Power Recovery and Power Uprate Evaluation Report" Section 4.3.5, March 31, 1993

E. MAIN STEAM ISOLATION VALVES

The main steam isolation valve (MSIV) safety related function is to prevent an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a MSLB from 102% power. The valves are tested to verify that they can close within 4 seconds upon receipt of an SGIS. The valves have been designed to the requirements of USAS B31.7, draft 68.

The MSIV non-safety related requirement is to be capable in passing the steam flow equivalent to a reactor thermal power of 1500 MWt.

The performance of the MSIVs was evaluated at the MUR power uprate conditions. The analysis concluded that:

- The disc impact energies of the check valve and isolation valve during a pipe rupture and the disc impact energy of the isolation valve during a spurious trip are bounded by the impact energies of the calculation of record.
- The valve actuator will keep the isolation valve at the fully open position during a spurious trip at MUR power uprate full flow conditions since the calculated opening torque for the isolation valve is 32,984 in-lb versus the combined closing torque of 27,400 in-lb.

Therefore, the MSIVs will perform their design function at the MUR power uprate conditions.

References (Section VI.2.1.E):

VI.2.1.6 USAR Section 10
VI.2.1.7 Design Basis Document SDBD-MS-125, Attachments 16 and 12
VI.2.1.8 Nuclear Services Corporation, Report No. SCH-01-02, "Analysis Report-Maximum Disk Impact Energy of Main Steam Check and Isolation Valves at Fort Calhoun Unit 1," dated December 15, 1975.
VI.2.1.9 Nuclear Services Corporation, Report No. SCH-01-03, "Structural Analysis of Main Steam Check and Isolation Valves at Fort Calhoun Unit 1," dated December 15, 1975.
VI.2.1.10Kalsi Engineering, Document No. 2289C, Rev. 0, "Evaluation of MSIV Response to Power Uprate," dated June 13, 2003.

VI.2.2 Condensate and Feedwater Systems

A. Condensate

The Condensate System is designed to deliver condensate from the main condensers through five pairs of low pressure feedwater heaters to the suction of the feedwater pumps. The condensate system includes three 50% capacity condensate pumps. These pumps are each designed to provide a flow of 5600 gpm with a dynamic head of 1160 feet. The condensate system design pressures and temperatures are as follows:

- From the condenser to the condensate pumps 50 psig and 150°F respectively
- From the condensate pumps to condensate header 650 psig and 450°F respectively

The capacity of the condensate system to perform its function at the 1.67% MUR power uprate was reviewed and it was concluded that:

- The design flow and pump motor horsepower of the condensate pumps bounds the MUR power uprate requirements
- The system piping design pressure and temperature bound the MUR power uprate design pressure and temperature
- The condensate system line velocities are lower than the accepted limits

It was concluded that operation at the MUR power uprate condition will not cause additional water hammer loads in the feedwater and condensate systems since they have adequate pressure margin to preclude boiling following a feedwater or condensate pump trip.

Therefore, the condensate system will perform its design function during the MUR power uprate conditions.

The Condensate Storage Tank provides makeup water to the main condenser and receives water dumped from the condenser and is not safety related. The Condensate Storage Tank capacity, its instrumentation and controls and the condensate recirculation, dump and makeup valves are not affected by the MUR power uprate.

References (Section VI.2.2.A):

VI.2.2.1 USAR Section 10.2
VI.2.2.2 FCS Secondary Piping Design Data
VI.2.2.3 System Training Manual Vol. 20 "Feedwater and Condensate System"
VI.2.2.4 Stone & Webster Evaluation, "Fort Calhoun Station Power Recovery and Power Uprate Evaluation Report" dated March 3, 1993
VI.2.2.5 Calculation FC-01427 "Steam Generator Feed System Design Notes"
VI.2.2.6 Condensate Pump Curve
VI.2.2.7 Fort Calhoun Heat Balance for the 1.67% MUR power uprate.

B. Feedwater System

The Feedwater system is designed to perform the following functions:

- Raise feedwater pressure from condensate system pressure to that required to feed the steam generators
- Control the steam generator water level during steady state and transient conditions
- Remove decay heat during shutdown operations if outside power is available
- Provide a means for adding chemicals to the secondary side of the steam generators for corrosion control
- Limit containment pressurization during an MSLB or feedwater line break inside containment

The Feedwater system consists of the following:

- 3 one-half capacity, motor-driven feed pumps (one pump remains in standby) that take suction from the condensate system
- 2 feedwater isolation valves
- 2 feedwater regulating valves
- 2 feedwater regulating bypass valves
- 2 feedwater regulating isolation valves
- 2 feedwater isolation check valves
- 2 high pressure feedwater heaters
- 10 low pressure feedwater heaters
- 2 drain coolers
- Piping, controls and instrumentation

The system has been designed and constructed to the standards of ASME Section III, USAS B31.7, 1968 Draft edition, USAS B 31.1 1967 and USAS B16.5, B16.11, B16.25.

The Feedwater system can be operated in the following modes:

- Normal operation
- Plant startup operation
- Plant shutdown operation
- Abnormal operation
- Emergency operation

During normal plant operation the Feedwater system supplies water flow into the steam generators equal to the steam flow and blowdown flow exiting the steam generators.

At the 1.67% MUR power uprate, the following feedwater system process parameters are affected:

- The feedwater mass flow increases by approximately 1.9%
- The feedwater pump suction pressure is reduced by approximately 6.5 psi based on the increased flow and condensate pump reduced hydraulic head
- Following the uprate, the feedwater pressure will increase slightly from the regulating valve outlet to the steam generators due to increased friction loss with the increased flow.
- Feedwater pressure will decrease from the feedwater pump discharge to the regulating valve due to the lower feedwater pump suction pressure.

- The feedwater pump discharge pressure will decrease by 13.3 psi.

At the 1.67% MUR power uprate conditions the feedwater system components will perform as follows:

- The feedwater pumps will perform their function since the system's pressure and flow are bounded by the feedwater pump's design.
- The feedwater pump's NPSH margin will be 423 feet.
- Additionally, the feedwater pumps will operate at the optimum point of the pump curve.

The full load current of the feed pump motors is currently 5 to 7% higher than the motor name plate full load current. At the MUR power uprate conditions, the feed pump motor brake horsepower will increase by approximately 28 BHP. The motor full load current will increase by approximately 3.4 amperes to 431.4 but remain below the full load current of 469 amperes corresponding to motor's 1.15 service factor.

The feedwater isolation valves maximum flow rate bounds the flow rate at the MUR power uprate conditions.

The increased feedwater flow will not affect the performance of the feedwater regulating valves at the MUR power uprate conditions. The feedwater regulating system maintains steam generator downcomer level within acceptable limits by positioning the feedwater regulating valves. Steam flow, feedwater flow and downcomer level are used in a three-element controller to maintain a preset level at each steam generator during steady state and transient operation. The review of feedwater flow and steam flow instrumentation concluded the following:

- The instrument ranges for feedwater flow is 0 to 400 inches of water in flow element differential pressure corresponding to 0 to 3.5 E6 lb/hr.
- The expected feedwater flows at the 1.67% MUR power uprate conditions are 381 inches of water and 3.41 E6 lb/hr respectively.
- The instrument ranges bound the expected feedwater and steam flows and therefore, this uprate does not affect the feedwater regulating system.

The feedwater regulating bypass valves are not affected by the MUR power uprate since these valves control steam generator water level up to 30% reactor power.

The feedwater regulating isolation valves maximum flow rate bounds the flow rate at the MUR power uprate conditions

The feedwater isolation check valves will perform their function since the system's pressure remains unchanged at the MUR power uprate conditions.

The design of all feedwater heaters and drain coolers bounds the temperature, and pressure at the MUR power uprate conditions. The tube side tube and nozzle velocities are < 10 ft/sec meeting the Heat Exchanger Institute (HEI) guidelines and therefore the feedwater heaters and drain coolers will not be adversely affected by at the MUR power uprate.

The existing piping analysis bounds the operating pressure, temperature conditions at the MUR power uprate.

References (Section VI.2.2.B):

VI.2.2.8 USAR Section 10.2.2
VI.2.2.9 Design Basis Document SDBD-FW-116 "Feedwater"
VI.2.2.10 Calculation FC-01427 "Steam Generator Feed System Design Notes"
VI.2.2.11 System Training Manual Vol. 20 "Feedwater and Condensate System"
VI.2.2.12 Fort Calhoun Heat Balance for the 1.67%MUR power uprate
VI.2.2.13 Calculation FC-05130 "Evaluation of Valve Stroke Time for HCV-1385 and HCV-1386"
VI.2.2.14 Feedwater Pump Curve
VI.2.2.15 Feedwater Heater Specification Sheets
VI.2.2.16 Flow Instrument Data Sheets
VI.2.2.17 Condensate Pump Curve
VI.2.2.18 Feedwater Pump Motor Engineering Data

VI.2.3 Auxiliary Feedwater System

The Auxiliary Feedwater system is designed to supply feedwater to the steam generators during startup, cooldown or emergency conditions whenever the reactor coolant temperature is above 300°F and the main feedwater system is not in operation. The Auxiliary Feedwater system consists of:

- 1 emergency feedwater storage tank
- 1 Steam turbine-driven pump
- 1 Electric motor-driven pump
- 1 non safety related diesel-driven pump and associated diesel fuel oil transfer pump and day tank with a water supply diverse from the emergency feedwater tank
- Remotely operated control valves and interconnecting piping to the main feedwater system and piping to the auxiliary feedwater nozzles in the steam generator

The safety grade portion of the system has been designed and constructed to the standards of ASME Section VIII, and USAS B31.7, 1968 Draft Class II/III. Per the systems design basis a minimum of 55,000 gallons of water is sufficient to remove stored heat above the isothermal condition corresponding to a steam generator pressure of 1056 psia and also remove the maximum decay heat produced during the eight hours after a reactor trip from 102% power. In accordance with the analysis of record the minimum emergency feedwater storage tank inventory is based on a reactor trip from 102% thermal power, which bounds the 1.67% MUR power uprate conditions.

References (Section VI.2.3):

VI.2.3.1 USAR Section 9.4
VI.2.3.2 Design Basis Document SDBD-FW-AFW-117 "Auxiliary Feedwater"
VI.2.3.3 Calculation FC-06148 "Auxiliary Feedwater Storage Requirements"
VI.2.3.4 EA FC 97-12 Rev 0 "Evaluation of Reduced Auxiliary Feedwater Flow"

VI.2.4 Feedwater Heater Drains

The Heater Drain system consists of the heater drain pumps, the heater drain Tank and associated drain control valves and piping. The system is designed to collect condensate drainage from the four moisture separator drain tanks and feedwater heaters, and pump the condensate drainage to the feedwater system. The system has been designed and constructed to the standards of USAS B16.5, B16.11, B16.25, and B31.1.

The design temperatures and pressures of the heater drain system components were compared with the temperatures and pressures from the FCS heat balance at the 1.67% MUR power uprate. The heat balance temperatures and pressures are bounded by the design temperatures and pressures of heater drain components.

The heater drains outlet nozzle calculated velocities are bounded by the Heat Exchanger Institute (HEI) recommendations or in some calculations if shown to be exceeding the HEI recommendations, current operating experience has not identified operational concerns. The fluid velocities in drains piping were evaluated for all heaters at a 5% uprate and found to be adequate.

The drain level control valves were evaluated for the MUR power uprate and concluded that they will perform their design function at the uprated conditions.

The operation of the heater drain pumps at the 1.67% MUR power uprate were evaluated and concluded that the design of the heater drain pumps bounds the MUR power uprate conditions.

References (Section VI.2.4):

VI.2.4.1 USAR Section 10.2.2
VI.2.4.2 Design Basis Document SDBD-FW-116 "Feedwater"
VI.2.4.3 Calculation FC 01427 "Steam Generator Feed System Design Notes"
VI.2.4.4 System Training Manual Vol. 20 "Feedwater and Condensate System"
VI.2.4.5 Fort Calhoun Heat Balance for the 1.67% MUR power uprate
VI.2.4.6 Stone & Webster Report "Extended Power Uprate Final Report, Attachment E, Table 3 " Dated June 7, 2002
VI.2.4.7 Heater Drain Pump Curve

VI.2.5 Condensers

The FCS is designed to operate with two condensers, one for each low pressure turbine. The condensers are a two-pass design with a total of approximately 315,000 gpm of circulating water. Each condenser is divided into two circulating water paths with a common discharge.

