January 16, 2004

Mr. R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MC0029)

Dear Mr. Ridenoure:

The Commission has issued the enclosed Amendment No. 224 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 18, 2003, as revised by letter dated August 28, 2003, and supplemented by letters dated October 31 and December 15, 2003.

The amendment revises the renewed operating license and the TSs to increase the licensed rated power by 1.6 percent from 1500 megawatts thermal (MWt) to 1524 MWt.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 224 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

January 15, 2004

Mr. R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MC0029)

Dear Mr. Ridenoure:

The Commission has issued the enclosed Amendment No. 224 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1 (FCS). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 18, 2003, as revised by letter dated August 28, 2003, and supplemented by letters dated October 31 and December 15, 2003.

The amendment revises the renewed operating license and the TSs to increase the licensed rated power by 1.6 percent from 1500 megawatts thermal (MWt) to 1524 MWt.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

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cc w/encls: See next page

*Date of SE Input Memo
**For previous concurrences see attached ORC

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee) dated July 18, 2003, as revised by letter dated August 28, 2003, and supplemented by letters dated October 31 and December 15, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Renewed Facility Operating License No. DPR-40 is amended by changes to the Operating License and the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 224, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance. Modifications associated with the measurement uncertainty recapture power uprate include: (1) implementation of control room alarm functions, and (2) Figure 2-1 of the Pressure-Temperature Limits Report will be revised prior to the reactor vessel reaching 39.9 effective full power years of operation.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and
Technical Specifications

Date of Issuance: January 16, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 224

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following page of Renewed Facility Operating License No. DPR-40 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
1	1
2-16	2-16
3-51	3-51

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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 in excess of 1524 megawatts thermal (rated power). |

B. Technical Specifications

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

C. Security and Safeguards Contingency Plans

The Omaha Public Power District 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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 224

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated July 18, 2003, as revised by letter dated August 28, 2003, and supplemented by letters dated October 31 and December 15, 2003, Omaha Public Power District (OPPD/the licensee) requested an amendment to the operating license (OL) and technical specifications (TSs) for the Fort Calhoun Station, Unit 1 (FCS). The proposed amendment would increase the licensed reactor core power level by 1.6 percent from 1500 megawatts thermal (MWt) to 1524 MWt. The proposed increase is considered a measurement uncertainty recapture (MUR) power uprate. The proposed changes are described below:

- Revise paragraph 3.A of the Renewed Facility Operating License No. DPR-40 to authorize operation at reactor core power levels not in excess of 1524 MWt.

- Revise TS 1.0, Rated Power, to reflect the increase from 1500 MWt to 1524 MWt.

The corresponding TS Bases changes are:

- TS 2.1.6 Basis, "Pressurizer and Main Steam Safety Valves" – change all instances of "1500 MWt" to "RATED POWER."

- TS 3.5 Basis – replace "a reactor power level of 1500 MWt" with "at RATED POWER."

The October 31 and December 15, 2003, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 18, 2003 (68 FR 54751).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. Title 10 of the *Code of Federal Regulation* (10 CFR), Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in the analyses. Appendix K to 10 CFR Part 50 allows licensees to assume a power level lower than 1.02 times the licensed power level (but not less than the licensed power level), provided licensees have demonstrated that the proposed value adequately accounts for instrumentation uncertainties. In its application, the licensee proposed to use a value of 1.004. To achieve this level of accuracy, the licensee will install the more accurate feedwater flow measurement meter by Westinghouse and the more accurate feedwater temperature instrumentation by Rosemont. Both these changes are consistent with NRC-approved Westinghouse Topical Report (TR) CENPD-397-P, Revision 01-P, "Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology." The NRC staff approved Westinghouse TR CENPD-397-P, Revision 01-P by a safety evaluation (SE) dated March 20, 2000.

The licensee proposed to increase the power output of the plant by the difference between the 1.02 multiplier of 10 CFR Part 50, Appendix K, that the licensee previously complied with and the 1.004 multiplier that the licensee proposed in its August 28, 2003, application as a result of the installation of the more accurate flowmeter and resistance thermal detectors (RTDs). Since the analyses of record for LOCA and ECCS performance assumed a power level of 1.02 times the licensed power level, a 1.6 percent increase in power could be achieved without necessitating reanalyses of these events. Other design-basis analyses are evaluated to ensure an appropriate accounting of power level uncertainties.

3.0 EVALUATION

The NRC staff's evaluation of the proposed MUR power uprate is based on the guidance provided by NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Applications." RIS 2002-03 delineates the appropriate scope and level of detail for the review of an MUR power uprate application. In keeping with the guidance in RIS 2002-03, the NRC staff has evaluated the licensee's application by considering whether the proposed MUR power uprate conditions are bounded by existing design and licensing bases analyses. In particular, the NRC staff considered whether the current analyses of record were performed at 102 percent of the current licensed power level (or a higher power level). Reduction in power level uncertainty through the reduced instrumentation error, as permitted by Appendix K, does not affect the results of such analyses, provided other assumptions upon which the analyses rest remain valid.

For every technical area where the proposed MUR power uprate conditions are bounded by existing design and licensing bases analyses, the NRC staff has confirmed that the proposed conditions will continue to be bounded and has provided a table which summarizes

- the topics identified in RIS 2002-03 within each primary technical area;

- where the topic is addressed in the licensee's application (unless otherwise indicated);
- where the topic is addressed in the plant's Updated Safety Analysis Report (USAR);
- references to NRC documents which describe analyses that bound the proposed conditions; and
- the NRC's conclusion of acceptability.

The corresponding references and notes for each table immediately follow the table.

For situations where the proposed MUR power uprate conditions are not bounded by existing design and licensing bases, the licensee has assessed the impact of the proposed MUR power uprate on the design and licensing bases. The NRC staff has noted each such area in the tables and has reviewed and evaluated the licensee's assessments. The NRC staff's review included an evaluation of the application of the methodologies used by the licensee for the assessments.

In several places in this SE, the NRC staff refers to NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition," as guidance used during the review. The NRC staff notes that the SRP was used solely for general technical guidance. The licensee's application was reviewed to determine if the plant's licensing basis was in compliance with the Commission's regulatory requirements, not NUREG-0800.

3.1 Instrumentation and Controls

3.1.1 Regulatory Evaluation

The NRC staff's review in the area of instrumentation and controls covers (1) the proposed plant-specific implementation of the feedwater flow measurement device, and (2) the power uncertainty calculations (RIS 2002-03, Attachment 1, Section I). The NRC staff's review is conducted to confirm that the licensee's use of TR CENPD-397-P, Revision 01-P is consistent with the NRC staff's approval of the topical report. The NRC staff also reviewed the power uncertainty calculations to ensure that (1) the proposed uncertainty value of 0.4 percent correctly accounts for the uncertainties due to power level instrumentation error, and (2) the calculations meet the relevant requirements of Appendix K to 10 CFR Part 50.

3.1.2 Technical Evaluation

In Section I of Attachment 1 to RIS 2002-03, the NRC staff identified information that a licensee should provide in relation to the proposed feedwater measurement technique in order to allow the NRC staff to conduct a review of an MUR power uprate.

The generic bases for the proposed MUR power uprate are provided in TR CENPD-397-P, Revision 01-P. This TR covers the CROSSFLOW system ultrasonic flow meter (UFM) and the ability of this flow meter to achieve increased accuracy of feedwater flow measurement. In its application, the licensee submitted an uncertainty assessment of the accuracy with which

reactor core thermal power may be determined using the new flow meter. The new flow meter will be installed, calibrated, and maintained in accordance with the recommendations of TR CENPD-397-P, Revision 01-P. On the basis of the proposed installation it is anticipated that thermal power measurement uncertainty will not exceed 0.4 percent of rated thermal power (RTP). This anticipated uncertainty limit will be confirmed during the commissioning process following installation. Therefore, the original 2 percent margin required by Appendix K will be reduced to 0.4 percent, allowing an MUR power uprate of 1.6 percent. The NRC staff finds that the licensee has addressed the NRC's approved TR and the approved feedwater flow measurement technique for an MUR power uprate and, therefore, has complied with the guidance in Items A and B of Section I of Attachment 1 to RIS 2002-03.

The CROSSFLOW system UFM sensors at FCS are attached to a mounting bracket installed on the main feedwater supply header to the steam generators, consistent with the guidelines of TR CENPD-397-P, Revision 01-P. The CROSSFLOW sensors are installed approximately 54 pipe diameters downstream of the nearest elbow, in an area with fully developed flow conditions. A plant-specific plant computer interface has been developed for use with the CROSSFLOW system. The CROSSFLOW/ERFCS (plant computer) interface provides data between the ERFCS and the CROSSFLOW computer. This data link sends the required plant data from the ERFCS to the CROSSFLOW computer (which generates a correction factor for feedwater flow), and returns the feedwater flow correction factor to the ERFCS. The CROSSFLOW UFM sensors will be used for continuous calorimetric power determination by data link to the plant computer system. An audible and visual alarm will be provided to alert plant operators when the UFM sensors are out-of-service. All components installed conform to the guidelines in CENPD-397-P, Revision 01-P. Based on the above discussion, the NRC staff finds that the licensee has addressed the plant-specific implementation of the guidelines in the TR and, therefore, has complied with the guidance in Item C of Section I of Attachment 1 to RIS 2002-03.

In the NRC staff's SE that approved TR CENPD-397-P, Revision 01-P, the NRC staff requested licensees to address four criteria in their request for an MUR power uprate license amendment:

1. The licensee should discuss the development of maintenance and calibration procedures that will be implemented with the CROSSFLOW UFM installation. These procedures should include the process and contingencies for an inoperable CROSSFLOW UFM and the effect on thermal power measurement and plant operation.

2. For plants that currently have the CROSSFLOW UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and confirm that the instrumentation is representative of the CROSSFLOW UFM and is bounded by the requirements set forth in TR CENPD-397-P, Revision 01-P.

3. The licensee should confirm that 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. If an alternative methodology is used, the application should be justified and applied to both the venturi and the CROSSFLOW UFM for comparison.

4. The licensee of a plant at which the installed CROSSFLOW UFM was not calibrated to a site-specific piping configuration should submit additional justification. This justification should show that the meter installation is independent of the plant-specific flow profile for the stated accuracy or should show that the installation is equivalent to the known calibrations and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for a previously installed and calibrated CROSSFLOW UFM system, the licensee should confirm that the plant-specific installation follows the guidelines in the CROSSFLOW UFM TR.

In the license amendment application, the licensee addressed these four criteria as follows:

<u>Response to Criterion 1</u>: 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. A one-sided confidence interval methodology was utilized to determine the plant-specific calorimetric measurement uncertainty. The plant-specific accuracy based on plant-specific instrumentation is 0.4 percent. This number is based on test data taken at FCS by Westinghouse. The final number will be determined after the system is installed and prior to an increase in plant power.

CROSSFLOW UFM failure will be detected and transmitted to the plant computer and will cause an audible alarm in the control room. If the CROSSFLOW system is not returned to service within 24 hours, power will be reduced and maintained at the appropriate power level until the CROSSFLOW UFM system is returned to service. The FCS operation procedures and training program will be revised to reflect the CROSSFLOW system unavailability condition.

<u>Response to Criterion 2</u>: At FCS, the location of the CROSSFLOW UFM is representative of the location requirements set forth in TR CENPD-397-P, Revision 01-P. 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 an NRC request for additional information (RAI) regarding WCAP15689-P, "Evaluation of Transit Time and Cross Correlation Ultrasonic Flow Measurement Experience with Nuclear Plant Feedwater Flow Measurement," 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 degree elbow. Therefore, the FCS CROSSFLOW system when installed will satisfy the requirements of TR CENPD-397-P, Revision 01-P and will be bounded by them.

<u>Response to Criterion 3</u>: The methodology used to calculate the uncertainty of the CROSSFLOW UFM in comparison to the current feedwater flow instrumentation is based on the accepted plant setpoint methodology for developing instrument uncertainty in Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety-Related Systems" and Instrument Society of America (ISA) S67.04, "Setpoint for Nuclear Safety-Related Instrumentation," as described in TR EMF-1961-P-A, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors." OPPD has completed the uncertainty calculation with a mass flow accuracy of

0.4 percent of rated feedwater flow for the FCS-specific installation. The FCS CROSSFLOW uncertainty calculations are consistent with the methodology described in TR CENPD-397-P, Revision 01-P.

Response to Criterion 4: For FCS there will be no site-specific piping configuration calibration because the installation is equivalent to the 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 upstream flow disturbance to conform to the proprietary installation requirements of TR CENPD-397-P, Revision 01-P.

Based on the above discussion, the NRC staff finds that the licensee has fully addressed the four criteria specified in the NRC staff's SE of TR CENPD-397-P, Revision 01-P and, therefore, has complied with the guidance in Item D of Section I of Attachment 1 to RIS 2002-03.

The RTP uncertainty is calculated by combining the individual error terms that contribute to uncertainty using square root sum of squares (SRSS) methodology, as described in RG 1.105 and ISA S67.04.

Attachment 3 of the licensee's August 28, 2003, letter documented the detailed methods used for the determination of the error terms associated with the RTP uncertainty, and using the plant-specific data and plant-approved methodology to determine the total power measurement uncertainty.

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

Table 3.1.2
FCS Process Parameter Inputs to Reactor Thermal Power

Independent Variable	**Term**	**Uncertainty**	**Sensitivity**
Feedwater flow	UW_{FW}	0.3922%	1.0107
Feedwater temperature	UT_{FW}	0.69 °F	0.4136
Steam generator pressure	UP_{SG}	14.68 psia	0.0144
Steam generator moisture carryover	UM_{CO} A/B	0.11%/0.05%	0.0011/0.0008
Steam generator blowdown flow	UW_{BD}	1873 lbm/hr	0.0034
Steam generator blowdown temperature	UT_{BD}	2.94°F	0.0038

The spreadsheet calculation demonstrates that the FCS total power measurement uncertainty is bounded within ±0.4 percent.

By letter dated October 14, 2003, the NRC staff requested the licensee to provide an independent "recheck" calculation based on a 0.4 percent uncertainty case to verify that the numbers calculated in the spreadsheet equations are correct. By letter dated October 31, 2003, the licensee provided the "Independent Re-check of Calculations," which shows that

the numbers are in agreement with the spreadsheet calculation performed in the August 28, 2003, submittal. Based on the above discussion, the NRC staff finds that the licensee has provided a calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty, and therefore, has complied with the guidance in Item E of Section I of Attachment 1 to RIS 2002-03.

Item I.1.F of Attachment 1 to RIS 2002-03 requests the licensee to provide information of the calibration and maintenance procedures for all instruments that affect the power calorimetric with respect to:

- maintaining calibration,
- controlling software and hardware configuration,
- performing corrective actions,
- reporting deficiencies to the manufacturer, and
- receiving and addressing manufacturer deficiency reports.

By letter dated October 31, 2003, the licensee addressed calibration and maintenance procedures as summarized below:

1. Maintaining calibration – Calibration and maintenance will be performed using site procedures developed from the CROSSFLOW system technical manual and plant operating and maintenance manuals. All maintenance work will be performed in accordance with site work control procedures.

2. Controlling software and hardware configuration – Any proposed hardware or software changes related to the CROSSFLOW system and its calibration and maintenance procedures will be controlled and evaluated by the plant design change process. This design change process includes a 10 CFR 50.59 evaluation.

3. Corrective actions – Corrective actions involving maintenance will be performed by qualified maintenance personnel who are formally trained on the CROSSFLOW system. As with other maintenance and calibration activities, applicable deficiencies and corrective actions related to the CROSSFLOW system are documented in the FCS condition report (corrective action) system.

4. Reporting deficiencies to the manufacturer – Reliability engineering personnel will monitor the reliability of the CROSSFLOW system. Deficiencies are documented in the condition report system, and those deficiencies meeting the established criteria are reported to the manufacturer.

5. Receiving and addressing manufacturer deficiency reports - The CROSSFLOW system vendor (Westinghouse) shall inform OPPD of any deficiencies in accordance with the reporting requirements. Manufacturer deficiency reports will be noted in the condition report system. These activities are consistent with the requirements of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program."

Based on the above discussion, the staff finds that the licensee has addressed the calibration and maintenance aspects of the CROSSFLOW system and complied with the guidance in Item F of Section I of Attachment 1 to RIS 2002-03.

By letter dated October 31, 2003, in response to the NRC staff's question of allowable outage time, the licensee stated that if the CROSSFLOW UFM system becomes unavailable, the operator will enter an operating procedure that will direct the operator through the actions for a CROSSFLOW system failure. The procedure will require that a power range nuclear instrumentation channel adjustment surveillance test be performed within one hour of the CROSSFLOW system failure, using the last good correction factor. The CROSSFLOW system must then be returned to service prior to the next power range channel surveillance. If the CROSSFLOW system cannot be returned to service prior to the next surveillance time, steady-state plant operations at a core thermal output up to rated power may continue for a maximum of 24 hours after the last valid UFM correction factor was used in the calorimetric calculation for the daily nuclear power range surveillance. The 24-hour period is based on the minimum frequency for the calibration of the power range channels found in the FCS TSs. With the above clarification, the staff finds that the licensee has addressed the allowable outage time aspects of the CROSSFLOW system and complied with the guidance in Item G of Section I of Attachment 1 to RIS 2002-03.

With respect to the proposed actions to reduce power level if the allowed outage time is exceeded, the licensee proposed reducing the RTP and maintaining at the 1500 MWt level until the Crossflow UFM system is returned to service. The basis for reducing power to 1500 MWt RTP is the calorimetric uncertainty required by the Appendix K rule. The NRC staff finds that the licensee has addressed the proposed actions to reduce power during the CROSSFLOW system outage, and has complied with the guidance in Item H of Section I of Attachment 1 to RIS 2002-03.

Based on the above discussion, the NRC staff finds that the information identified in RIS 2002-03 and addressed each of the above items (I.1.A through I.1.H) in its application. The NRC staff finds this information acceptable.

On September 5, 2003, Westinghouse issued Technical Bulletin TB-03-6, "Crossflow Ultrasonic Flow Measurement System Signal Issues," to all CROSSFLOW users. TB-03-6 identified a potential for contamination of the signals used to determine feedwater flow rate. Potential errors in the correction factors, produced by the UFM, are used in the calorimetric calculation for plant power. The NRC staff has advised Westinghouse to verify the integrity of the information contained in previously approved TR (CENPD-397-P, Rev. 01-P) for generic applications of the CROSSFLOW UFM system and to establish guidelines instructing users of the UFM how to operate their system in a manner that will minimize the potential for signal contamination in the future.

In response to the NRC staff's RAI, the licensee provided the following status with respect to "Future Actions" as outlined in TB-03-6:

1. Westinghouse/AMAG (Advanced Measurement Analysis Group, Inc.) will complete the root cause analysis and communicate the detailed technical results to the

CROSSFLOW user community. A draft root cause analysis has been forwarded to OPPD. OPPD will close out this item when the formal root cause analysis is received.

2. Westinghouse/AMAG will update the Users Manual to include technical criteria for identifying potential contamination issues associated with plant hardware changes. Westinghouse informed OPPD that a Nuclear Safety Advisory Letter woud be sent to OPPD in December 2003.

3. Westinghouse/AMAG will evaluate the viability of procedural changes to formally obtain and document the frequency spectrum analysis as part of the quality-assured baseline plant data records. The FCS plant baseline data has already been obtained, analyzed, and found acceptable.

4. If baseline plant data records are currently unavailable, Westinghouse/AMAG will perform frequency spectrum analysis to establish these records for future use. The FCS plant baseline data has already been obtained, analyzed, and found acceptable.

5. Westinghouse/AMAG will evaluate the viability of modifying CROSSFLOW electronics and associated software with the goal of protecting against the effects of potential signal contamination. AMAG is developing new software to allow utilities to independently perform frequency spectrum analyses on demand.

The licensee stated that it will implement applicable Westinghouse/AMAG recommendations as identified above to ensure the operability of the FCS CROSSFLOW system is maintained. Resolution of these items will be tracked under the FCS corrective action program. Based on the above discussion, the NRC staff finds that there is reasonable assurance that the operability of the FCS CROSSFLOW system will be maintained.

In the past several years, the NRC staff has sought, in the course of our review of license amendments, documentation of plant instrument setpoint methodology. The requirements for instrument setpoint are derived from 10 CFR 50.36 which requires that the TSs establish and control limiting safety system settings (LSSS) for those variables that have significant safety functions. ISA Standard S67.04 was endorsed by RG 1.105, Revision 3. Part II of the standard, not endorsed by the NRC staff, includes three methods for calculating an allowable value (AV) as required by 10 CFR 50.36. Methods 1 and 2 calculate AVs that are sufficiently conservative and are acceptable to the NRC staff. Method 3, however, used by some licensees, does not provide an adequate margin to assure that the analytical limit (AL) is not violated. Method 3 subtracts the total loop uncertainty (TLU) value from the AL to derive the trip setpoint value, and then adds back the uncertainty associated with the instrument channel operational test/channel functional test (COT/CFT) to derive the AV. The TLU is the statistical combination of all uncertainties of a given instrument channel. The COT/CFT uncertainty is the statistical combination of all uncertainties associated with those instrument channel components that would be tested during the COT/CFT which may include instrument drift, instrument reference accuracy, and setting tolerance. This method is unacceptable because it does not account for all uncertainties not measured during COT/CFT. An acceptable method for deriving the AV will require an independent calculation that will assure that the margin between AV and AL would include all the uncertainties not measured during COT/CFT.