At the 1.67% MUR power uprate conditions, the steam flow to each condenser will increase. An evaluation performed at a 1.67% power uprate, which bounds the MUR power uprate conditions, concludes that with an 85°F river water temperature the circulating water system discharge temperature will not exceed 110°F.

The condenser vacuum system is designed to remove air and non-condensable gasses from the condenser shells. The system consists of three electrically driven mechanical vacuum pumps and associated piping. During normal plant operation, two vacuum pumps operate with

the third in standby. Since the primary source of non-condensables is condenser in-leakage, and in-leakage is independent of power level, the MUR power uprate does not affect the condenser vacuum system.

References (Section VI.2.5):

VI.2.5.1 USAR Section 10.2
VI.2.5.2 Design Basis Document SDBD-FW-116 "Feedwater"
VI.2.5.3 System Training Manual Vol. 20 "Feedwater and Condensate System"
VI.2.5.4 Calculation SWEC 03192.00-PH-020 Rev 1 "Maximum Expected Circulating Water Outlet Temperature at 1725M Wt"
VI.2.5.5 Stone & Webster Evaluation, "Fort Calhoun Station Power Recovery and Power Uprate Evaluation Report" dated March 31, 1993

VI.2.6 Extraction Steam

The extraction steam system is designed to transport steam from the high and low pressure turbines to the shell side of the feedwater heaters through the extraction pipes for feedwater heating.

The design pressure and temperature of the high and low pressure turbine extraction steam piping as well as the shell side of the feedwater heaters bound the corresponding pressure and temperature at the 1.67% MUR power uprate. The evaluation of the nozzle velocities on the steam side of the feedwater heaters concluded that the inlet velocities for the 3rd, 4th, 5th and 6th heaters remain within the HEI recommended. The inlet nozzle velocities for the 1st and 2nd heaters exceed the HEI limits; however, they remain below the design velocities of the 1st and 2nd heaters respectively. The increase in steam velocities with the MUR power uprate will increase the potential for impingement plate damage, shell erosion or localized tube vibration. The steam inlet nozzles, impingement plates, and heater shells of the 1st and 2nd point heaters are recommended for inspection in the next outage. The current ongoing inspection process allows for monitoring erosion and tube vibration effects at the MUR power uprate conditions.

The working pressure of the extraction steam line inlet valves has been compared with the pressure at the 1.67% MUR power uprate conditions. It was concluded that the working pressure of the extraction steam line inlet valves bound the pressure at the MUR power uprate conditions.

The feedwater heater shell side overpressure relief capacity for heaters FW-14A, B and FW-15A, B were evaluated and found adequate to support the MUR thermal uprate. The relief flow capacity of heater FW-16A, B was found to be less than required to support the MUR power uprate. Both relief valves associated with feedwater heaters FW-16A, B will be replaced in the next refueling outage.

References (Section VI.2.6):

VI.2.6.1 USAR Section 10
VI.2.6.2 Design Basis Document SDBD-FW-116 "Feedwater"
VI.2.6.3 Fort Calhoun Heat Balance for the 1.67% MUR power uprate
VI.2.6.4 Stone & Webster Report "Extended Power Uprate Final Report, Attachment E, Table 3 " Dated June 7, 2002

VI.2.6.5 LONERGAN Composite Drawing D Series Relief Valve File # 42464
VI.2.6.6 Heat Exchange Institute (HEI) Standards for Closed Feedwater Heaters, 6th edition.
VI.2.6.7 Feedwater Heater Specification Data Sheets

VI.2.7 Steam Generator Blowdown System

The Steam Generator Blowdown System is designed to:

- Maintain steam generator water chemistry within its limits by continuous removal of impurities
- Monitor steam generator effluent for radioactivity to detect potential primary-to-secondary leakage
- Recirculate steam generator water or transfer water from one steam generator to the other
- Drain the steam generator for dry layup.

The steam generator blowdown system consists of the blowdown tank, transfer pump, and respective flow control and isolation valves.

The blowdown flow rates during plant operation can vary depending on feedwater quality. The operation of the steam generator blowdown system at the 1.67% power uprate will not be significantly affected since neither the rate of addition of dissolved solids nor the rate of addition of particulates into the steam generators will be significantly altered.

Reference (Section VI.2.7):

VI.2.7.1 System Training Manual Vol. 25 "Main Steam & Steam Generator System"

VI.3 Cooling and Support Systems

VI.3.1 Component Cooling Water System

The component cooling water (CCW) system is designed to cool components carrying radioactive or potentially radioactive fluids. It also serves as a cooling medium for containment coolers and the control room economizer.

The system is a closed loop consisting of three motor driven circulating water pumps, four heat exchangers, a surge tank, valves, piping, instrumentation and controls. The system has been designed and constructed to the standards of ASME Section III, Class C 1968, ASME Section III, 1968, USAS B31.1 1967 and USAS B31.7, 1968 Class II/III.

The CCW system is designed to support the following operating modes:

- Normal operation
- Shutdown operation
- Emergency Operation following a design basis accident

The minimum hydraulic design requirements for the CCW system have been established based on the credited containment spray system heat removal rate of 280 x 10^6 BTU/hr following the

limiting design basis accident (per References VI.3.1.2 and VI.3.1.3). The heat load associated with this operating mode is the most limiting and bounds the heat loads of the other two operational modes. Since the design basis accident analysis has been performed at 102% power, it bounds the design basis accident analysis results at the MUR power uprate. It is, therefore, concluded that the CCW system will perform its design function at the MUR power uprate.

An assessment was performed to verify that the analyses of record for the peak CCW temperature as documented in Reference VI.3.1.6 would not be impacted by MUR power uprate. The CCW temperature analysis of record was performed at 1560 MWt. Thus, the CCW temperature analysis continues to remain bounding for the MUR power uprate.

References (VI.3.1):

VI.3.1.1 USAR Section 9.7
VI.3.1.2 Technical Specification Section 4.2.3
VI.3.1.3 USAR Section 6.4.1.2
VI.3.1.4 Design Basis Document SDBD-AC-CCW-100 "Component Cooling Water System"
VI.3.1.5 OPPD Calculation FCO5693, Component Cooling Design Loads Rev 1. 8/21/91
VI.3.1.6 Combustion Engineering Document Number 002-ST97-C-028, "Evaluation of Containment Spray Flowrate with Measurement Uncertainty for FC", 9/25/97. Base deck input case 18A Calculation 002-AS95-C-013, Rev. 0, Component Cooling Water and Raw Water Performance Analysis.

VI.3.2 Turbine Plant Cooling Water System

The Turbine Plant Cooling Water System (TPCWS) is designed to provide cooling to the following:

- Turbine lube oil coolers
- Air compressor cylinder jackets, intercoolers and aftercoolers
- Feed pump lube oil coolers and seal water coolers
- Turbine generator alternator exciter air cooler
- Heater drain pump mechanical seal coolers and stuffing box coolers
- Secondary sample coolers
- Condenser evacuation pump seal water coolers
- Condensate pump motor upper bearing oil reservoirs
- Isolated bus duct coolers
- Electrohydraulic control system hydraulic oil coolers

The system is a closed loop consisting of two circulating water pumps, two water coolers, an expansion tank, valves, piping, instrumentation and controls. The TPCWS is cooled by water from the condenser circulating water system. The system has been designed and constructed to the standards of USAS B31.1 1967 and USAS B31.7, 1968 Class II/III.

The following sub-systems of the TPCWS are affected by the 1.67% MUR power uprate:

- Turbine lube oil coolers
- Feed pump lube oil coolers and seal water coolers

- Turbine generator alternator exciter air cooler
- Heater drain pump mechanical seal coolers and stuffing box coolers
- Condensate pump motor upper bearing oil reservoirs
- Isolated bus duct coolers
- Electrohydraulic control system hydraulic oil coolers

Review and comparison of the current operating conditions of the above sub-systems with the expected 1.67% MUR thermal uprate conditions revealed that:

- The current operating temperatures are on the low side of the sub-system's manufacturer recommended operating range.
- The incremental heat load in each of the above sub-systems at the 1.67% MUR power uprate conditions is expected to be small.

Each TPCWS water pump and associated cooler are designed for 100% heat load removal capacity. Therefore, the cooling capability with both cooling loops in operation is more than required to remove all heat loads generated at the MUR power uprate conditions.

It is therefore concluded that the TPCWS will perform its function at the MUR power uprate conditions.

References (Section VI.3.2):

VI.3.2.1 USAR Section 9.9
VI.3.2.2 Calculation FC-03278

VI.3.3 Circulating Water System

The Circulating Water System supplies water from the Missouri River to remove heat from the steam discharged from the low pressure turbines into the condensers and the turbine plant coolers and returns it to the river. Three circulating water pumps normally operate to supply river water. Each pump is designed to provide 120,000 gpm of flow at a total dynamic head of 33 feet.

Following the 1.67% MUR power uprate, the Circulating Water System flow will remain essentially unchanged while the outlet temperature will increase by approximately 0.5°F as a result of the increase in rejected heat. The increase in rejected heat will result in a slight backpressure increase in the condensers. The increase in backpressure remains within acceptable limits. The increase in outlet temperature is bounded by the Circulating Water System design.

References (Section VI.3.3):

VI.3.3.1 USAR Section 10.2
VI.3.3.2 System Training Manual Vol. 7 "Circulating Water System"
VI.3.3.3 Calculation SWEC 03192.00-PH-020 Rev 1 "Maximum Expected Circulating Water Outlet Temperature at 1725 MWt"
VI.3.3.4 Stone & Webster Evaluation, "Fort Calhoun Station Power Recovery and Power Uprate Evaluation Report" dated March 31, 1993

VI.3.4 Raw Water Cooling System

The safety related function of the raw water system is to provide cooling for the component cooling water system. It also provides direct cooling for the shutdown cooling heat exchangers, the high/low pressure safety injection pump bearing oil and seal coolers, the containment spray pump bearing oil and seal coolers, and the control room air conditioners in the event that the component cooling water system is unavailable.

The non-safety functions of the raw water system are to provide a means of transferring water from the emergency feedwater storage tank drainage and overflow, blowdown transfer pump discharge, potable water tank drain and relief valve discharge, bearing water head tank overflow, and the sample system discharge from the steam generator blowdown analyzer to the river. For limited conditions, the raw water system can also provide direct cooling to the containment air coolers in the event that the component cooling water is unavailable.

The raw water system consists of four motor driven pumps, two strainers and motor sets, valves, piping, instrumentation and controls. The system has been designed and constructed to the standards of USAS B31.7, 1968 Class ii/iii and B31.1 1967.

The raw water system can be operated in the following modes:

- Normal operation
- Plant startup operation
- Plant shutdown operation
- Emergency operation
- Abnormal operation (Component cooling water unavailable)

The raw water system has been designed to remove the highest heat loads in the emergency operating mode. Per References VI.3.5.2 and VI.3.5.3the raw water system in conjunction with the component cooling water system are credited for removing 280 x 10^6 BTU/hr from the containment spray system following the limiting design basis accident. Since the design basis accident analysis has been performed at 102% power, it bounds the design basis accident analysis results at the MUR power uprate. It is concluded, therefore, that the raw water system will perform its design function at the MUR power uprate.

References (Section VI.3.4):

VI.3.4.1 USAR Section 9.7
VI.3.4.2 USAR Section 6.4.1.2
VI.3.4.3 Technical Specifications Section 4.2.3
VI.3.4.4 Design Basis Document SDBD-AC-RW-101 "Raw Water"
VI.3.4.5 Calculation FC-04177 "Determination of Minimum Required Post-DBA Raw Water Flowrate"
VI.3.4.6 Calculation FC-05663 "Raw Water Flow-Direct Cooling Mode"
VI.3.4.7 Calculation FC-05693 "Component Cooling Design Heat Loads and Flows"
VI.3.4.8 Calculation FC-06574 "Raw Water System Post-DBA Performance for Normal and LCO System Alignments"
VI.3.4.9 Calculation FC-06697 "Recalculation of River Limits"

VI.3.5 Containment Air Cooling and Filtration

The containment air cooling and filtering system is designed to limit the leakage of airborne radioactivity from the containment during a design basis accident. This is accomplished by:

- The removal of heat released to the containment atmosphere during the Design Basis Accident (DBA) to the extent necessary to maintain the containment structure below design pressure and then reduce the pressure to nearly atmospheric thereby restricting leakage to within design limits.
- The prevention of the accumulation of hydrogen pockets by maintaining continuous filtered flow throughout the containment.