The NRC staff raised the above concern during the MUR power uprate review. In response to the NRC staff's concern, the licensee, by letter dated December 15, 2003, stated that the uncertainty calculations for reactor protection system (RPS) and engineered safety feature (ESF) setpoints at FCS were reconstituted in the early 1990s using the ISA 67.04 methodology. Since setpoints and associated procedural tolerances already existed prior to the reconstitution effort, uncertainty calculations were performed with the intent of demonstrating that total instrument loop uncertainty was bounded by the current setpoint values identified in plant calibration procedures. There was no attempt to make use of allowable values as defined in Method 3 of ISA 67.04. The setpoint tolerance utilized for quarterly functional checks is used as a component of the overall loop uncertainty and is accounted for in the TLU calculation. The concept of AV was intended as a means to remove some of the conservatism from the loop uncertainty calculation for periodic setpoint checks. This was not needed at FCS for RPS and ESF setpoints, because historical trends showed that there was very little drift in these setpoints.

The licensee has provided a document titled "Low Steam Generator Pressure Trip Setpoint Calculation" (FCS Calculation No. FC05722), as an example to illustrate that the current plant instrument setting meets the requirements for the intended low steam generator pressure trip function:

Analytical Limit (AL, from accident analysis)	478 psia
With Total Loop Uncertainty (18.0 psi) on top of AL	496.0 psia
Technical Specification Setting Limit:	500 psia
Plant Instrument Setting:	507.5 psia

There is always adequate margin between the AL and the worst case setpoint to account for the required components included in the TLU calculation. If a setpoint is found outside the TS setting limit, the loop is considered inoperable and action is taken to restore it to the proper range. TS setting limit as a minimum has a margin equivalent to total loop uncertainties from the AL. OPPD has not made use of AV in calibration procedures or functional check since it was not considered necessary, and has used TS setting limit for the operability determination. Because the current RPS and ESF setpoints do not incorporate AVs as defined by Method 3 of ISA 67.04, any change to these setpoints to incorporate Method 3 AVs would require prior NRC approval of changes to the applicable TSs.

Based on the above clarifications provided in the December 15, 2003, letter the staff considers that the FCS setpoint methodology issue is resolved.

3.1.3 Summary

The NRC staff has reviewed the licensee's proposed plant-specific implementation of the feedwater flow measurement device and the power uncertainty calculations. The NRC staff has determined that the licensee's proposed use of TR CENPD-397-P, Revision 01-P is consistent with the NRC staff's approval of the TR. The NRC staff has also determined that the licensee has adequately accounted for the uncertainties due to power level instrumentation error in its power level uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to instrumentation and controls.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

The NRC staff's review in the area of reactor systems covers the impact of the proposed MUR power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the reactor and reactor coolant system, and (5) LOCA and non-LOCA transient analyses (RIS 2002-03, Attachment 1, Sections II, III, and VI). The review is conducted to verify that the licensee's analyses bound plant operation at the proposed power level and that the results of the licensee's analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Guidance for the NRC staff's review of reactor systems is contained in SRP Chapters 4, 5, 6, and 15.

3.2.2 Technical Evaluation

The NRC staff reviewed the licensee's application related to reactor systems performance and determined that, with the exception of the spent fuel pool cooling (SFPC) system, the existing analyses of record bound operation of the plant at the proposed MUR power level. The results of the NRC staff's review of the effects of the proposed MUR on the SFPC system are discussed in Section 3.3.2.1.8 of this SE. The results of the NRC staff's review for the remaining areas discussed in Section 3.2.1 are summarized in Table 3.2.2 below.

Table 3.2.2
Reactor Systems - Summary of NRC Staff Review

Topic	Application Section and Page Number	USAR Section	Bounded by NRC-approved Analysis (Y/N and Reference)	NRC Staff Conclusion
Core Evaluation				
Fuel Design	IV.8, pg 60	3.8	Y [2, 3, and 4]	Evaluated at 101.7% power. Acceptable. See Note 1.
Fuel Structural Evaluation	IV.8, pg 60	3.7	Y [2, 3, and 4]	Evaluated at 101.7% power. Acceptable. See Note 1.
Nuclear Design	IV.8.1, pg 60	3.4	Y [2, 3, and 4]	Evaluated at 101.7% power. Acceptable. See Note 1.
Core Thermal-Hydraulic Design	IV.8.4, pg 62	3.6	Y [2, 3, and 4]	Evaluated at 101.7% power. Acceptable. See Note 1.
Accidents and Transients Analyses of Record				
Control Element Assembly (CEA) Withdrawal	II.1.1, pg 23	14.2	Y [1]	Analyzed at 102% power. Acceptable.

Table 3.2.2
Reactor Systems - Summary of NRC Staff Review

Topic	Application Section and Page Number	USAR Section	Bounded by NRC-approved Analysis (Y/N and Reference)	NRC Staff Conclusion
Boron Dilution	II.1.2, pg 23	14.3	Y [1]	Bounded by CEA Withdrawal. Acceptable.
CEA Drop	II.1.3, pg 23	14.4	Y [1]	Analyzed at 102% power. Acceptable.
Mal-Positioning of the Non-Trippable CEAs	II.1.4, pg 24	14.5	N/A [1]	Not permitted by TS. See Note 3.
Loss of Coolant Flow Event	II.1.5, pg 24	14.6.1 (Cy 21 update)	Y [1]	Analyzed at 102% power + pump heat. Acceptable.
Seized Rotor Event	II.1.6, pg 24	14.6.2	Y [1]	Analyzed at 102% power. Acceptable.
Idle Loop Startup	II.1.7, pg 24	14.7	N/A [1]	Not permitted by TS. See Note 3.
Turbine Generator Overspeed Incident	II.1.8, pg 25	14.8	N/A [1]	Not affected by power level. See Note 4.
Loss of Load to Both Steam Generators	II.1.9, pg 25	14.9.1	Y [1]	Analyzed at 102% power + pump heat. Acceptable.
Loss of Load to One Steam Generator	II.1.10, pg 25	14.9.2	Y [1]	Analyzed at 102% power + pump heat. Acceptable.
Loss of Feedwater Flow	II.1.11, pg 25	14.10.1	Y [1]	Analyzed at 102% power + pump heat. Acceptable.
Loss of Feedwater Heating	II.1.12, pg 25	14.10.2	Y [1]	Analyzed at 102% power + pump heat. Acceptable.
Excess Load	II.1.13, pg 26	14.11	Y [1]	Analyzed at 102% power. Acceptable.
Main Steam Line Break (MSLB) Accident	II.1.14, pg 26	14.12	Y [2 and3]	Acceptable. See Note 2.
CEA Ejection	II.1.15, pg 27	14.13	Y [1]	Analyzed at 102% power. Acceptable.
Steam Generator Tube Rupture Accident	II.1.16, pg 28	14.14	Y [1]	Analyzed at 102% power. Acceptable.
Large Break LOCA	II.1.17, pg 28	14.15.4	Y [1]	Analyzed at 102% power. Acceptable.
Small Break LOCA	II.1.18, pg 29	14.15.5	Y [1]	Analyzed at 102% power. Acceptable.
Long Term Core Cooling	II.1.19, pg 29	14.15.6	Y [1]	Analyzed at 102% power. Acceptable.

Table 3.2.2 Reactor Systems - Summary of NRC Staff Review				
Topic	**Application Section and Page Number**	**USAR Section**	**Bounded by NRC-approved Analysis (Y/N and Reference)**	**NRC Staff Conclusion**
Containment Pressure Analysis for MSLB	II.1.20, pg 29	14.16	Y [1]	Analyzed at 102% power. Acceptable.
Containment Pressure Analysis for LOCA	II.1.21, pg 30	14.16	Y [1]	Analyzed at 104% power. Acceptable.
Generation of Hydrogen in Containment	II.1.22, pg 30	14.17	N/A [2 and 3]	Related to Zr-water reaction, not power level
Fuel Handling Accident	II.1.23, pg 30	14.18	Y, USAR	Analyzed at 102% power for core inventory. Acceptable.
Gas Decay Tank Rupture	II.1.24, pg 30	14.19	Y, USAR	Analyzed at 102% power for core inventory. Acceptable.
Waste Liquid Incident	II.1.25, pg 30	14.20	Y, USAR	Analyzed at 102% power for core inventory. Acceptable.
Reactor Coolant System Depressurization	II.1.27, pg 31	14.22	Y [1]	Analyzed at 102% power + pump heat. Acceptable.
Control of Heavy Loads	II.1.29, pg 31	14.24	Y, USAR	Analyzed at 102% power for radiological consequences. Acceptable.
Control Room Habitability	II.1.28, pg 31	14.23	N/A [2]	Not affected by power level.
Feedwater Line Break Analysis	II.1.30, pg 31	Auxiliary Feedwater (AFW) system sizing studies	N [1] Not in USAR licensing basis.	Analyzed at 102% power + pump heat. Acceptable.
Anticipated Transients Without Scram (ATWS) (10 CFR 50.62)	II.1.31, pg 32	7.2.11	Y [2]	Analyzed at 1565 MWt. DSS satisfies ATWS Rule. Acceptable. See Note 6.
System Design				
Chemical and Volume Control System	VI.1.1, pg 70	9.2	Y [2 and 3]	Evaluated for operation at 101.67% power. Acceptable.
Shutdown Cooling System	VI.1.2, pg 72	9.3	Y [2 and 3]	Evaluated for shutdown from 101.67% power. Acceptable.
Safety Injection	VI.1.3, pg 74	6.2	Y [2 and 3]	Analyzed for events

Table 3.2.2 Reactor Systems - Summary of NRC Staff Review				
Topic	**Application Section and Page Number**	**USAR Section**	**Bounded by NRC-approved Analysis (Y/N and Reference)**	**NRC Staff Conclusion**
System				occurring at 102% power. Acceptable.
Containment Spray System	VI.1.4, pg 75	6.3	Y [2 and 3]	Analyzed for events occurring at 102% power. Acceptable.
Regulating Systems	VI.1.5, pg 75	7.4	Y [2 and 3]	Evaluated for operation at 101.67% power. Acceptable.
Engineered Safeguards Controls and Instrumentation System	VI.1.6, pg 77	7.3	Y [2 and 3]	Actuations occur during events analyzed at 102% power. Acceptable.
Instrumentation Systems	VI.1.7, pg 77	7.5	Y [2 and 3]	Evaluated for operation at more than 101.67% power. Acceptable.
Refueling Systems	VI.1.8, pg 78	9.5	Y [2 and 3]	Refueling systems will accommodate fuel for storage and operation at uprated level. Acceptable. See Note 5.
Containment Systems	VI.1.9, pg 79	6.3 and 6.4	Y [2 and 3]	No changes are planned due to MUR. MSLB and LOCA mass and energy releases were analyzed at 102% power. Acceptable.
Other				
Low Temperature Overpressure Protection System	VI.5, pg 95	4.3.9	Y [5,6,7,8]	Evaluated for decay heat present after operation at 102% power. Acceptable. See Note 8.
Reactor Vessel Fluence Assessment	IV.1.1, pg 42	3.4.6	Y [5,6,7,8]	MUR uprated reactor will reach 40 effective full power years (EFPY) fluence in 39.9 EFPY. Pressure-temperature (P-T) limits will be modified as required. Acceptable. See Note 7.

References for Table 3.2.2

1. EA-FC-02-016, "Cycle 21 Transients Summary," June 19, 2002.

2. Letter LIC-03-0067, from W.G. Gates, Vice President, OPPD to USNRC, "Fort Calhoun Station Unit 1, License Amendment Request (LAR), Measurement Uncertainty Recapture Power Uprate," Docket No. 50-285, July 18, 2003, ADAMS Accession No. ML032030066.

3. Letter LIC-03-0122, from S.K. Gambhir, Division Manager, Nuclear Projects, OPPD to USNRC, "Fort Calhoun Station Unit 1, License Amendment Request (LAR), Measurement Uncertainty Recapture Power Uprate," Docket No. 50-285, August 28, 2003, ADAMS Accession No. ML032450518.

4. EMF-1961(P)(A) Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000 (not publicly available).

5. WCAP-15443, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel" by S. Anderson, Westinghouse Electric Company LLC, July 2000.

6. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," US Nuclear Regulatory Commission, March 2001.

7. Letter LIC-02-0109, from D. J. Bannister, Manager, Fort Calhoun Station, OPPD to USNRC, "Fort Calhoun Station Unit 1, License Amendment Request (LAR), Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," Docket No. 50-285, October 8, 2002, ADAMS Accession No. ML022950374.

8. "Fort Calhoun Station, Unit 1 - Issuance of Amendment (TAC No. MB6468)," Docket No. 50-285, August 15, 2003, Amendment 221, Approving the Use of RCS Pressure and Temperature Limits Report (PTLR), ADAMS Accession No. ML032300305.

<u>Notes for Table 3.2.2</u>

Note 1 – Fuel and Core Thermal-Hydraulic Evaluation

The MUR power uprate to 101.6 percent is covered by the 102 percent power assumed for the current transient analyses (with the exception of the MSLB event – see Note 2). The core peaking factors are still within previously established limits.

The following table lists the reactor information that was used for the mechanical design evaluations as compared to current parameter values. Core thermal hydraulic (T/H) analyses and evaluations were performed at a 1.7 percent uprated core power level (1526 MWt) which bounds the proposed 1.6 percent uprate. The departure from nucleate boiling (DNBR) design limits and safety limits were the same as the values used in the current design basis analyses.

Parameter	**Current Value**	**MUR Power Uprate Value**
Core Thermal Power, MWt	1500	1524

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	596.0	596.8
Maximum Overpower,%	112	112
Fraction of Heat from Fuel Rods	0.975	0.975
Core Average Linear Heat Rate (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

Note 2 – Main Steam Line Break Accident

The hot full power case analyses of the MSLB are analyzed at 100 percent power (1500 MWth). The licensee claims that the initial power level used in such analyses has an insignificant effect upon the post scram return to power. The NRC staff agrees.

The MSLB is analyzed assuming that the most reactive CEA is stuck outside the core at the time of scram. As the cooldown-induced post scram reactivity excursion overcomes the shutdown margin, and the core generates power, very high hot channel factors are produced in the vicinity of the stuck CEA, which could result in departure from nucleate boiling (DNB) and fuel clad damage. The MSLB analyses focus upon the possibility of fuel clad damage occurring at some time after the scram. Thus, the MSLB analysis scenario can be construed to begin only after the reactor scram is executed, which makes the assumed initial power level relatively insignificant.

Note 3 – Mal-Positioning of the Non-Trippable CEAs and Idle Loop Startup

These events are not evaluated, since the TS do not permit operation in plant configurations that make the occurrence of these events possible.

Note 4 – Turbine Generator Overspeed Incident, Generation of Hydrogen in Containment, and Control Room Habitability

These events are not evaluated, since the rated power level would have no effect upon the analysis results.

Note 5 – Refueling Systems

Refueling systems are evaluated assuming the storage requirements for operation at the proposed uprated power level.

Note 6 – Anticipated Transients without Scram

ATWS is evaluated at a power level that is greater than the Appendix K requirement (102 percent). However, the ATWS rule (10 CFR 50.62) does not require an analysis. Instead, it requires the installation of an alternate diverse shutdown system (DSS). ATWS is not affected by the proposed uprating, since there are no plant-specific, power-dependent analysis results in the record. Rather, the requirements for ATWS are met by the DSS. Operation of the DSS, if needed, would essentially transform ATWS events into anticipated transients, which are that anticipated operational occurrences that have been evaluated, in the applicant's submittals, under the conditions of the proposed uprating.

Note 7 – Evaluation of Reactor Vessel Fluence at the Uprated Power Level

FCS is using approved P-T limit curves for 40 EFPYs of operation and an approved pressure temperature limits report (PTLR) (References 7 and 8). The pressure vessel critical element for the calculation of the P-T curves is longitudinal weld 3-410 located at an azimuthal angle of 60°.

WCAP-15443 (Reference 5) includes the FCS fluence calculations to 48 EFPYs. The calculations reported in Reference 5 are acceptable because they follow the guidance in the Draft Guide 1035, which is essentially the same as RG 1.190 (Reference 6). The fluence value for 40 EFPYs at 60° is 2.15×10^{19} n/cm^2. The MUR fluence increase is estimated to be numerically equal to the power uprate. This is a reasonable assumption provided that the fuel loading mode is preserved. The licensee estimated that the MUR uprated reactor will reach the 40 EFPY fluence in 39.9 EFPYs. As noted above, the licensee has an approved PTLR and stated that the P-T limits will be modified as required.

In summary, the licensee used an acceptable value of the fluence to modify the time of the next revision of the P-T curves using an approved methodology, and the NRC staff finds it acceptable.

Section 3.6.2 of this SE further discusses the reactor vessel fluence and P-T limits.

Note 8 – Low Temperature Overpressure Protection (LTOP) System

The LTOP system was evaluated assuming the decay heat present at the time the LTOP system might be required to function, after operation at a power level of 102 percent. The LTOP system, as evaluated for the 102 percent power/decay heat conditions, is found to be adequate for the proposed MUR power uprating. No setpoint changes are required.

The methodology used for this evaluation (Reference 7) was approved by the staff in Amendment 221 dated August 15, 2003.

Section 3.6.2 of this SE further discusses the P-T limits.

3.2.2.1 Impact of Power Uprate on Non-Bounding Reactor Systems Analyses

3.2.2.1.1 Spent Fuel Pool Cooling System

The licensee has identified the SFPC system as a non-bounded system analysis. The staff's review of the SFPC system analysis is discussed in Section 3.3.2.1.8 of this SE.

3.2.2.2 Nuclear Steam Supply System (NSSS) Design Transients

The design transients establish pressure and temperature criteria for the design specifications of plant components. The design transients and their associated frequencies, provided in the component specifications, are used to determine the thermal fatigue usage factors. Thermal fatigue is dependent upon temperature and pressure changes on the component.

Design transients considered include:

Heatup, 100°F/hr
Cooldown, 100°F/hr
Loading, 10 percent/min
Unloading, 10 percent/min
Step load increase, 10 percent
Step load decrease, 10 percent
Reactor trip
Hydrostatic test
Leak test
Starting and stopping reactor coolant pumps (RCPs)
Secondary side hydrostatic test
Secondary side leak test
Cold feedwater following hot standby

Many of these transients, such as the heatup and cooldown, and the starting and stopping of the RCPs, are not affected by the plant power level. Transients that are postulated to occur at subcritical or no-load conditions would also be unaffected by the power uprate, since there are no proposed changes to the no load conditions.

NSSS design transients that are dependent upon reactor power were re-analyzed assuming a reactor power level greater than 101.6 percent. The increase in power level did not have a significant effect upon the results, since the original NSSS design transient analyses had been based upon conservatively high full load reactor coolant system temperatures. The MUR power uprate results in a small increase in decay heat generation. This small amount of additional heat is removed by the turbine bypass system.

The NRC staff agrees that the NSSS design transient analyses cover the MUR uprated reactor power level.

3.2.2.3 System Design

3.2.2.3.1 NSSS Interface Systems

The NSSS interface systems, presented in Section VI.1 of the licensee’s submittal, consist of:

Chemical and volume control system
Shutdown cooling system
Safety injection system
Containment spray system
Regulating systems
Engineered safeguards controls and instrumentation systems
Instrumentation systems
Refueling systems
Containment systems

These systems have been evaluated for operation at power levels exceeding the proposed MUR uprated power level. The NRC staff agrees that these systems’ performance bounds the requirements of the FCS plant at the MUR uprated power level. The results are summarized in Table 3.2.2.

3.2.2.3.2 NSSS Control Systems

Section VI.5 of the licensee's submittal addresses NSSS control systems by discussing the plant safety limits and the effects the MUR uprate may have upon them. The licensee has determined that the safety system setpoints and plant safety limits continue to provide adequate protection at the MUR uprated power level without adjustment. Accordingly, the licensee has not requested any changes to protection system settings in the TSs.

Section VI.5 also contains an evaluation of the LTOP system. The MUR uprated power level affects the LTOP by producing a small increase in the decay heat that would be present when an LTOP event would occur. Since the LTOP system has been evaluated for a 102 percent power level, the system would continue to provide adequate protection at the MUR uprated power level. The staff has recently approved the use of a new PTLR for FCS in Amendment 221.