The containment air cooling and filtering system consists of four air handling units, each with its own fan, a common plenum discharge system and instrumentation and controls. The system was designed and constructed to the standards of ASME Sections VIII, IX, XI and USAS B31.7 1968, B31.1 1967, B16.1, B16.5, and B16.11.

The system is designed to mitigate the peak containment pressure for the MSLB accident, which results in the most limiting containment response. This MSLB containment response analysis was performed at 102% power and bounds the containment response of the MSLB at the MUR power uprate conditions. Therefore, it is concluded that the containment air-cooling and filtration system will perform its design function at the MUR power uprate.

Dose Consequences for the accident scenarios evaluated were evaluated for the site boundary and control room in accordance with R. G. 1.183. (Implementation of Alternative Source Terms, Site Boundary and Control Room Dose Analyses for Fort Calhoun Station, January 2001, Stone and Webster). These dose consequence evaluations did not credit containment filtration system for removal of radionuclides. NRC approved the FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221)". Thus, containment filtration systems are not impacted by the MUR power uprate.

References (Section VI.3.5):

VI.3.5.1 USAR Section 6.4
VI.3.5.2 Design Basis Document SDBD-VA-CON-139 "Containment HVAC"

VI.4 Auxiliary Building Heating, Ventilating and Air-Conditioning Systems

The auxiliary building HVAC system is designed to maintain a suitable environment for the equipment and personnel and to protect personnel and the public from airborne radioactivity. The auxiliary building HVAC system ventilates the following controlled and uncontrolled areas:

Controlled Areas

- Safety injection pump rooms
- Fuel storage area
- Waste holdup and disposal area
- RCA exit area

- Steam generator blowdown equipment area
- Shutdown and letdown cooling heat exchanger room
- Mechanical penetration areas
- Room 69
- Interconnecting hallways and open spaces in the controlled areas

Uncontrolled Areas
- Switchgear and cable spreading rooms
- Battery rooms
- Electrical penetration areas
- QA vault
- Room 81
- Room 19
- Emergency diesel generator rooms
- Elevator machinery room
- Interconnecting hallways and open spaces in the uncontrolled areas

The review of the systems in the controlled areas concluded that there would not be any change to the cooling load requirements as a result of the MUR power uprate conditions including the safety injection pumps that are designed to mitigate accident conditions from 102% power.

The review of the systems in the uncontrolled areas concluded that at the 1.67% MUR power uprate, the change in the steam flow from the generators to the steam turbine only affects Room 81. This area is not identified as a critical area in the calculation of record. The cooling load is based on temperature differences and the convection coefficients. The temperature difference in the main steam piping located in Room 81 during the MUR power uprate conditions will not change and the convection coefficients were based on conservative values. Since the steam flow increase is small, the convection coefficients are not required to change and the cooling load analysis of record remains valid for the MUR power uprate conditions.

References (Section VI.4):
VI.4.1 USAR Section 9.10
VI.4.2 Design Basis Document SDBD-VA-AUX-138 "Auxiliary Building HVAC"
VI.4.3 Calculation FC05905 "Auxiliary Building HVAC Cooling Load"
VI.4.4 Calculation FC05904 "Auxiliary Building HVAC Heat Load"

VI.5 Nuclear Steam Supply Control Systems

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.

Detailed information regarding the statistical setpoint evaluations are provided in Attachment 6, Section 5.0.

LTOP
Evaluation of the MUR Uprate on the LTOP analysis

The MUR Uprate program doesn't change any plant conditions that would impact the LTOP analysis or system. The LTOP system was recently reanalyzed for protection capabilities based on the RELAP5/Mod 3.2 computer code utilizing a power level equal to 102%. This power level is translated into the LTOP analysis via the amount of decay heat that is assumed to exist in the RCS at the time when an LTOP event occurs. Thus, for LTOP events, the MUR uprating does not impact the plant response for these events. The request to implement a Pressure-Temperature Limits Report (letter LIC-02-0109 from OPPD to NRC, dated October 8, 2002) contains the LTOP methodology that is currently under review by the NRC. This review and approval is expected to be completed prior to the commencement of the power uprate at FCS.

Conclusions for the impact of the MUR power uprate on the setpoint analysis continues to be bounded by the analysis of record. There are no proposed changes to setpoint methodology as a result of MUR power uprate.

References (Section VI.5):

VI.5.1 EMF-1961 (P)(A) Revision 0, Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors, Siemens Power Corporation, July 2000.

VI.6 Instrument Air System

The instrument air system is designed to provide oil-free, filtered, and dried air for pneumatic controls, instrumentation, and the actuation of valves, dampers and similar devices.

The basic function of the instrument air system is to provide sufficient stored instrument grade air in local accumulator tanks to permit the operation of safety related pneumatic devices during and following an accident and to provide sufficient instrument grade air to meet the demand of all pneumatic instruments, controls, valves, and dampers during normal plant operation.

At the 1.67% MUR power uprate conditions the demand on the instrument air system will not be affected since the operation of pneumatic devices, controls, valves, instruments and dumpers is

not affected by the uprate. Additionally, the instrument air system will not be affected as a result of pipe whip, jet impingement or environmental conditions following a high energy line break at the MUR power uprate conditions.

References (VI.6):

VI.6.1 Design Basis Document SDBD-CA-IA-105 "Instrument Air"

VII. Other

VII.1 Control Room, Control Room Habitability and Simulator

There are no control panel annunciators being added to the control room as a result of the MUR Uprate Program. Notification of the operators of the CROSSFLOW system condition will be through computer alarms and annotation of the computer display. Response to this computer alarm will be proceduralized. This will be finalized in the design change to implement the MUR Uprate Program in coordination with Operations to ensure that implementation meets operations and design requirements. Control room instrumentation and displays will be re-scaled as a result of implementation of the 1.67% power uprate. This will be addressed in the design change package that implements the installation of the CROSSFLOW system. Modifications associated with the MUR power uprate will be completed prior to implementation.

The control room habitability system is designed to perform the following functions:

- Provides cooling, heating and ventilation for the control room under normal and accident conditions
- Protects the control room operators from airborne radioactivity and direct radiation in the event of a design basis accident
- Supplies outside air during normal operating conditions and maintains a slight positive pressure within the control room
- Protects the control room operators from toxic gasses in the event of a chemical accident

The system consists of two 100% capacity air conditioning units, two 100% capacity outside air filter units each with its own fan, an outside air intake plenum and distribution ductwork. Each of the two outside air filter units consists of an electric air heater, a prefilter bank, an upstream HEPA filter bank, two banks of charcoal filters and a downstream HEPA filter bank.

The control room habitability system has been designed to maintain a habitable atmosphere in the event of a toxic chemical accident or a design basis accident. The analysis of record shows that the control room habitability system limits the control room personnel radiation exposure during the design basis accident LOCA from 102% power to the levels specified by GDC 19. This analysis of record previously discussed in section II.2.5 bounds the consequences of the 1.67% MUR power uprate. Additionally, the analysis of record of the toxic chemical accident is not affected by the MUR power uprate.

The FCS simulator, which reflects the design of the control room, will be modified in the future.

References (Section VII.1):

VII.1.1 USAR Section 9.10
VII.1.2 Design Basis Document SDBD-VA-CR-140 "Control Room Habitability"
VII.1.3 Calculation FC06813 "Site Boundary and Control Room Doses Following a LOCA Accident Using Alternate Source Terms"
VII.1.4 EA-94-012 "Transportation And Nearby Facility Accident IPEEE Report For The Fort Calhoun Station"

VII.2 Operator Actions

No changes are required to the FCS EOP program as a result of the MUR Uprate Program. Specific procedures within the EOP program may require review and revision based upon the MUR Uprate Program plant parameters for thermal power, temperature, and pressure values. These changes will be identified and implemented under the design change process to implement the MUR Uprate Program. Specifically, values in the EOPs, and the AOPs may need to be revised based upon the 1.67% power uprate levels. Any changes to values that are referenced in the EOPs or AOPs will be revised by the EOP/AOP control program, to fully implement the MUR Uprate Program.

The MUR Uprate Program will have no impact on the time available for operator actions as assumed in the accident analysis. Specific impacts on operating procedures are further discussed in section VII.4 of this license amendment request.

VII.3 Power Uprate Modifications

As demonstrated in Sections II through VI, the current plant analyses, design, and operation ensure that the applicable acceptance criteria are met for the MUR Uprate Program. No changes to the RCS or NSSS systems are required to support the MUR Uprate Program.

The changes in flowrates, pressures, and other operating parameters can be accommodated by all existing equipment in the condensate or feedwater systems. Therefore, no plant changes/modifications are required to the condensate or feedwater systems to implement the MUR Uprate Program other than the installation of the CROSSFLOW flow instrumentation itself and change out of FW-16 relief valves (scheduled for 2003 RFO).

As the impacts of the MUR Uprate Program are bounded by the current design and operation of the AFW system, no modifications are required to this system for implementation of the MUR Uprate Program.

Feedwater Heater FW15-A/B will be replaced in 2003 due to component reliability requirements. No plant changes/modifications are required to the feedwater heaters and drains for implementation of the MUR Uprate Program.

The 1.67% power uprate will result in minimal changes in the CCW system flow requirements. These changes are bounded by current system design; therefore, no plant changes/modifications are required to the CCW system to implement the MUR Uprate Program.

The review of electrical systems in support of the proposed uprate indicates that no changes are required to support the MUR Uprate Program.

VII.4 Plant Operating Procedure Changes

Procedural impacts for the RCS and NSSS systems will be identified in the process for the implementation of the design change package that installs the CROSSFLOW system. Impacts are anticipated to normal operating, alarm response, AOP and EOP procedures. In particular surveillance procedures for reactor thermal power will be affected, as well as operator responses to an out-of-service condition on the CROSSFLOW system, as described in Section I. These changes will be implemented prior to raising plant core power above 1500 MWt.

No other procedural impacts were identified in the review of NSSS, BOP, and support systems and their associated analyses.

With respect to temporary operation above "full steady-state licensed power levels," OPPD will continue to follow the guidance in NRC Inspection Procedure 61706 (Reference VII.4.1) through appropriate changes to existing plant procedures during the implementation of this amendment when approved. OPPD does not commit to the guidance in Section VII.4 of RIS 2002-03. This is consistent with the direction provided in Reference VII.4.2.

References (Section VII.4):

VII.4.1 NRC Inspection Procedure 61706 dated July 14, 1986 (Inspection and Enforcement Manual, item d) Core Thermal Power Evaluation.

VII.4.2 NRC Memorandum, Donna Skay-Senior Project Manager Section 1 to Ledyard B. Marsh-Deputy Director, "Summary Meeting Held on October 23, 2002 Between the U. S. Nuclear Regulatory Commission (NRC) Staff and Industry Licensing Action Task Force" January 7, 2003.

VII.5 Environmental Review

OPPD has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. OPPD has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment 1 of LAR, Section 5.1, No Significant Hazards Consideration, this proposed amendment does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The MUR Uprate Program thermal power increase will not alter or increase the inventory of radionuclides in the RCS above the current analysis of record. This change will not alter the fuel cladding in a way that affects its mechanical and structural integrity or affects its leakage characteristics. This power uprate will not alter or increase the primary pressure, so there is no additional challenge to the RCS or other fission product barriers. Additionally, increasing core thermal power by 1.67% will not affect or increase water production or inventory use in any way that will affect effluent volume or production. Finally, the 1.67% uprated plant heat discharge will remain below the site FCS limit. The 1.67% power uprate is bounded by the previously evaluated thermal effluent limits. Therefore, this change will not result in a significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The MUR Uprate Program thermal power increase will not alter or increase the analysis of record inventory of radionuclides in the RCS. The radionuclide source terms applicable to personnel dose determination were calculated assuming a core thermal power of 1530 MWt, which bounds the uprated core power of 1525 MWt. This change will not alter the fuel cladding in a way that affects its mechanical and structural integrity or affects its leakage characteristics; therefore, there is no additional challenge to the RCS or other fission product barriers. Finally, no new effluents or effluent release paths are created by the MUR Uprate Program. Therefore, this change will not result in an increase in individual or cumulative occupational radiation exposures.

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

VII.6 Programs

VII.6.1 Environmental Qualification Program

The purpose of the Electrical Equipment Qualification (EEQ) Program is to demonstrate and maintain qualification of Class 1E electrical equipment at the FCS. The EEQ program ensures the Class 1E electrical equipment is qualified and capable to perform its design function under the most adverse environmental conditions expected during their operating life and any design basis accident.