The NRC staff agrees that no changes to any protection system or emergency system settings are required by the proposed MUR power uprating. In addition, Section 3.1 has some additional comments on the current setpoint methodology.

3.2.3 Summary

The NRC staff has reviewed the licensee’s assessment of the impact of the proposed MUR power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the NSSS, and (5) LOCA and non-LOCA transient analyses. Based on the above, the NRC staff has determined that the results of the licensee’s analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Where

additional assessments and analyses were necessary, the NRC staff has reviewed these assessments and analyses and finds that the licensee has satisfactorily addressed the areas discussed above, the input parameters of the analyses adequately represent the plant conditions at the proposed uprated power level, and the analytical results meet the applicable acceptance criteria. Based on the above, the NRC staff finds the proposed MUR power uprate acceptable with respect to reactor systems' performance.

3.3 Plant Systems

3.3.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) SFPC analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, and (7) ESF heating, ventilation, and air conditioning (HVAC) systems (RIS 2002-03, Attachment 1, Sections II, III, and VI). The review is conducted to verify that the licensee's analyses bound plant operation at the proposed MUR power level and that the results of the licensee's analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Guidance for the NRC staff's review of reactor systems is contained in SRP Chapters 3, 6, 9, 10, and 11.

3.3.2 Technical Evaluation

The NRC staff reviewed the licensee's application related to plant systems' performance and determined that the existing analyses of record bound operation of the plant at the proposed MUR power level with the exception of the following systems:

- Fire protection system
- Main steam and steam dump system
- Condensate and feedwater systems
- Feedwater heater drains
- Steam generator blowdown
- Turbine auxiliary cooling water system
- Circulating water system
- SFPC system
- Auxiliary building heating and ventilation systems

The results of the NRC staff's review of the effects of the proposed MUR on the above systems are discussed in Section 3.3.2.1 of this SE. The results of the NRC staff's review for the remaining areas discussed in Section 3.3.1 are summarized in Table 3.3.2 below.

Table 3.3.2 Plant Systems - Summary of NRC Staff Review					
Topic	**Application Section and Page Number**	**USAR Section**	**Bounded by Current Licensing Basis[NOTE 1] (Y/N and Reference)**	**Bounded by Revised Analyses**	**NRC Staff Conclusion**
Post-LOCA Containment Hydrogen Generation	II.3.5, pg 35	14.17	Y, USAR 14.17		Acceptable
Long-Term LOCA Mass and Energy Release Analysis	II.3.3, pg 35	14.16	Y, Reference 1		Acceptable
Short-Term LOCA Mass and Energy Release Analyses	II.3.2, pg 35	14.16	Y, Reference 2		Acceptable
Fire Protection Systems					
Fire Protection Evaluation	III.2.1, pg 41	9.11	Y, References 3 through 14, see page 41 of submittal and Reference 15	Y, SE Section 3.3.2.1.1	Acceptable
Power/Steam Systems					
Main Steam System and Steam Dump System	VI.2.1, pg 80	10.2.1		Y, SE Section 3.3.2.1.2	Acceptable
Condensate and Feedwater Systems	VI.2.2, pg 83	10.2.2		Y, SE Section 3.3.2.1.3	Acceptable
Auxiliary Feedwater System and Condensate Storage System	VI.2.3, pg 86	9.4	Y, USAR 9.4		Acceptable
Feedwater Heater Drains	VI.2.4, pg 87	10.2.2		Y, SE Section 3.3.2.1.4	Acceptable
Steam Generator Blowdown System	VI.2.7, pg 89	11.1		Y, SE Section 3.3.2.1.5	Acceptable
Cooling and Support Systems					
Component Cooling Water System See Note 2	VI.3.1, pg 89	9.7	Y, USAR 9.7		Acceptable
Raw Water Cooling System	VI.3.4, pg 92	9.7	Y, USAR 9.7		Acceptable
Turbine Auxiliary Cooling Water System	VI.3.2, pg 90	9.9		Y, SE Section3.3.2.1.6	Acceptable
Emergency Diesel Generator	V.8, pg 67	8.4	Y, USAR 8.4		Acceptable
Circulating Water System	VI.3.3, pg 91	10.2		Y, SE Section 3.3.2.1.7	Acceptable

Table 3.3.2
Plant Systems - Summary of NRC Staff Review

Topic	Application Section and Page Number	USAR Section	Bounded by Current Licensing Basis[NOTE 1] (Y/N and Reference)	Bounded by Revised Analyses	NRC Staff Conclusion
Spent Fuel Pool Cooling System	III.1.1, pg 40	9.6		Y, see pages 40-41 of submittal and Reference 16. See Section 3.3.2.1.8	Acceptable
Heating, Ventilation, and Air Conditioning Systems					
Auxiliary Building Ventilation Systems	VI.4, pg 93	9.1	Y, USAR 9.10		Acceptable
Containment Air Cooling and Filtration System	VI.3.5, pg 93	6.4, 9.10	Y, USAR 6.4 and 9.10		Acceptable
Auxiliary Feedwater Pump Room Coolers	VI.4, pg 94	9.10	Y, USAR 9.10		Acceptable
Control Room Ventilation System	VII.1, pg 96	9.10	Y, USAR 9.10, 14.23		Acceptable

References for Table 3.3.2

1. NRC to Combustion Engineering, "NRC Approval of CENPD-140-A, 'Description of CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis'," April 1974.

2. NUREG-75/112, Safety Evaluation Report related to Preliminary Design of the CESSAR, December 1975.

3. Letter to T. E. Short, OPPD, from George Lear, "Amendment No. 38 to Facility Operating License No. DPR-40," February 14, 1978.

4. Letter to T. E. Short, OPPD, from Robert W. Reid, "Amendment No. 40 to Facility Operating License No. DPR-40," August 23, 1978.

5. Letter to W. C. Jones, OPPD, from Robert A. Clark, "Amendment No. 53 to Facility Operating License No. DPR 40," November 17, 1980.

6. Letter to W. C. Jones, OPPD, from Thomas M. Novak, "Safety Evaluation Report on 10 CFR Part 50, Appendix R, Items III.G and III.L," April 8, 1982.

7. Letter to W. C. Jones, OPPD, from Robert A. Clark, "FCS Design Meets 10 CFR 50, Appendix R, Items III.G and III.L with respect to Safe Shutdown in the Event of a Fire in the Control Room or Cable Spreading Room," August 12, 1982.

8. Letter to R. L. Andrews, OPPD, from Edward J. Butcher, "Approval of Exemptions from 10 CFR Part 50, Appendix R," July 3, 1985.

9. Letter to R. L. Andrews, OPPD, from Edward J. Butcher, "Safety Evaluation on Alternate Shutdown Capability for the Upper Electrical Penetration Room," November 5, 1985.

10. Letter to R. L. Andrews, OPPD, from Donald E. Sells, "Approval of Deviations from Requirements and Commitments of July 3, 1985 Safety Evaluation Report," July 1, 1986.

11. Letter to W. Gary Gates, OPPD, from Dennis M. Crutchfield, "Denial of Exemption Request for Fire Area 34B," November 14, 1990.

12. Letter to Terry L. Patterson, OPPD, from Steven Bloom, "Clarification of July 1, 1986 Safety Evaluation," March 17, 1993.

13. Letter to Terry L. Patterson, OPPD, from Steven Bloom, "Amendment No. 160 to Facility Operating License No. DPR-40," January 14, 1994.

14. Letter to S. K. Gambhir, OPPD, from L. Raynard Wharton, "Issuance of Exemption from 10 CFR Part 50, Appendix R, Section III.0," May 21, 1998.

15. Calculation FC-06669, "Heat Removal Success Paths to Maintain RCS Temperature Below 300°F for the Fort Calhoun Station," Revision 0, August 22, 1997.

16. Calculation FC-5988, "Thermal-Hydraulic Analysis of Fort Calhoun Station Spent Fuel Pool with Maximum Density Storage," Revision 2, February 6, 2003.

<u>Notes for Table 3.3.2</u>

1. One of the generic assumptions used in the deterministic analysis accounts for steady state operational and instrumentation errors (measurement uncertainties). This assumption, which is applicable to all design basis accident analyses, includes a 2 percent error (30 Mwth) for calorimetric error. The design parameters for safety systems are the bounding parameters for the safety systems and, because the design basis accident analyses are performed at 102 percent thermal power, the MUR thermal power uprate loads are bounded for these systems.

2. The component cooling water (CCW) system is designed to support three operating modes: normal, shutdown and emergency. The emergency heat load bounds the other two. Since the design basis accident (DBA) loads were performed at 102 percent thermal power, it bounds the DBA results for the MUR 1.6 percent increase.

3.3.2.1 Impact of Power Uprate on Non-Bounding Plant Systems Analyses

The licensee has reviewed the following balance-of-plant (BOP) systems for the effects of the MUR on the piping systems including the valves and instrumentation. This analyses included recalculation of temperature, pressure, flows, heat loads, power requirements and setpoints needed to ensure the MUR conditions are bounded by the existing system analyses.

3.3.2.1.1 Fire Protection System

The licensee reanalyzed the Appendix R cold shutdown to determine the effect of the increase in reactor power. The reanalysis has demonstrated that the plant can reach cold shutdown conditions within 72 hours for an Appendix R shutdown. The NRC staff agrees that the safe shutdown fire analyses and required systems are acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.2 Main Steam and Steam Dump System

The licensee reanalyzed the main steam and steam dump system to determine the effects of the MUR. The licensee has concluded that the existing piping analysis bounds the operating temperature and pressure and steam flow conditions resulting from the MUR. The NRC staff agrees that the main steam and steam dump system are acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.3 Condensate and Feedwater Systems

The licensee reanalyzed the condensate and feedwater systems to determine the effects of the MUR. The licensee has concluded that the existing piping analysis bounds the operating temperature and pressure and feedwater flow conditions resulting from the MUR. The NRC staff agrees that the condensate and feedwater systems are acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.4 Feedwater Heater Drains System

The licensee reanalyzed the feedwater heaters drains system to determine the effects of the MUR. The licensee has concluded that the existing piping analysis bounds the operating temperature and pressure and flow conditions resulting from the MUR. The NRC staff agrees that the feedwater heater drains system is acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.5 Steam Generator Blowdown

The licensee reanalyzed the steam generator blowdown system to determine the effects of the MUR. The licensee has concluded that the existing piping analysis bounds the operating temperature and pressure and flow conditions resulting from the MUR. The NRC staff agrees that the steam generator blowdown system is acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.6 Turbine Auxiliary Cooling Water System

The licensee reanalyzed the turbine auxiliary cooling water system to determine the effects of the MUR. The licensee has concluded that the existing piping analysis bounds the operating temperature and pressure and flow conditions resulting from the MUR. The NRC staff agrees that the turbine auxiliary cooling water system is acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.7 Circulating Water System

The licensee reanalyzed the circulating water system to determine the effects of the MUR. The licensee has concluded that the existing piping analysis bounds the operating temperature and pressure and flow conditions resulting from the MUR. The NRC staff agrees that the circulating water system is acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.8 SFPC System

The licensee reanalyzed the SFPC system to determine the effects of the MUR. The licensee did a linear extrapolation of the decay heat load for this reanalysis. This resulted in a decrease in the time to boil from 7.2 hours to 6.5 hours. This change is not significant. The NRC staff agrees that the spent fuel pool system does provide sufficient time for the licensee to prevent boiling in the spent fuel pool, and therefore this system is acceptable for the 1.6 percent MUR power uprate.

3.3.2.1.9 Auxiliary Building Heating and Ventilation Systems

The licensee reanalyzed auxiliary building heating and ventilation systems to determine the effects of the MUR. The licensee has concluded that there would be no change to the cooling load requirements as a result of the MUR. The safety injection pumps located in the auxiliary building were designed for 102 percent power and these heat loads were already accounted for. The NRC staff agrees that the auxiliary building heating and ventilation systems are acceptable for the 1.6 percent MUR power uprate.

3.3.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, and (7) ESF heating, ventilation, and air conditioning systems. The NRC staff has determined that the results of licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Where additional assessments and analyses were necessary, the NRC staff has reviewed these assessments and analyses and finds that the licensee has satisfactorily addressed the areas discussed above; the input parameters of the analyses adequately represent the plant conditions at the proposed uprated power level, and the analytical results will continue to meet the applicable requirements. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to plant systems.

3.4 Electrical Systems

3.4.1 Regulatory Evaluation

The NRC staff's review in the area of electrical engineering covers the impact of the proposed MUR power uprate on (1) grid stability, including performance of the main generator, main transformer, isophase bus, and unit auxiliary transformer/reserve auxiliary transformer, (2) emergency diesel generator loading, (3) station blackout (SBO), and (4) environmental qualification of electrical equipment (RIS 2002-03, Attachment 1, Section V). This review is conducted to verify that the results of the licensee's analyses related to these areas continue to meet the requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, 10 CFR 50.63, and 10 CFR 50.49 following implementation of the proposed MUR power uprate.

3.4.2 Technical Evaluation

The NRC staff reviewed the licensee's application related to electrical system performance. The results of the NRC staff's review of the effects of the proposed MUR on the plant are discussed in Section 3.4.2.1 of this SE. The results of the NRC staff's review are summarized in Table 3.4.2 below.

Table 3.4.2
Electrical Systems - Summary of NRC Staff Review

Topic	Application Section and Page Number	USAR Section	Bounded by NRC-approved Analysis (Y/N and Reference)	NRC Staff Conclusion
Grid Stability	V.7, pg 67	N/A	SE Section 3.4.2.1.2.1	Acceptable
Main Generator	V.1, pg 64	10.2	SE Section 3.4.2.1.2.2	Acceptable
Main Transformer	V.2, pg 65	8	SE Section 3.4.2.1.2.3	Acceptable
Isolated Phase Bus	V.3, pg 65	8	SE Section 3.4.2.1.2.4	Acceptable
Unit Auxiliary Transformer / House Service Transformer	V.9, pg 68	8.3	SE Section 3.4.2.1.2.5	Acceptable
4160/480 Volts Distribution System	V.4, pg 65	8	SE Section 3.4.2.1.2.6	Acceptable
Motor Loads and Power Cables	V.5, pg 66	8	SE Section 3.4.2.1.2.7	Acceptable
DC Distribution System	V.6, pg 66	8	SE Section 3.4.2.2	Acceptable
Emergency Diesel Generators	V.8, pg 67	8.4	SE Section 3.4.2.3	Acceptable
Station Blackout	V.11, pg 69	n/a	SE Section 3.4.2.4 References 2 and 3	Acceptable
Environmental Qualification of Electrical Equipment	V11.6.1, pg 99	n/a	SE Section 3.4.1.1 Reference 3	Acceptable

References for Table 3.4.2

1. Letter to W. C. Jones, OPPD, from Robert A. Clark, NRC, "Safety Evaluation for Environmental Qualification of Safety-Related Electrical Equipment," May 29, 1981.

2. Letter to W. Gary Gates, OPPD, from David L. Wigginton, "Safety Evaluation Report and Technical Evaluation Report for Implementation of the Station Blackout Rule, 10 CFR 50.63," May 1, 1991.

3. Letter to W. Gary Gates, OPPD, from David L. Wigginton, NRC, "NRC Supplemental Safety Evaluation for Implementation of the Station Blackout Rule, Fort Calhoun Station, Unit 1," April 13, 1992.

3.4.2.1 Impact of Power Uprate on Non-Bounding Electrical Analyses

3.4.2.1.1 Environmental Qualification (EQ) of Electrical Equipment

Regulatory Evaluation

The term "environmental qualification" applies to equipment important-to-safety to assure this equipment remains functional during and following design basis events. The staff's review covers the environmental conditions that could affect the design and safety functions of electrical equipment including instrumentation and control. The staff's review verified compliance with the acceptance criteria thus ensuring that the equipment continues to be capable of performing its design safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions. Acceptance criteria are based on 10 CFR 50.49 as it relates to specific requirements regarding the qualification of electrical equipment important-to-safety that is located in a harsh environment. Specific review criteria are contained in SRP Chapter 3.11.

Technical Evaluation

The licensee performed the following analyses for the pressure, temperature, humidity/spray and radiation dose on the electrical equipment using a reactor thermal power greater than the thermal power of the 1.6 percent MUR power uprate:

- LOCA,
- MSLB accident, and
- High energy line break (HELB) accident

The current pressure, temperature, humidity/spray and radiation dose profiles bound the environmental conditions expected at the 1.6 percent MUR power uprate conditions. The licensee reviewed and verified that the sub-compartment mass and energy releases remain bounding for the 1.6 percent 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 1.6 percent 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 environmental qualification of electrical equipment program.

Conclusion

The staff has reviewed the licensee's submittal of the effects of the proposed power uprate on the environmental qualification of the electrical equipment and concludes that the information provided demonstrates compliance with 10 CFR 50.49 and, therefore, is acceptable.

3.4.2.1.2 Offsite Power System

Regulatory Evaluation

Prior to the introduction of GDC 17 of Appendix A to 10 CFR Part 50, Criterion 39 of the Atomic Energy Commission's interim acceptance criteria was used to evaluate the adequacy of the electric power systems. Criterion 39 requires that sufficient offsite and redundant, independent, and testable standby auxiliary sources of electrical power are provided to attain a prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources is adequate to accomplish all required engineered safety feature functions under all postulated design basis accident conditions. Acceptance criteria are based on Criterion 39.

Technical Evaluations Related to Offsite Power System

3.4.2.1.2.1 Grid Stability

The generator output is fed through 648 mega-volt amperes (MVA), 22-kV/345-kV main power transformer to a bay in the 345-kV substation (substation 3451) located in the switchyard. The substation is directly connected to the 345 kV transmission network via three lines. In addition, the 345-kV system is connected to the 161 kV system through two 345 kV/161 kV, 500 MVA auto-transformers in the switchyard. The 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 161 kV/4.16 kV transformers. The second offsite source is brought in from the 345 kV system by opening the motor-operated main generator disconnect switch and backfeeding the plant through the main power transformer and the unit auxiliary transformers.

A review of the results of the current basis for grid stability performance indicates that the power uprate is not expected to have any adverse impact on the stability or any of the other neighboring generating units in the network. This conclusion is based on the substantial stability margin that the plant and neighboring generators have as reflected in applicable stability simulation studies. It is also based on the fact that the power uprate reflects an increase of only 1.6 percent in electric power output which is too small to have a perceptible impact on the system stability characteristics.

The NRC staff reviewed the licensee's submittal and concluded that the impact of the power uprate on the grid stability is insignificant. Therefore, the plant continues to meet the requirements of Criterion 39 for grid stability with this power uprate.

3.4.2.1.2.2 Main Generator

The 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 0.85. 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.

The NRC staff reviewed the main generator capability curve and concluded that the generator will continue to operate at the anticipated power uprate and, therefore, the design is acceptable.

3.4.2.1.2.3 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 MVA capability of the main generator remains at 590.8 MVA which is within the rating of the main transformer.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 1.6 percent is below the maximum main transformer design rating of each unit and, therefore, operating the main power transformers at the uprated power condition is acceptable.

3.4.2.1.2.4 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. At a power factor of 0.85, the 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.

The NRC staff reviewed the licensee's submittal and concluded that the impact of power uprate of 1.6 percent is below the design rating of the isophase bus of each unit and, therefore, operating the isophase bus at the uprated power condition is acceptable.

3.4.2.1.2.5 Unit Auxiliary and House Service Transformers (UAT and HST)

The UAT and 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 77 kVA, if the plant was in one of these conditions. After the MUR power uprate, this maximum load would be increased to 14.7 MW, which is below the rating of either transformer.

The NRC staff reviewed the licensee's submittal and concluded that the UAT and HST loading resulting from the 1.6 percent power uprate is below their maximum design rating and, therefore, operating these transformers at the uprated power condition is acceptable

3.4.2.1.2.6 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 1.6 percent 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 following the MUR power uprate. The degraded voltage protection analysis concludes that the 4160/480 distribution system has adequate margin to support the incremental needs during the 1.6 percent MUR power uprate conditions.

3.4.2.1.2.7 Motor Loads and Power Cables

The auxiliary power system consists of 4160V, 480V, vital 120V, 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 the MUR power uprate.

Conclusion

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the offsite power system and concludes that the offsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed power uprate. The NRC staff further concludes that the impact of the proposed power uprate on grid stability is insignificant. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the offsite power system.

3.4.2.2 Direct Current (DC) Distribution System

Regulatory Evaluation

The DC power systems include those DC power sources and their distribution systems and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the DC onsite power system. Acceptance criteria are based on Criterion 39 and 10 CFR 50.63 as they relate to the capability of the DC onsite electrical power to facilitate the functioning of structures, systems, and components important to safety. Specific review criteria are contained in SRP Chapters 8.1 and 8.3.2.

Technical Evaluation

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 eight hours upon loss of all alternating current (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 1.6 percent MUR power uprate does not affect the DC system.

Conclusion

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the DC onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. The staff further concludes that the DC onsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the DC onsite power system.