The electrical equipment qualification criteria for pressure, temperature, humidity/spray and radiation dose have been based on the results of the following analyses:

- Loss of Coolant Accident
- Main Steam Line Break Accident and
- High Energy Line Break Accident

The above analyses were performed using a reactor thermal power greater than the thermal power of the 1.67% MUR power uprate. The current pressure, temperature, humidity/spray and radiation dose profiles bound the environmental conditions expected at the 1.67% MUR power uprate conditions.

Reference VII.6.1.5 documents the mass and energy releases for subcompartment analysis. This calculation was reviewed and it was verified that the Mass and Energy analysis was performed at 1560 MWt. The subcompartment mass and energy releases remain bounding for the 1.67% MUR power uprate.

The electrical component aging evaluations are based on the ambient temperatures and dose rates at the current operating conditions. The effect of the 1.67% MUR power uprate has a negligible impact on ambient temperatures and dose rates. Additionally, the aging analysis of record has approximately 10°F conservative margin in ambient temperature. Therefore, the MUR power uprate does not affect the EEQ program. NRC issued SER for Environmental Qualification as Reference VII.6.1.6.

References (Section VII.6.1):

VII.6.1.1 Fort Calhoun Electrical Equipment Qualification Manual
VII.6.1.2 USAR Appendix M "Postulated High Energy Line Rupture Outside Containment"
VII.6.1.3 Westinghouse Report "Review of Containment Subcompartment Mass and Energy Releases for Power Uprate"
VII.6.1.4 Westinghouse Report "Review of HELB in Aux. Building Rooms 63, 64, 65 for Power Uprate"
VII.6.1.5 Calculation O-PEC-140 Rev. 0, "Fort Calhoun Mass/Energy Release Data for Steam Generator and Reactor Vessel Inlet, Outlet Nozzles Breaks, 1/22/80.
VII.6.1.6 NRC SER Environmental Qualification of Safety Related Electrical Equipment, May 21, 1981.

VII.6.2 Motor-Operated Valve Program

The FCS Motor Operated Valve (MOV) program includes 27 safety related and 2 non-safety related MOVs. The mission of this program is to assure that the installed MOVs will function properly upon demand. The following valves are included in this program:

- HPSI Header Isolation HCV-311, 312, 314, 415, 317, 318, 320, 321
- LPSI Header Isolation HCV-327, 329, 331, 333
- Containment Sump Isolation HCV-383-3, 383-4
- SIRWT to Charging Pump Isolation LCV-218-3
- Charging Pump Discharge to HPSI Isolation HCV-308
- Boric Acid Pump to Charging Suction Isolation HCV-268
- Volume Control Tank Outlet Isolation LCV-218-2
- Concentrated Boric Acid Tank Gravity Feed Isolation HCV-258, 265
- Shutdown Cooling Isolation HCV-347, 348
- PORV Isolation HCV-150, 151
- Main Feedwater Isolation HCV-1385, 1386
- Auxiliary FW to Main FW Header Cross-Connect HCV-1384
- Feedwater Regulating Valves HCV-1103, 1104

The review of the MOV program valves concluded that the maximum opening or closing conditions that could prevail for each valve at the current 100% power depend on the following:

- HPSI pump head
- SIRWT static head
- LPSI pump head
- Shutdown cooling pressure limit
- Containment pressure following a large break LOCA
- Boric Acid Makeup Pump head
- Boric Acid Storage Tank head
- Relief valve setpoint
- Safety valve setpoint
- Feedwater pump head with minimum recirculation

The above parameters are either not affected by the MUR power uprate or bound the MUR power uprate conditions. Therefore, the 1.67% power uprate does not challenge the capability of valves in the MOV Program to satisfy their design functions. No changes are required to the MOV Program as a result of the MUR Uprate Program.

At the MUR power uprate conditions process temperatures may increase by approximately 2° F. The resulting stresses in isolated piping at the MUR conditions will be slightly higher than at the current operating conditions, however; these piping stresses will be much lower than the faulted allowable stresses defined in PED-MEI-5.

References (Section VII.6.2):

VII.6.2.1 PBD-8 "Program Basis Document Motor Operated Valve"
VII.6.2.2 Calculation FC05880 "An Evaluation of Operating Parameters Acting on Motor Operated Valves for the Determination of Maximum Operating Conditions"

VII.6.3 Air Operated Valve Program

The FCS Air Operated Valve (AOV) program is still under development. The program classifies the AOVs in three categories based on the following criteria:

- Category 1 Includes AOVs that perform an active function of high safety significance, and are safety related
- Category 2 Includes AOVs that perform an active function that does not have high safety significance, and are safety related. It also includes non-safety related AOVs that are classified as high risk.
- Category 3 Includes AOVs that are not safety related, do not support safety related systems and are not classified as high risk.

The review of the Category 1 AOV program calculations concluded that the maximum expected differential pressures used as input for the AOV component design basis analyses at the current 100% power depend on the following:

- HPSI pump head
- SIRWT static head

- RAW pump head
- River level elevation head
- Containment pressure following an accident
- Containment sump pump head
- Relief valve setpoint
- Auxiliary feedwater pump head

The above parameters are either not affected by the 1.67% MUR power uprate or bounded by the MUR power uprate conditions. The AOVs of the feedwater, condensate and heater drain systems will process approximately 1.67% higher flows at the MUR power uprate conditions. The pressure drops across the AOVs associated with these systems at the MUR power uprate conditions will be reduced (inlet pressure reduced, outlet pressure slightly increased) as a result of the higher flows. The pressure drops across the AOVs are bounded by the respective AOV's shut off differential pressure (DP) and therefore, it is concluded that AOVs associated with the feedwater, condensate and heater drain systems will not be affected by the MUR power uprate.

References (Section VII.6.3):

VII.6.3.1 Calculation No. 00234-C-001 Rev. 0 "RAW Water System AOV Functional and MEDP Calculation"
VII.6.3.2 Calculation No. 00234-C-002 Rev. 0 "Chemical and Volume Control System AOV Functional and MEDP Calculation"
VII.6.3.3 Calculation No. 00234-C-003 Rev. 0 "Containment Isolation System AOV Functional and MEDP Calculation"
VII.6.3.4 Calculation No. 00234-C-004 Rev. 0 "Safety Injection AOV Functional and MEDP Calculation"
VII.6.3.5 Calculation No. 00234-C-005 Rev. 0 "AFW and MS-AFW System AOV Functional and MEDP Calculation"

VII.6.4 Flow-Accelerated Corrosion (FAC) Program

The purpose of the FAC program is to predict, detect, monitor, and mitigate FAC in plant systems. The scope of the program includes all piping and components that cannot be demonstrated to be non-susceptible to FAC as documented in the current FAC Program System Susceptibility Evaluation. The program conducts ultrasonic pipe wall thickness measurements, predicts corrosion wear rate, establishes pipe section replacement criteria and initiates corrective actions to ensure that all applicable piping systems are adequate to continue performing their design function.

The 1.67% MUR power uprate changes the operating pressure, temperature, quality and velocity in several of the BOP systems. Review of the FAC program and analyses performed by the FAC program erosion prediction model, using MUR power uprate conditions, concluded that:

- The MUR power uprate conditions affect the FAC wear rates in several BOP piping systems

- No additional piping systems should be added to the FAC program

- Changes to piping wear rates at the MUR power uprate conditions have been identified. Monitoring, and mitigating actions are being pro-actively planned in accordance with the FAC program requirements

- The FCS FAC program is adequate to support the MUR power uprate, and will include continued monitoring

The FAC program is not affected by the 1.67% MUR power uprate.

References (Section VII.6.4):

VII.6.4.1 PED-3 "Flow Accelerated Corrosion"
VII.6.4.2 CHECWORKS Predicted Wear Rates at the MUR Uprate Conditions

VII.6.5 High-Energy Line Break

The FCS HELB Program was reviewed in support of the MUR Uprate Program. This review determined that no HELB program changes are required to be implemented as a result of the MUR Uprate Program. The activities, elements, and philosophy that currently constitute the HELB Program are not affected by the MUR Uprate Program. In accordance with FCS's design change process, the design change package for installing the CROSSFLOW system will be evaluated against the HELB Program requirements as required in the FCS plant modification process. No new piping is added, no postulated break locations changed, and no changes are made to the assumed blowdown from any currently-postulated breaks; therefore, there is no impact on the current FCS HELB analysis.

Reference VII.6.5.1 documents the results of a High Energy Line Break in the Auxiliary Building diesel rooms. This calculation assumed a constant mass release of 1000 lbm/hr at an enthalpy of 1197.8 btu/lbm. Steam generator pressure is expected to decrease approximately 2 psi due to the uprate, ensuring that the uprated condition is bounded by the current analysis. The increased steam flow will increase the pressure losses down stream in the steam lines. The steam line pressure in the diesel room is controlled by a regulator and not affected by power changes. The Reference VII.6.5.1 document remains applicable for the OPPD Appendix K uprate.

The review of the high energy line break analysis of record for Auxiliary Building Rooms 63, 64, 65 and 81 concluded that the analysis of record bounds the HELB conditions at the 1.67% MUR power uprate.

Reference (Section VII.6.5):

VII.6.5.1 Calculation 002-AS92-C-003, Rev. 0, "High Energy Line Break Analysis for Fort Calhoun Station Aux. Building Rooms 63-65" dated 10/27/92.

VII.6.6 Inservice Inspection Program

The In-service Inspection (ISI) program defines the scope and method of examination of Class 1, 2 and 3 components and supports as well as the procedures and examination schedule of these components and support at the FCS.

The MUR power uprate does not impact the scope, method of examination, schedule and requirements or criteria of the ISI program. Additionally, the operating condition changes associated with the MUR power uprate are bounded by the design of the ISI components and supports and do not affect the program scope, selection criteria, or acceptance standards. Therefore, the ISI program is not affected by the MUR power uprate.

References (Section VII.6.6):

VII.6.6.1 PBD - 2 "Inservice Inspection Performance Instruction"

VII.6.7 Inservice Testing Program

The In-service Testing (IST) program defines the scope of Class 1, 2 and 3 pumps and valves to be tested, the test method and test schedule at the FCS.

The MUR power uprate does not impact the scope, test methods, schedule and requirements or criteria of the IST program. Additionally, the operating condition changes associated with the MUR power uprate are bounded by the design of the IST pumps and valves and do not affect the valve scope, selection criteria, or acceptance standards. Therefore, the IST program is not affected by the MUR power uprate.

References (Section VII.6.7):

VII.6.7.1 PBD - 2 "Inservice Inspection Performance Instruction"

VII.6.8 Radiological Environmental Monitoring Program (REMP)

No changes will be required to the REMP for monitoring the types or amounts of any effluents that may be released offsite. The current USAR Chapter 14 radiological accident analysis fully bounds the MUR Uprate Program. Also, the power uprate will not increase the inventory of radionuclides in the RCS above analyzed limits, nor will increasing power affect the fuel cladding in a way that alters its structural integrity. The radionuclide activity core inventory used in the radiological consequences analyses was calculated at a core thermal power of 1530 MWt. Therefore, no changes are required to FCS's REMP as a result of the MUR Uprate Program.

<u>Radioactive Waste Disposal System</u>
The radioactive waste disposal system (RWDS) is designed to protect plant personnel and the public from exposure to radioactive wastes in accordance with 10 CFR Part 20; 10 CFR 50, Appendix I; 40 CFR Part 190; 10 CFR 50 Appendix A General Design Criteria 60, 63, and 64; 10 CFR 50 Appendix B for reviews and audits; and the intent of NUREG-0472, Draft Revision 3 (per Reference VII.6.8.1).

The RWDS includes equipment to collect, store, process, and treat as required, monitor, and dispose of liquid, solid, and gaseous radioactive wastes. The RWDS is designed to process and remove radioactive wastes from the plant adequately and safely when 1% of the core fuel elements have failed and corrosion and fission product concentrations in the reactor coolant are at design values. The design of the RWDS is based on the plant operating cycle shown in Reference VII.6.8.1.

The accumulated radioactive waste inventory has been calculated assuming operation with 1% failed fuel in the core. The Alternate Source Term per R. G. 1.183 was implemented for FCS in 2001, Reference VII.6.8.1, and was used to calculate the time dependent fission activity levels of individual nuclides in the fuel rods and coolant. The radioactive waste inventory was computed based on a power level of 1530 MWt. This inventory was accepted by the USNRC SER of FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221). Section 2.2.1 of the SER addresses that FCS consistently used the 1.02% core power per R. G. 1.59, and as such approved of the application.

The parameters used in the radioactive waste system calculation are summarized in Reference VII.6.8.1.
The liquid waste collection and storage system is divided into three sections; hydrogen bearing reactor coolant liquids, auxiliary systems process wastes, and hotel wastes. Anticipated annual quantities of liquid waste releases and the corresponding annual average concentrations in the discharge tunnel are given in Reference VII.6.8.1 for those nuclides expected to have annual average concentrations greater than $1x10^{-20}$ μCi/cc. As illustrated by the table, it is expected that no single nuclide will exceed 1% of 10 CFR Part 20 limits on an annual average basis (based on a 1530 MWt core inventory). Cumulative dose contributions from radioactive materials in liquid effluents released to unrestricted areas shall be determined on a quarterly basis in accordance with the ODCM.

Radioactive waste gases are collected, compressed, stored, analyzed, and monitored in the radioactive waste disposal system. Waste gas found to be suitable for discharge in accordance with the requirements set forth in 10 CFR Part 20 are released under controlled conditions to the auxiliary building ventilation system for dilution prior to discharge at the plant stack. The total expected annual activity release to the atmosphere from the (1) waste gas system, (2) containment purges, (3) auxiliary building ventilation and (4) primary-to-secondary leakage and (5) Radioactive Waste Processing and CARP buildings are (accounted for in the auxiliary building ventilation release) in Reference VII.6.8.1. The waste gas inventory calculated for Reference VII.6.8.1 was based on a 1530 MWt core inventory.

All components of the system were evaluated with respect to any slight increase in temperature or pressure. The impact of the MUR power uprate is such that the RCS temperature may be approximately 0.8°F higher. There is no pressure change anticipated. As such, the liquid process waste stream may be 0.8°F higher than previously processed. The impact of such is negligible. As shown in Reference VII.6.8.1 the normal operating temperature range has sufficient margin compared to the design temperature range. As such, the MUR power uprate will not impact the performance of the Radioactive Waste Processing components.

The system has been evaluated for waste processing a 102% power core inventory. Therefore, it is concluded that the radioactive waste disposal system will perform its design function at the MUR power uprate.

References (Section VII.6.8):

VII.6.8.1 USAR Section 11
VII.6.8.2 USNRC SER of FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221).

VII.6.9 Radiological Dose Monitoring and Radiological Dose Control Programs

The proposed power uprate will not result in radiation exposures in excess of the criteria (for restricted and unrestricted areas) provided in 10 CFR 20 "Standards for Protection Against Radiation" from an operations perspective, radiation levels are not expected to increase as a result of the power uprate. Individual worker exposures will be maintained within acceptable limits by the site ALARA program, which controls access to radiation areas.

Gaseous and Liquid Effluents were calculated based on a 1530 MWt core inventory, Reference VII.6.9.1. Offsite release concentrations and doses were found to be well within the limits of the current 10 CFR 20 and 10 CFR 50, Appendix I "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet Criterion "As Low as Reasonably Achievable" for Radioactive Materials in Light Water Cooled Nuclear Power Effluents by the site radioactive effluent control program. The core inventory has been evaluated for 1530 MWt and thus the power uprate will not result in an increase of inventory of radionuclides in the RCS. The fuel cladding will not be impacted by this uprate in a way that alters its structural integrity or leakage characteristics. The current Reference VII.6.9.1 effluent discharge calculations were conservatively based on 1% failed fuel and as such bound anticipated power uprate conditions.

<u>Normal Operations</u>
The maximum normal operation radiation sources will be unchanged from the values in effect prior to power uprate with possible exception of doses from neutron/gamma flux outside the reactor vessel (Reference VII.6.9.1). This only affects dose rates inside containment. TS 2.3 and 2.20 Limits the primary activity and primary to secondary leakage respectively. These TSs limit the normal radiation sources. The maximum allowable normal operation sources with exception to neutron/gamma flux are unchanged and bounded by existing TID design basis sources (EA-FC-01-020 R0, "AST Impact on the Post Accident Vital Area Access/Shielding"). The ex-vessel fluxes are considered bound based on conservative inputs used for the original design basis calculations. See Shielding section for further discussion.

<u>Normal Operation Offsite Doses</u>
TS 2.3 and 2.20 limits the primary and primary to secondary leakage, TS RETS also provides requirements for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. These requirements are implemented by the Offsite Dose Calculation Manual (ODCM) and plant procedures per Reference VII.6.9.1. Additionally a review of recent Annual Radioactive Effluent Release Reports demonstrates that the actual releases from the plant are historically a very small percentage of the allowable limits. Therefore, the offsite doses from normal effluent releases will remain significantly below the bounding limits of 10 CFR 20 for power uprate.

<u>Shielding for Normal Operations</u>
The power uprate has no impact on shielding designs or radiation zones outside containment. The plant TSs control the normal radiation sources to less than design basis values assumed in Reference VII.6.9.1.

The dose rates inside containment may increase by the same percentage as the power uprate under uprated conditions due to the increase in neutron/gamma flux. The increase in neutron/gamma flux is not expected to be significant because of current fuel management techniques that reduce neutron flux leakage at the perimeter of the core (extreme low radial leakage core). It is anticipated that the actual flux conditions could be lower than the original

design basis calculations in Reference VII.6.9.1. The values reported in Reference VII.6.9.1 were based on a very conservative axial peaking profile of 1.8 with no reduction in radial neutron leakage. Since the original design calculations (April 1967) OPPD has opted for extreme low radial leakage patterns that limit the flux to the perimeter of the core, and the axial peaking factors are not greater than 1.2-1.3. Thus, current fuel management programs would limit the flux and are thereby lower than the actual original design basis dose rate ex-vessel.

Current occupational doses are below the dose criteria of 10 CFR 20, and will remain below 10 CFR 20 limits following uprate. The source term calculated for the AST (at 1530 MWt) was determined to bound the existing shielding source term (i.e. the existing shielding source term was more limiting), and is thus appropriate for power uprate. Individual worker exposures will be maintained within acceptable limits by the site ALARA program, which control radiation areas.

Vital Access
Vital access dose considerations were evaluated recently in regards to implementation of FCS AST against the original design basis TID source terms (EA-FC-01-020). It was determined in this analysis, that the FCS AST source term at 1530 MWt was bounded by the existing TID source term. As such, the evaluation concluded that there was no impact on the post accident vital area access/shielding assessments previously documented. The MUR power uprate to 1.67% thus will not have an impact on the vital access calculations/assumptions.

References (Section VII.6.9):

VII.6.9.1 USAR Section 11
VII.6.9.2 USNRC SER of FCS license amendment (T. S. Amendment 201) for implementation of R. G. 1.183 in NRC Letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB 1221).
VII.6.9.3 EA-FC-01-020

VII.6.10 Coatings

Coatings used within the containment were specified to withstand accident conditions. Evidence that all primers and top coats stated in Reference VII.6.10.1 have been tested or are acceptable for their service applications is shown by Carboline Corporation and DuPont specification and test sheets. The containment building is designed to withstand an internal pressure of 60 psig at 305°F including all thermal loads resulting from the temperature associated with this pressure (Reference VII.6.10.2). The coatings within containment will not be impacted by a power uprate condition as the mass and energy values are not changed for MUR condition from that previously analyzed as part of post DBA (LOCA and MSLB, Reference VII.6.10.3). The MUR will not have any impact on the Coatings Program.

References (Section VII.6.10):

VII.6.10.1 USAR 5.2
VII.6.10.2 USAR Appendix G, criterion 10 and 49
VII.6.10.2 USAR 14

VII.6.11 Alloy 600 Program

Industry experience in PWRs has shown that Alloy 600 (Inconel 600) components and Alloy 82/182 weld filler metals are susceptible to primary water stress corrosion cracking (PWSCC). The Alloy 600 assessment provided the technical basis for the FCS Alloy 600 Program Basis Document (PDB-18, R0). This program includes all Alloy 600 components and Alloy 82/182 welds that are part of the reactor coolant system (RCS) pressure boundary, integral attachments to the RCS pressure boundary, or can have a direct or indirect effect on the integrity of the RCS pressure boundary. These components include: partial penetration welded nozzles and penetrations in the RCS fabricated from Alloy 600 material, J-groove welds made with Alloy 82 or 182 filler metal, full-penetration welds made with Alloy 82, 82 and 182 filler metal, and Alloy 600 piping components such as safe ends, non-pressure boundary Alloy 600 components such as welded internal attachments to vessels, and thermal sleeves, are also within the scope of the assessment. Steam generator tubes, and the associated tube-to-tubesheet seal welds, are specifically excluded from this program.

This program has assessed the Alloy 600 components and for each of them has documented the risks of failure. As part of the program system reliability is evaluated with respect to potential for equipment degradation. The system reliability is in part based on FCS susceptibility modeling of Alloy 600 components. PWSCC has been shown to be predominantly temperature and environmentally dependent. As such, with an increase in RCS temperature Alloy 600 susceptibility could potentially be challenged. Therefore, a review was performed on the impact of a temperature increase as a result of a power uprate with regards to Alloy 600 susceptibility.

As part of the MUR power uprate it was determined that the RCS temperature would only increase by 0.8°F on the hot leg. The RCS pressure, flow, and cold leg temperatures would remain the same. Thus, it is anticipated worst case scenario that the overall increase experienced by Alloy 600 materials is a 0.8°F increase. The review of this increase on the Alloy 600 components program assessments which are consistentconsistent with the applicable regulations concluded that, this increase in temperature affects the Alloy 600 component aging but has an insignificant impact on the components risk of failure.

Reference (Section VII.6.11):

VII.6.11.1 FCS Alloy 600 Program Basis Document (PDB-18, R0)

VII.6.12 Steam Generator Program

The purpose of the Steam Generator Program is to ensure tube structural and leakage integrity through the implementation of the following program elements:

- Assessment of existing degradation mechanisms in the reactor coolant pressure boundary within the steam generator
- Steam generator inspection in accordance with the EPRI PWR steam generator examination guidelines
- Assessment of tube integrity after each steam generator inspection to ensure that the performance criteria for the operating period have been met and will continue to be met for the next period
- Maintenance, plugging and repairs of steam generator tubes
- Primary to secondary leakage monitoring

- Maintenance of steam generator secondary side integrity
- Primary Side and Secondary Side Water Chemistry
- Foreign Material exclusion
- Self-assessment of the steam generator program
- Preparation of NRC and Industry reports

A review of the steam generator program elements has concluded that the program elements are symptom based, augmented by regular inspections, maintenance and chemistry activities, and industry experiences. At the MUR power uprate conditions, the steam generator tubes will be exposed to a 0.8°F increase in T_{hot} temperature. This temperature increase will marginally increase the stress corrosion cracking rate in the steam generator tubes. The existing plugging margin and inspection program elements are sufficient to ensure tube integrity. The steam generator program elements are independent of the reactor thermal power and therefore, the steam generator program elements will not be affected by the 1.67% MUR power uprate.

References (Section VII.6.12):

VII.6.12.1 PED-SEI-42 "Steam Generator Program"
VII.6.12.2 EA-FC-00-141 Rev 0 "Steam Generator Program"

VII.6.13 Containment Leak Rate Program

The FCS Containment Leak Rate Testing program performs the type A, B, and C containment leakage testing to verify the integrity of the containment and those systems and components which penetrate the containment walls.

The 1.67% MUR power uprate does not impact the scope, requirements or criteria of the containment leak rate testing program. Additionally, the operating condition changes associated with the MUR power uprate do not affect the containment or the systems and components which penetrate the containment walls. The containment pressure following a design basis accident from the MUR power uprate conditions is bounded by the analysis of record performed at 102% thermal power (Reference VII.6.14.2). Therefore, the containment leak rate testing program is not affected by the MUR power uprate.

Reference (Section VII.6.13):

VII.6.13.1 PBD - 5 "Containment Leak Rate"
VII.6.13.2 USAR – 14.15

VII.6.14 Zinc Injection

An assessment was performed to indicate whether the MUR power uprate had an impact on the zinc injection program. It was determined that there was no impact on the zinc injection system as a result of MUR.

VII.6.15 Sampling Systems

The design temperatures and pressures for the primary and secondary sampling systems valves, piping, accumulators and tanks are provided in Reference VI.6.15.1. This attachment was reviewed with respect to impact of the MUR on system performance and design conditions.

It was determined that the operating conditions anticipated for MUR would be bound by the design and maximum operating design conditions identified for the sampling components. The functional design of the sampling systems will not be impacted by MUR (Reference VI.6.15.2). The sampling systems are designed to permit chemists to control the flow for sampling, and will not be impacted by any minor changes that may be seen as a result of MUR (i.e. possible 0.8°F increase on primary sampling). None of the sampling equipment setpoints will require changing as a result of the MUR.