3.4.2.3 Emergency Diesel Generators

Regulatory Evaluation

The AC onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the AC onsite power system. Acceptance criteria are based on Criterion 39 as it relates to the capability of the AC onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Chapters 8.1 and 8.3.1.

Technical Evaluation

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 review of the ESF loads concluded that they have been conservatively determined for the most restrictive design basis accident (LOCA) from 102 percent power. These loads are not affected by the 1.6 percent 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.

Conclusion

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the AC onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. The staff further concludes that the AC onsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with respect to the onsite AC power system.

3.4.2.4 Station Blackout

Regulatory Evaluation

SBO refers to the complete loss of AC electric power to the essential and non-essential switchgear buses in a nuclear power plant. SBO involves the loss-of-offsite power concurrent with turbine trip and failure of the onsite emergency AC power system. SBO does not include the loss of available AC power to buses fed by station batteries through inverters or the loss of power from "alternate AC sources" (AAC). The NRC staff's review focuses on the impact of the proposed power uprate on the plant's ability to cope with and recover from an SBO event since SBO is based on 10 CFR 50.63. Specific review criteria are contained in SRP Chapter 8.1 and Appendix B to SRP Chapter 8.2.

Technical Evaluation

An SBO is defined as the complete loss of AC electric power to the essential and non-essential switchgear buses. The plant meets all the SBO rule requirements and is capable of coping for four hours under SBO conditions. The analysis (References 2 and 3 of Table 3.4.2) concludes that:

- The coping duration of four 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 SBO equipment.
- Containment integrity is ensured by the containment isolation valves.

The licensee's review determined the impact of the additional 1.6 percent 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 an SBO event. The coping duration of four hours continues to be satisfied, the diesel generator reliability meets the required guidelines, the DC batteries capacity is sufficient to handle an SBO event, containment integrity is assured by the containment isolation valves, and the loss of HVAC will not affect the operability of SBO equipment. Additionally, a review was performed to determine if the containment pressure and temperature for SBO conditions were more severe than that for an MSLB or a LOCA. Because the pressure at these SBO conditions was only slightly less than half of the value for the MSLB and LOCA, 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.

Conclusion

The staff has reviewed the licensee's submittal on the effect of the proposed power uprate on the plant's ability to cope with and recover from an SBO event for the period of time established on the plant's licensing basis. The plant has adequate condensate inventory for decay heat removal during an SBO of four hours duration. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed power uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following the implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with respect to an SBO.

3.4.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on (1) grid stability, including performance of the main generator, main transformer, isophase bus, and unit auxiliary transformer/reserve auxiliary transformer, (2) emergency diesel generators, (3) SBO, and (4) environmental qualification of electrical equipment. The NRC staff has determined that the results of the licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Where additional assessments and analyses were necessary, the NRC staff has reviewed these assessments and analyses and finds that the licensee has satisfactorily addressed the areas discussed above, the input parameters of the analyses adequately represent the plant conditions at the proposed uprated power level, and the analytical results will continue to meet the requirements of 10 CFR Part 50, Appendix A, GDC 17, 10 CFR 50.63, and 10 CFR 50.49. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to electrical engineering.

3.5 Mechanical and Civil Engineering

3.5.1 Regulatory Evaluation

The NRC staff's review in the area of mechanical and civil engineering covers the structural and pressure boundary integrity of NSSS and BOP systems and components (RIS 2002-03, Attachment 1, Section IV, Items 1.A, 1.B, and 1.D). The NRC staff's review focuses on the impact of the proposed MUR power uprate on (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) reactor vessel and supports; (4) control rod drive mechanism; (5) SGs and supports; (6) RCPs and supports; (7) pressurizer and supports; (8)

reactor internals and core supports; and (9) safety-related valves. Technical areas covered by this review include stresses, cumulative usage factors, flow-induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs. The review is conducted to confirm that (1) the results of the analyses continue to meet allowable limits as defined in the American Society of Mechanical Engineers (ASME) code of record for the plant, (2) the safety-related valves will continue to perform acceptably, and (3) the safety-related valve programs will continue to be adequate. Guidance for the NRC staff's review of the topics within the mechanical and civil engineering area are contained in SRP Chapters 3 and 5.

3.5.2 Technical Evaluation

The NRC staff reviewed the licensee's application related to the mechanical and civil engineering areas discussed in Section 3.5.1. The results of the NRC staff's review of the effects of the proposed MUR are discussed in Section 3.5.2.1 of this SE. The results of the NRC staff's review for the remaining bounded areas discussed in Section 3.5.1 above are summarized in Table 3.5.2 below.

Table 3.5.2
Mechanical and Civil Engineering - Summary of NRC Staff Review

Topic	Application Section and Page Number	USAR Section	Bounded by Current Licensing Basis (Y/N and Reference)	Bounded by Revised Analyses	NRC Staff Conclusion
Reactor Vessel Structural Evaluation	IV.1, pg 42	4.3.3		Y, SE Section 3.5.2.1.1	Acceptable
Reactor Internals	IV.1.2, pg 44	3.7.1		Y, SE Section 3.5.2.1.2	Acceptable
Piping and Supports	IV.2, pg 47	4.3.6		Y, SE Section 3.5.2.1.7	Acceptable
Control Element Drive Mechanisms	IV.3, pg 50	3.7.2		Y, SE Section 3.5.2.1.3	Acceptable
Reactor Coolant Pumps and Motor	IV.4, pg 51	4.3.5		Y, SE Section 3.5.2.1.5	Acceptable
SGs	IV.5, pg 51	4.3.4		Y, SE Section 3.5.2.1.4	Acceptable
Pressurizer	IV.6, pg 60	4.3.7		Y, SE Section 3.5.2.1.6	Acceptable
NSSS Auxiliary Equipment	IV.7, pg 60	4.3.8 4.3.9	Y, USAR 4.3.8 4.3.9		Acceptable
Balance of Plant					
Main Steam System	VI.2.1, pg 80	10		Y, SE Section 3.3.2.1.	Acceptable
Steam Dump System	VI.2.1, pg 80	10		Y, SE Section 3.3.2.1.2	Acceptable
Condensate and Feedwater System	VI.2.2, pg 83	10.2.2		Y, SE Section 3.3.2.1.3	Acceptable

Table 3.5.2 Mechanical and Civil Engineering - Summary of NRC Staff Review					
Topic	**Application Section and Page Number**	**USAR Section**	**Bounded by Current Licensing Basis (Y/N and Reference)**	**Bounded by Revised Analyses**	**NRC Staff Conclusion**
Auxiliary Feedwater System	VI.2.3, pg 86	9.4	Y, USAR 9.4		Acceptable
SG Blowdown System	VI.2.7, pg 89	11.1		Y, SE Section 3.3.2.1.5	Acceptable
Programs					
High-Energy Line Break Program	VII.6.5, pg 103	USAR Appendix M	Y, USAR Appendix M		Acceptable
Motor-Operated Valve Program	VII.6.2, pg 100	n/a		Y, SE Section 3.5.2.1.8 References 1 and 2	Acceptable
Air-Operated Valve Program	VII.6.3, pg 101	n/a		Y, SE Section 3.5.2.1.9	Acceptable

References for Table 3.5.2

1. Letter to Terry L. Patterson, OPPD, from Steven Bloom, "Closure of NRR Staff Review of Generic Letter 89-10 Program - Fort Calhoun Station," December 14, 1994.

2. Letter to S. K. Gambhir, OPPD, from Alan Wang, "Safety Evaluation of Licensee Response to Generic Letter 96-05, 'Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves', Fort Calhoun Station, Unit No. 1," May 15, 2001.

3.5.2.1 Impact of Power Uprate on Non-Bounding Mechanical and Civil Engineering Analyses

The NRC staff reviewed the FCS power uprate amendment as it relates to the effects of the power uprate on the structural and pressure boundary integrity of the NSSS and BOP systems. Affected components in these systems included piping, in-line equipment and pipe supports, the reactor pressure vessel (RPV), core support structures, reactor vessel internals, steam generators, control rod drive mechanisms (CRD), RCPs, and pressurizer. The staff's SE concerning the effects of the power uprate on the pertinent components is provided below.

3.5.2.1.1 Reactor Vessel

The proposed power uprate will increase the core power by approximately 1.6% above the currently licensed level of 1500 MWt. The licensee reported that the power increase will result in changing the design parameters identified in Table 3 of Attachment 2 to the licensee's August 28, 2003, application. Table 3 of the licensee's August 28, 2003, application, provides a comparison of the current design parameters and the revised design parameters at the proposed uprated power level of 1524 MWt.

The licensee evaluated the reactor vessel for the effects of the uprated design conditions provided in Table 3 of the August 28, 2003, application, with respect to the core power level of 1524 MWt. The evaluation was performed for the limiting vessel locations with regard to stresses and cumulative fatigue usage factors (CUFs) in each of the regions, as identified in the reactor vessel stress reports for the core power uprated conditions. The regions of the reactor vessel affected by the power uprate include the outlet and inlet nozzles, the head flange, studs, and vessel flange, control element drive mechanism (CEDM) housing, safety injection nozzles, inlet and outlet supports, core barrel stop lugs, core barrel snubber lugs and closure head instrumentation penetrations. In its October 31, 2003, response to the staff's request for additional information, the licensee indicated that the evaluation of the reactor vessel was performed in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, 1965 Edition which is the code of record. The table on page 12 of the licensee's October 31, 2003, letter provides the calculated maximum stresses and CUFs for the reactor vessel critical locations. The results indicate that the maximum primary plus secondary stresses are within the code allowable limits, and the CUFs remain below the allowable ASME Code limit of 1.0. Therefore, the staff agrees with the licensee's conclusion that the current design of the reactor vessel continues to be in compliance with the licensing basis codes for the proposed power uprate condition.

3.5.2.1.2 Reactor Core Support Structures and Vessel Internals

The licensee evaluated the reactor vessel core support and internal structures. The limiting reactor internal components were evaluated in Appendix 7 of the licensee's August 28, 2003, application. The licensee indicated that the design of the reactor internals was evaluated in accordance with requirements of the 1965 Edition of the ASME Code, Section III, through and including the 1967 Winter Addenda.

The licensee evaluated these critical reactor internal components considering the revised design conditions provided in Table 3 of the August 28, 2003, application for FCS for a core power of 1524 MWt. The calculated stresses for the limiting reactor internals provided in Appendix 7 are less than the Code allowable limits. The calculated CUFs as provided in the amendment request are less than the ASME code allowable limit of 1.0. Based on the above evaluations, the NRC staff agrees with the licensee's conclusion that the reactor internal components at FCS will be structurally adequate for the proposed power uprate.