References (Section VII.6.15):

VII.6.15.1 SDBD-SL-135, Attachment 4
VII.6.15.2 USAR 9.13

VII.7 Mechanical Piping Design

Maximum design pressures and temperatures will not change as result of the 1.67% power uprate. The design pressures and temperatures bound the MUR power uprate operating pressures and temperatures. Therefore, existing code piping analyses are not affected by the proposed power uprate and will have no effect on qualification or adequacy of piping components. No changes are required to the mechanical piping design and code piping analyses as a result of the MUR Uprate Program.

VIII. Changes to Technical Specifications, Protection System Settings, and Emergency System Settings

The proposed license amendment would revise the FCS OL and TS to increase licensed power level to 1525 MWt, or 1.67% greater than the current level of 1500 MWt. The proposed changes, which are indicated on the marked-up pages in Attachment 8 and 9, are described below:

Revise Paragraph 3.A. in OL DPR-40 to authorize operation at a steady state reactor core power level not in excess of 1525 MWt.

Revise the definition of RATED POWER in TS to reflect the increase from 1500 MWt to 1525 MWt.

Corresponding TS Bases changes are also proposed:

In the Basis to TS 2.1.6, page 2.15a and 2-16, Pressurizer and Main Steam Safety Valves, change all instances of "1500 MWt to "RATED POWER."

In the Basis to TS 3.5, page 3-51, replace "a reactor power level of 1500 MWt" with "RATED POWER."

LIC-03-0067 Attachment 4

Calorimetric Uncertainty Evaluation Non-Proprietary Version

TABLE OF CONTENTS

1.0 PURPOSE

This analysis determines the overall calorimetric uncertainty associated with plant computer application XC105. The analysis accounts for the instrument uncertainties associated with the independent variables that have the largest impact on the calculation of calorimetric power. These variables are:

Feedwater Flow
Feedwater Temperature
Steam Generator Pressure
Steam Generator Moisture Carryover
Steam Generator Blowdown Flow
Steam Generator Blowdown Temperature

This analysis combines the individual instrument uncertainties associated with the independent variables to determine the overall calorimetric uncertainty. The individual contributions to the power uncertainty are combined using a statistical summation to determine the total power measurement uncertainty. This approach is consistent with the methods described in ANSI/ISA-67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation" (Reference 2.3)

This calculation is applicable for use with the CROSSFLOW ultrasonic feedwater flow measurement system.

2.0 REFERENCES

The following are references used in developing this document.

2.1 OPPD Production Engineering Division Procedure, "Calculation Preparation, Review and Approval", PED Quality Procedure QP-3, Revision 3, dated 4/8/94.

2.2 OPPD Production Engineering Division Standard, "Instrument Loop Uncertainty Setpoint / Tolerance Calculation Methodology" Document Number EEI-3.

2.3 ANSI/ISA-67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation"

2.4 FC06898, "Steam Generator Pressure and Feedwater Temperature Instrument Uncertainty Analysis", Rev.0

2.5 FC06907, "Steam Generator Blowdown Flow and Temperature Instrument Uncertainty Analysis" Rev.0

2.6 Basic Engineering Data Collection and Analysis, Vardeman/Jobe.

2.7 "Steam Generator Moisture Carryover Test Results", CE-18074-2087 dated August 14,1986.

2.8 Calculation FC06091, "Uncertainties Report for Fort Calhoun Station Secondary Calorimetric".

2.9 "Applied Numerical Analysis", Sixth Edition, Curtis F. Gerald and Patrick O. Wheatley

2.10 "Improved Flow Measurement Accuracy using Crossflow Ultasonic Flow Measurement Technology", CENPD-397-P-A Rev.01.

2.11 "Upgraded ERFCS Functional Requirements Specification", Rev.1.02.20.02.

2.12 ASME Steam Tables

3.0 ASSUMPTIONS AND GIVEN CONDITIONS

The following assumptions and given conditions (A&GC) are used in development of this calculation.

4.0 METHOD OF CALCULATION

4.1 ENERGY AND FLOW EQUATIONS

4.1.1 STEAM GENERATOR ENERGY EQUATION

4.1.2 STEAM GENERATOR FEEDWATER FLOW EQUATION

4.1.3 STEAM GENERATOR BLOWDOWN FLOW EQUATION

4.2 CALCULATION OF CALORIMETRIC UNCERTAINTIES

4.2.1 INSTRUMENT UNCERTAINTIES

4.2.2 WEIGHTING FACTORS

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4.2.3 COMBINING ERROR TERMS

4.3 ONE SIDED 95% CONFIDENCE INTERVAL

4.4 NSS ENERGY LOSS AND ELECTRICAL ENERGY WEIGHTING FACTORS

5.1 INPUT VARIABLES

The following values are used in this analysis to calculate weighting factors for each independent variable. The values provided are for the nominal (100%) power values of the independent variables used in the energy equation. Also provided are the incremental values used to determine the weighting factors.

5.2 EQUATIONS

5.3 WEIGHTING FACTORS

5.3.12

5.3.13										

5.3.14										

5.4 CALORIMETRIC UNCERTAINTY

5.5 NSSS ENERGY LOSS AND ELECTRICAL ENERGY WEIGHTING FACTORS

5.5.1

5.5.2

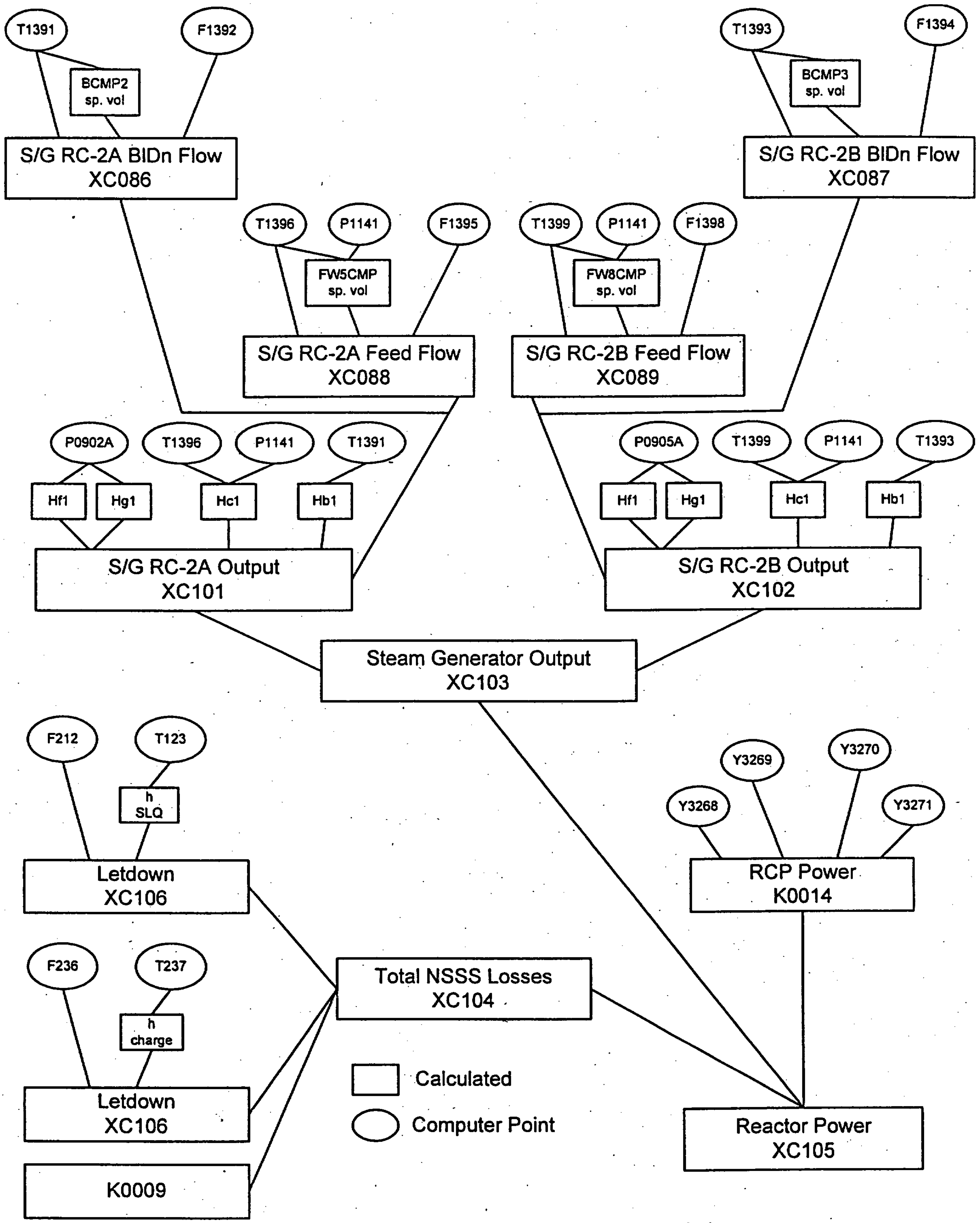
6.0 CONCLUSIONS

This calculation determined the calorimetric uncertainty based on an upgrade of feedwater flow and temperature instrumentation which is used for the XC105 calculation.

Allowable Uprate: 2%-0.2694% = 1.7306%

Based on this an uprate of 1.67% is acceptable and enveloped by this analysis.

ATTACHMENT 1



Block Diagram of XC105 Calculation

ATTACHMENT 2

26/28

Reactor Power

XC105 Values	
Time Interval	MW thermal
Real Time	1502.87
2 Minute Average	1497.52
5 Minute Average	1497.87
10 Minute Average	1498.23
30 Minute Average	1498.10
1 Hour Average	1498.24
2 Hour Average	1498.29
4 Hour Average	1498.48
8 Hour Average	1498.61

XC105 Inputs			
Parameter	Value	Units	
S/G 2A Blowdown Temperature	499.25	°F	T1391
S/G 2A Blowdown Flow	4.54	" water	F1392
S/G 2A Blowdown Flow	25,555	lb/hr	F1392C
S/G 2B Blowdown Temperature	497.25	°F	T1393
S/G Blowdown Flow	4.48	" water	F1394
S/G Blowdown Flow	25,406	lb/hr	F1394C
S/G 2A Feedwater Temperature	442.31	°F	T1396
S/G 2A Feedwater Flow	359.00	" water	F1395
S/G 2A Feedwater Flow	3,315,690	lb/hr	F1395C
S/G 2B Feedwater Temperature	441.88	°F	T1399
S/G 2B Feedwater Flow	368.50	" water	F1398
S/G 2B Feedwater Flow	3,365,500	lb/hr	F1398C
Feedwater Pressure	1,001.56	psig	P1141
S/G 2A Pressure	817.19	psia	P0902A
Regen Heat Exchanger Temperature	403.70	°F	T237
Letdown Flow	34.00	gpm	F212
S/G 2B Pressure	807.81	psia	P0905A
Charging Pump Flow	38.73	gpm	F236

ERF Display: XCF, MWT

Return to Plant Information Page

Return to FCS Home Page

ATTACHMENT 3

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LIC-03-0067 Attachment 5

Affidavit for Calorimetric Uncertainty Evaluation

I, Tim Leibel, depose and say that I am a Principal of PL Integrated Services LLC (PLIS), duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and described below. I have personal knowledge of the criteria and procedures utilized by PLIS in designating information as a trade secret, privileged, or as confidential commercial or financial information.

This affidavit is submitted in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding proprietary information. The information for which proprietary treatment is sought, and which document has been appropriately designated as proprietary, is *Calculation Number* FC-6896-P "Secondary Calorimetric Uncertainty Analysis".

Pursuant to 10 CFR 2.790(b)(4) of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information included in the document listed above should be withheld from public disclosure.

1. The information sought to be withheld from public disclosure is owned and has been held in confidence by PLIS. It consists of the methodologies concerning the technical basis and implementation of a secondary calorimetric uncertainty analysis for the Fort Calhoun Station.

2. The information consists of methodologies for the development and implementation of a secondary calorimetric uncertainty analysis, the application of which results in substantial competitive advantage to PLIS.

3. The information is of a type customarily held in confidence by PLIS and not customarily disclosed to the public.

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

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

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of PLIS because:

   a. A similar product or service is provided by competitors.
   b. PLIS has invested substantial funds and engineering resources in the development of this information. A competitor would have to undergo similar expense in generating equivalent information.
   c. The information consists of the technical basis and implementation of a secondary calorimetric uncertainty analysis for the Fort Calhoun Station, the application of which provides PLIS a competitive economic advantage. The availability of such information to competitors would enable them to design their product or service to better compete with PLIS, take marketing or other actions to improve their product's position or impair the position of PLIS product, and avoid developing similar technical analysis in support of their processes, methods or apparatus.

Tim Leibel
Principal
PL Integrated Services LLC

Sworn to before me this 13th day of June 20003

William J. Ponec, Notary Public

My commission expires: April 4, 2004

GENERAL NOTARY-State of Nebraska
WILLIAM J. PONEC
My Comm. Exp. April 4, 2004

LIC-03-0067 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))

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:	D. M. Brown, Manager PWR Engineering	6/16/03 Date
Approved:	for D. E. Garber, Manager BWR Engineering	6-13-03 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	[signature]	06/16/03

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

Change Number	Description of Change

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

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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.

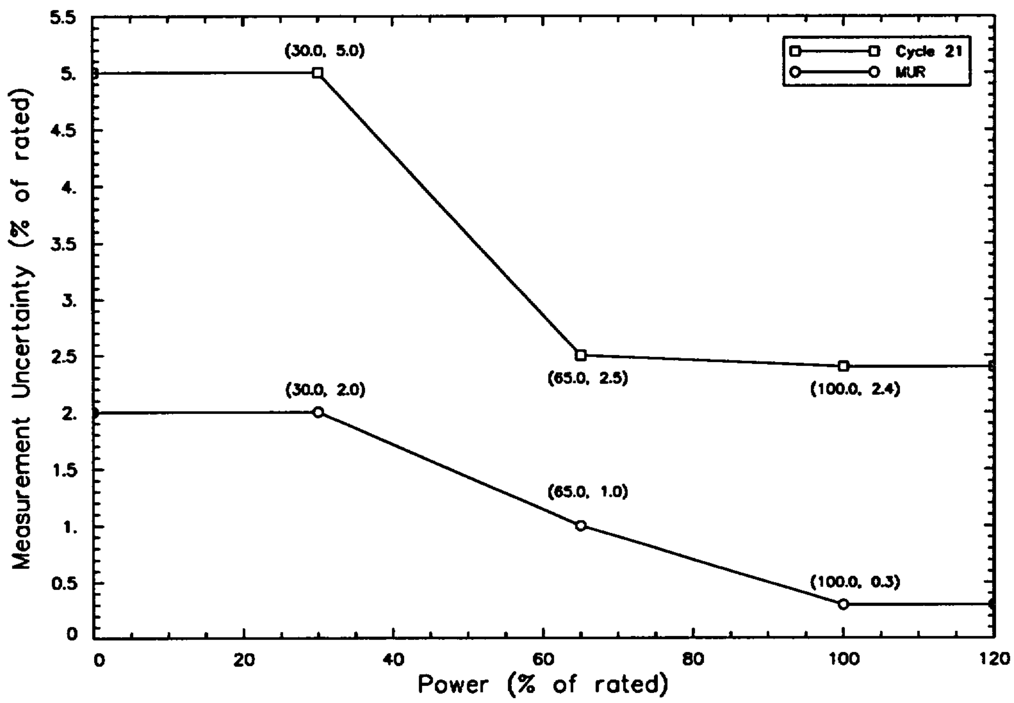

Figure 5.1 Power Measurement Uncertainty

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	$1.483\ \frac{\%\ \text{power}}{°F}$
K_β	$2.824 \times 10^{-3}\ \frac{\%\ \text{power}}{°F^2}$
K_γ	$2.866 \times 10^{-3}\ \frac{\%\ \text{power}}{°F^2}$
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).

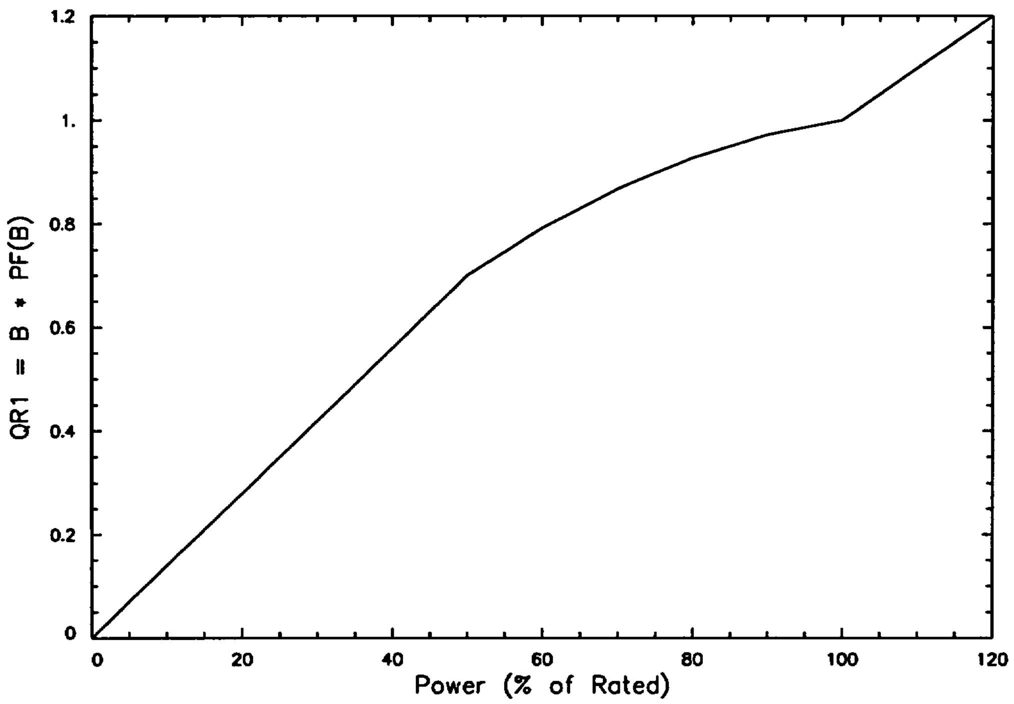

Figure 5.2 TM/LP QR_1 Function (B · PF(B))

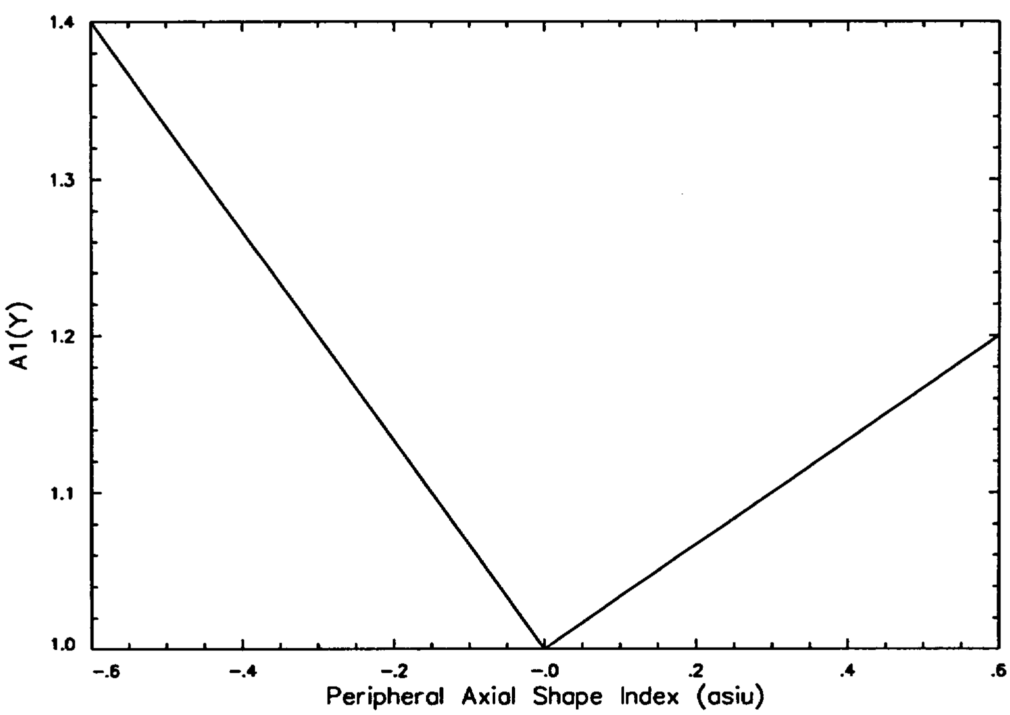

Figure 5.3 TM/LP $A_1(Y)$ Function

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.

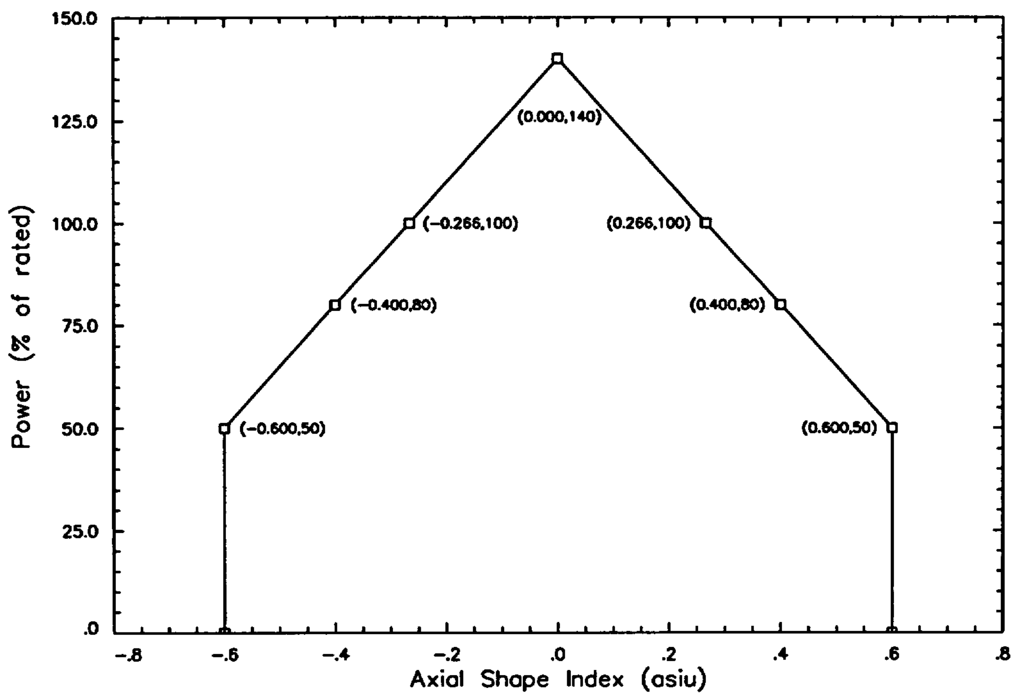

Figure 5.4 APD LSSS "Barn" (Plant Settings)

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.

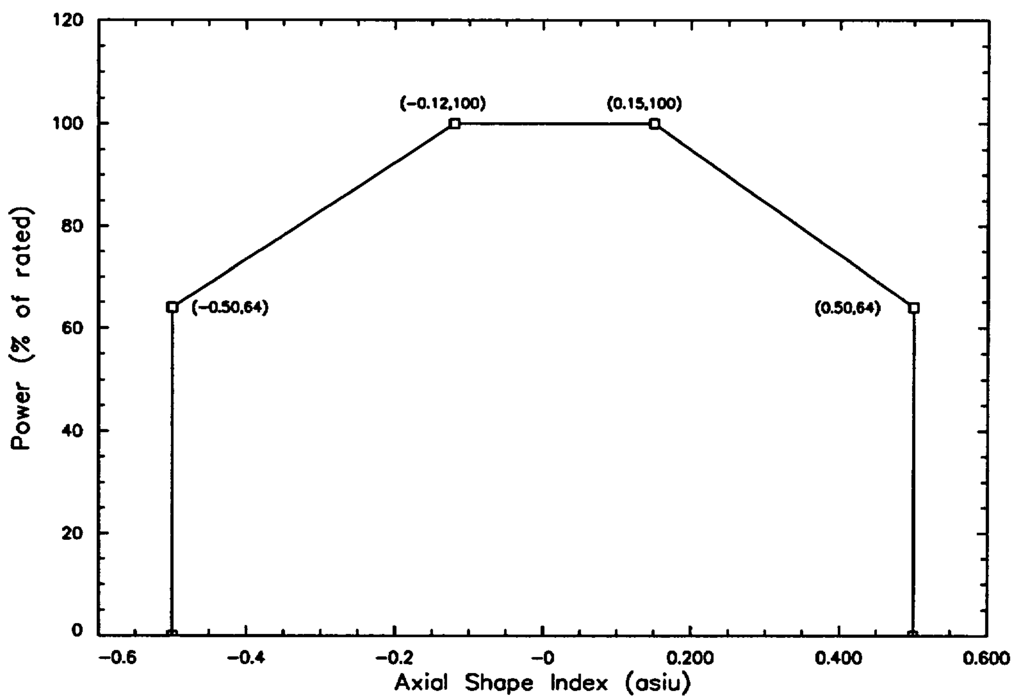

Figure 5.5 DNB/BASSS LCO "Barn" (Plant Settings)

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

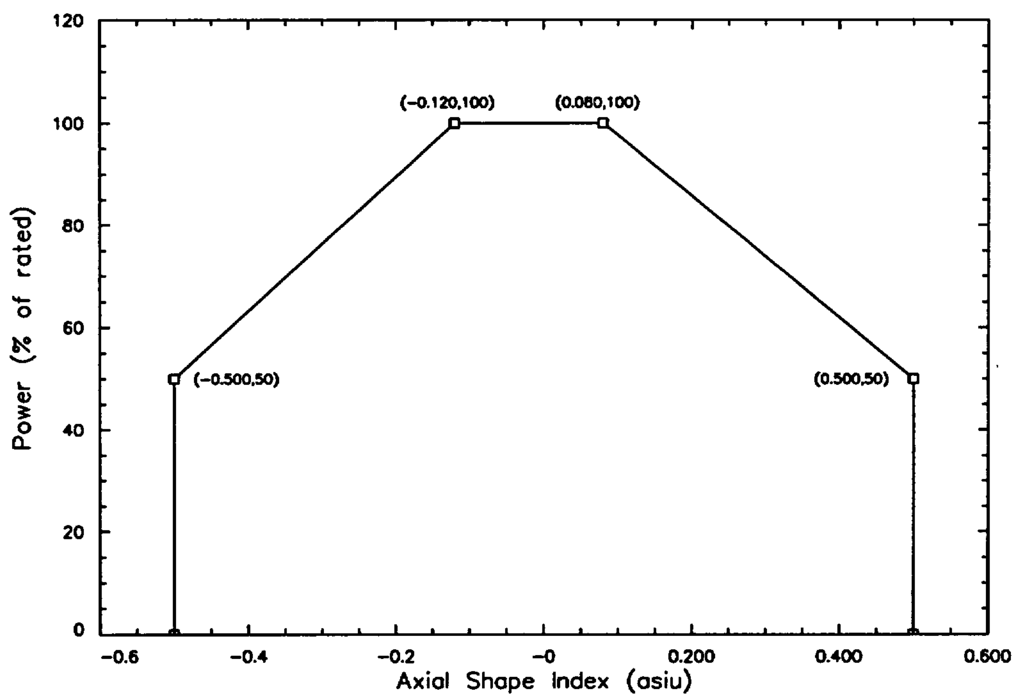

Figure 5.6 Excore LHR LCO "Barn" (Plant Settings)

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

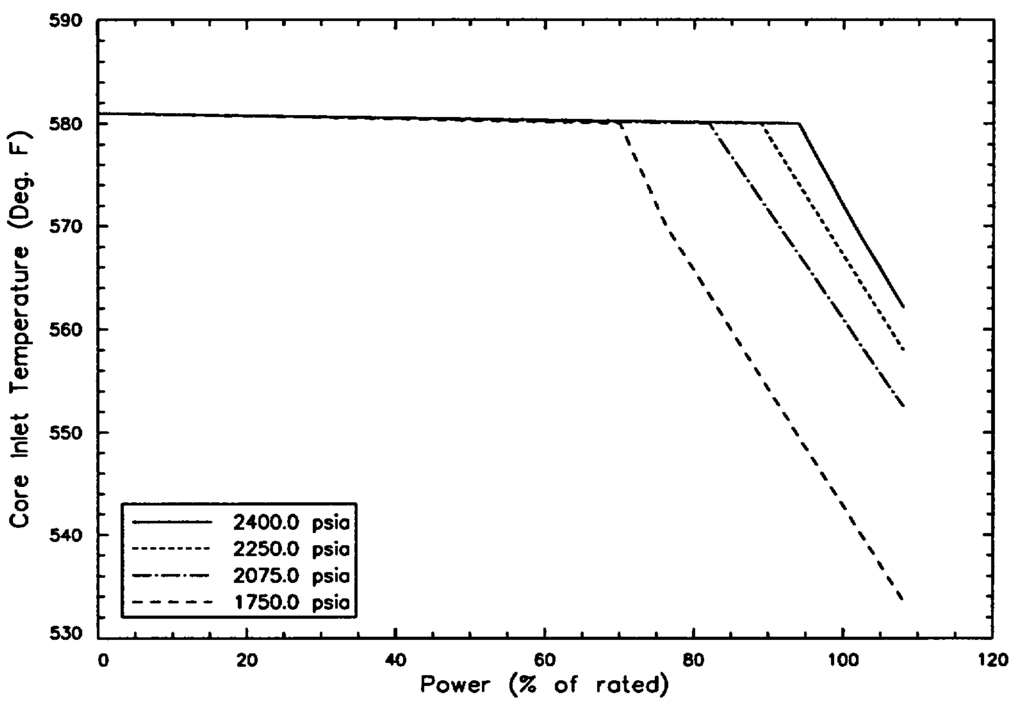

Figure 5.7 Thermal Margin/Low Pressure Limit Lines

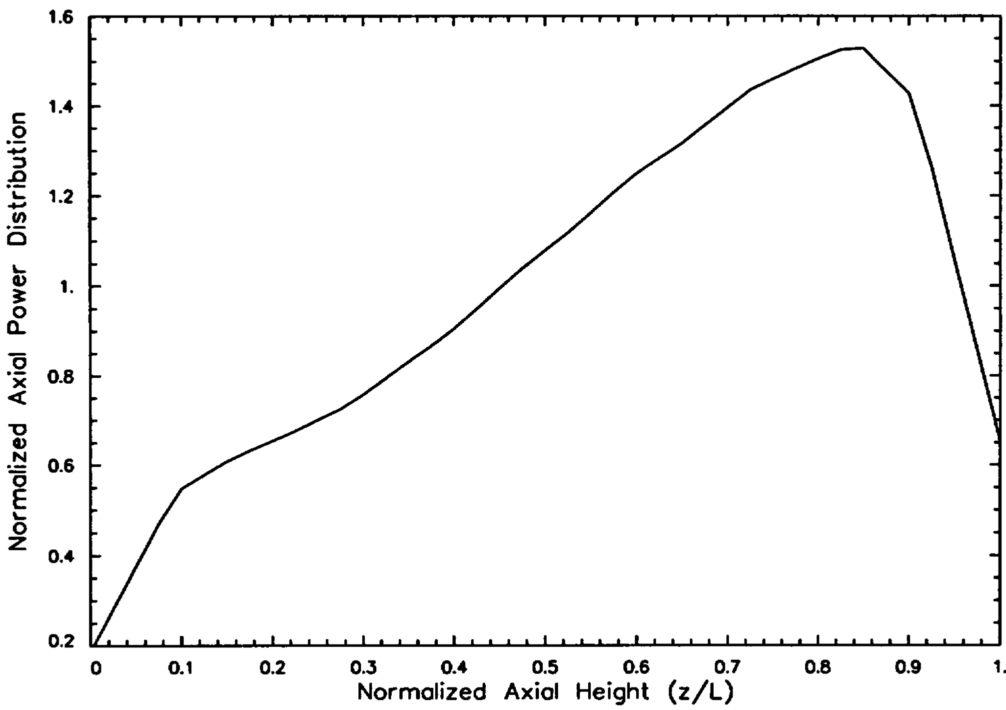

Figure 5.8 Thermal Margin/Low Pressure Limit Line Design Axial

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 (2.824×10^{-3} $\frac{\%\ \text{power}}{°\text{F}^2}$ and 2.866×10^{-3} $\frac{\%\ \text{power}}{°\text{F}^2}$, 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.

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Document Control

LIC-03-0067 Attachment 7

Westinghouse Reactor Vessel Structural Evaluation

Westinghouse Reactor Vessel Structural Evaluation

Core Shroud Under Revised Thermal Loadings[10]

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.

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[10] **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[11]	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[12]	Primary	15,242	22,200	31
	Primary plus Secondary	31,541	44,400	29
Core Shroud Panel @ 6th Girth Rib[13]	Primary	4,580	22,099	79
	Primary plus Secondary	14,197	44,400	68
Girth Rib Flexure[14]	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

[11] Allowable stress = as-irradiated yield stress rather than 3 × Sm for bolting material, as used in Ref. 2.
[12] Fatigue usage = .01 < 1.0 allowable fatigue usage
[13] Fatigue usage = .217 < 1.0 allowable fatigue usage
[14] 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 6th 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:

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

for $T_c \le T \le T_c + 200$

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.

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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~~ 1525 megawatts thermal (rated power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 215, 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

A steady state reactor core output of ~~1500~~ MWt.
1525

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)

d. With both PORVs inoperable in Modes 4 or 5 depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

(5) Two power-operated relief valves (PORVs) and their associated block valves shall be operable in Modes 1, 2, and 3.

a. With one or both PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore PORV to operable status or close its associated block valve and remove power from the block valve; restore the PORV to operable status within the following 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN with the following 36 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to operable status or close both block valves, remove power from the block valves, and be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to operable status or place the associated PORV(s) in the closed position. Restore at least one block valve to operable status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to operable within 72 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

RATED POWER

Basis

The highest reactor coolant system pressure reached in any of the accidents analyzed resulted from a complete loss of turbine generator load without simultaneous reactor trip while operating at ~~4500 MWt~~.[2] This pressure was less than the 2750 psia safety limit and the ASME Section III upset pressure limit of 10% greater than the design pressure.[1] The reactor is assumed to trip on a "High Pressurizer Pressure" trip signal.

The pressurizer safety valves are required to be calibrated to within ± 1% of the specified setpoint value using ASME Section XI test methods. ASME Section XI requires that valves in steam service use steam as the test medium for establishing the setpoint. With the presence of a water-filled loop seal, establishing the valve setpoint with steam may result in in-situ valve actuation at pressures outside the ± 1% tolerance specified. Under transient conditions, it is expected that the valve(s) will actuate at no less than 4% below, nor greater than 6% above, the specified setpoint, which is within the tolerance assumed in the safety analysis.[2] These analyses are based on a minimum of any four of the five main steam safety valves on each main steam header being OPERABLE.

The power-operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

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]~~ 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,~~ 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.

RATED POWER

RATED POWER

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.

at RATED POWER,

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, ~~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.

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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 1525 megawatts thermal (rated power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 215, 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

A steady state reactor core output of 1525 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.

d. With both PORVs inoperable in Modes 4 or 5 depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

(5) Two power-operated relief valves (PORVs) and their associated block valves shall be operable in Modes 1, 2, and 3.

a. With one or both PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore PORV to operable status or close its associated block valve and remove power from the block valve; restore the PORV to operable status within the following 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN with the following 36 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to operable status or close both block valves, remove power from the block valves, and be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to operable status or place the associated PORV(s) in the closed position. Restore at least one block valve to operable status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to operable within 72 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

Basis

The highest reactor coolant system pressure reached in any of the accidents analyzed resulted from a complete loss of turbine generator load without simultaneous reactor trip while operating at RATED POWER.[2] This pressure was less than the 2750 psia safety limit and the ASME Section III upset pressure limit of 10% greater than the design pressure.[1] The reactor is assumed to trip on a "High Pressurizer Pressure" trip signal.

The pressurizer safety valves are required to be calibrated to within ± 1% of the specified setpoint value using ASME Section XI test methods. ASME Section XI requires that valves in steam service use steam as the test medium for establishing the setpoint. With the presence of a water-filled loop seal, establishing the valve setpoint with steam may result in in-situ valve actuation at pressures outside the ± 1% tolerance specified. Under transient conditions, it is expected that the valve(s) will actuate at no less than 4% below, nor greater than 6% above, the specified setpoint, which is within the tolerance assumed in the safety analysis.[2] These analyses are based on a minimum of any four of the five main steam safety valves on each main steam header being OPERABLE.

The power-operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

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-03-0067 Attachment 10

List of Regulatory Commitments

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.