3.5.2.1.3 Control Element Drive Mechanisms

The pressure boundary portion of the CEDMs are those exposed to the vessel/core inlet fluid. The licensee evaluated the adequacy of the CEDMs by comparing the design-basis input parameters against the revised design conditions in Table 3 of Appendix 2 in the licensee's August 28, 2003, application for the power uprate. The licensee indicated in their August 28, 2003, application that the key input parameters such as the hot leg maximum temperature, and maximum pressure for the uprated power condition are bounded by the design basis analysis. The licensee also indicated that with regard to the CEDM seal leak off temperature at which grease hardening may occur, there is sufficient margin to accommodate the hot leg temperature increase of 0.8°F. As a result of its evaluation, the licensee concluded that the FCS CEDMs will remain functional in accordance with its design requirements.

On the basis of its review, the NRC staff concurs with the licensee's conclusion that the current design of CEDMs continues to be in compliance with its design basis requirements for the proposed 1.6 percent power uprate.

3.5.2.1.4 Steam Generators

The licensee reviewed the existing structural and fatigue analyses of the steam generators at FCS and compared the power uprate conditions with the design parameters of the analysis of record for the steam generators at FCS. The comparison of key parameters is shown in Table IV-1 of the licensee's August 28, 2003, application for the current rated power and the proposed power uprate conditions. In its response to the staff's RAI, the licensee provided the steam generator key design parameters for design, design operating, current operating and the power uprate conditions. The comparison shows that the power uprate conditions are bounded within the range of design conditions.

The licensee evaluated the affected steam generator internal components such as feedwater sparger and sparger supports, separator deck and separators, dryer deck, shroud and shroud supports, for the power uprate condition. As a result of its evaluation, the licensee indicated that the calculated stress intensities and cumulative fatigue usage factors are less than the code-allowable limits and are, therefore, acceptable. The NRC staff concurs with the licensee's conclusion.

In addition, the licensee evaluated the flow-induced vibration of the U-bend tubes for the steam generators at FCS. The licensee indicated that the calculated fluid-elastic stability ratios provided in Table IV-2 of the licensee's August 28, 2003, application are less than the allowable limit of 1.0, and that the maximum flow-induced displacement values due to turbulence and the vortex shedding are insignificant. As a result, the licensee concluded that the flow-induced vibration of steam generator tubes will remain within the allowable limits for the power uprate condition. The NRC staff concurs with the licensee's conclusion.

On the basis of its review, the staff concludes that the licensee has demonstrated the maximum stresses and CUFs for the limiting steam generator components to be within the Code allowable limits and, therefore, acceptable for the proposed 1.6 percent power uprate.

3.5.2.1.5 Reactor Coolant Pumps

The licensee reviewed the existing design basis analyses of the FCS RCPs to determine the impact of the revised design conditions in Table 3 of the licensee's August 28, 2003, application.

After the core power uprate, the RCS pressure remains unchanged. The licensee indicated that the design parameter of the RCP temperature (reactor pressure vessel inlet) as provided in Table 3 of the licensee's August 28, 2003, application for the power uprate condition is not changed. Also, the RCS flow through the RCP will remain unchanged. Thus, the overall temperature, pressure and flow of reactor coolant through the RCPs remains unchanged. As a result of the evaluation, the licensee concluded that the proposed power uprate conditions will not impact the existing RCPs.

On the basis of its review, the NRC staff concurs with the licensee's conclusion that the RCPs, when operating at the proposed uprated conditions with a 1.6 percent power increase from the current rated power, will remain in compliance with the requirements of the codes and standards under which the FCS was originally licensed.

3.5.2.1.6 Pressurizer

The licensee evaluated the limiting design locations of the pressurizer components. The components in the lower end of the pressurizer (such as the surge nozzle, lower head well and penetration, and support skirt) are affected by the pressure and the hot leg temperature. The components in the upper end of the pressurizer (such as the spray nozzle, instrument nozzle, safety and relief nozzle, and upper head and shell) are affected by the pressure and the cold leg temperature for operation at the uprated conditions. The evaluation was performed using the ASME Code, Section III, 1965 Edition, through Summer 1966 addenda, which is the Code of record for the FCS pressurizer.

The key parameters in the current FCS pressurizer stress report were compared against the revised design conditions in Table 3 of the licensee's August 28, 2003, application for the proposed power uprate. The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg and cold leg (T_{cold}) temperatures are low. Because the proposed power uprate does not change the maximum RCS pressure and the pressurizer temperature (T_{sat}), the increase in T_{hot}, will reduce thermal stress for components at the lower end of the pressurizer for the proposed power uprate. Also, there is no change in T_{cold} for the power uprate condition. Thus, the power uprate condition does not impact the components in the upper end of the pressurizer. As a result of the above evaluation, the licensee concluded that the existing pressurizer components will remain adequate for plant operation at the proposed 1.6 percent power increase while the RCS pressure remains unchanged. The staff agrees with the licensee's conclusion.

3.5.2.1.7 NSSS Piping and Pipe Supports

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

The RCS components adhere to the following ASME 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

The licensee evaluated the NSSS piping and supports by reviewing the design basis analysis against the uprated power design system parameters, transients and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. The evaluation of RCS piping for the power uprate was based on USAS B31.1 Power Piping Code, 1955 Edition which is the Code of record. Other reactor coolant presssure boundary system piping and components were designed and evaluated in accordance with ASME Code, Section III, 1965 Edition, 1971 Edition and other Code and Code Cases as specified in the FCS USAR.

The licensee stated that a 1.67 percent power uprate was assumed in its evaluation. According to the licensee, at these conditions, the reactor coolant pressure and inlet temperature (Tc) remained unchanged at 2100 psia and 543°F, respectively. The licensee stated that 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. The RCS flow remains unchanged since the cold leg temperature remains unchanged at the increased thermal power. 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 licensee concluded that the change in temperature and pressure will not have significant effects on the NSSS piping. The NRC staff concurs with the licensee's conclusions because the RCS was designed for 2500 psia pressure at 650°F. The small temperature change from 568.2°F to 568.6°F will not have a significant impact on the material integrity of the reactor coolant system piping.

On the basis of its review of the licensee's submittal, the NRC staff concurs with the licensee's conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the design bases criteria, as defined in the USAR, and are therefore, acceptable for the proposed 1.6 percent power uprate.

3.5.2.1.8 BOP Systems and MOVs

The licensee evaluated the adequacy of the BOP systems based on comparing the existing design basis parameters with those for the core power uprate conditions. The BOP piping systems that were evaluated for the power uprate include main steam, steam dump, feedwater, steam generator blowdown, feedwater heater drains, and auxiliary feedwater systems. Table 1, "System and Program Review Summary" of the licensee's August 28, 2003, application summarizes the results of evaluations that were performed on the NSSS and BOP systems and components and plant programs. In Section VI of the licensee's August 28, 2003, application, the licensee reasonably demonstrated that the changes in these design parameters for the proposed power uprate are acceptable for the above affected piping systems. As a result of its evaluation, the licensee concluded that the existing design basis analyses for the BOP piping, pipe supports, and components for operation at the proposed 1.6 percent power uprate condition will be in compliance with the Codes of record.

The licensee also reviewed the programs, components, structures, and non-NSSS system issues as they relate to the power uprate. In Section VII of the licensee's August 28, 2003,

application, the licensee indicated that the MOV program used the maximum design basis parameters such as containment pressure following large pipe break, shutdown cooling pressure limit, safety and relief valve setpoints that are expected during the normal and emergency operation of MOVs. The plant operational parameter changes due to the power uprate are bounded by the conditions evaluated in the MOV program. Therefore, the licensee concluded that the safety-related MOVs will be capable of performing their intended functions at the uprated power condition.

In its response to the NRC staff's RAI, the licensee provided its review of the Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Operated Gate Valves," program associated with the pressure locking and thermal binding for safety related gate valves. The licensee indicated that the existing analysis conditions which preclude pressure locking or thermal binding of safety related power operated gate valves are bounding for the 1.6 percent power uprate. The licensee reviewed the evaluation of their GL 96-06, "Assurance of Equipment Operability and Containment Integrity Design-Basis Accident Conditions," program regarding the over-pressurization of isolated piping segments. The licensee concluded that the existing evaluation for GL 96-06 was based on the evaluation of the LOCA and MSLB design basis accidents, which were performed at 102 percent of the current rated power and are therefore, bounding for the proposed power uprate of 101.6 percent rated power level. On the basis of the above review, the NRC staff concurs with the licensee's conclusions that the power uprate will have no adverse effects on the safety-related valves and that conclusions of the OPPD GL 95-07, and GL 96-06, as well as GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" programs, remain valid.

As a result of the above evaluation, the NRC staff concludes that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and continue to satisfy the design basis requirements for the proposed power uprate.

3.5.2.1.9 Air-Operated Valve (AOV) Program

The FCS AOV program classified the AOVs in three categories:

1. AOVs that perform an active function of high safety significance and are safety-related.

2. AOVs that perform an active function that does not have high safety significance but are safety-related. This category also included non-safety-related AOVs that are classified as high risk.

3. AOVs that are not safety-related, do not support safety-related systems and are not classified as high risk.

The licensee has determined that the Category 1 and 2 AOVs identified in this review are not affected by the MUR or are bounded by the MUR uprate conditions. On the basis of this review, the NRC staff concurs with the licensee's conclusion that the AOV program ensures that the AOVs will be operable at the uprated conditions and therefore are acceptable for the proposed power uprate.

3.5.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on NSSS and BOP systems and components with regard to stresses, cumulative usage factors, flow induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs. The NRC staff has determined that the results of the licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Where additional assessments and analyses were necessary, the NRC staff has reviewed these assessments and analyses and finds that the licensee has satisfactorily addressed the areas discussed above, the input parameters of the analyses adequately represent the plant conditions at the proposed uprated power level, and the analytical results will continue to meet applicable requirements. The NRC staff concurs with the evaluations performed by the licensee for the NSSS and BOP piping, components, and supports, the reactor vessel and internal components, the CEDMs, steam generators, RCPs and the pressurizer. The NRC staff finds the licensee's evaluation to be bounded by the licensing codes of record and the original design basis and, therefore, concludes the foregoing components to be acceptable for MUR uprate operations at the proposed core power level of 1524 MWt. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the areas of mechanical and civil engineering.

3.6 Materials and Chemical Engineering

3.6.1 Regulatory Evaluation

The NRC staff's review in the area of materials and chemical engineering covers the effects that the proposed MUR power uprate would have on (1) the structural integrity evaluations for the reactor vessel, (2) steam generator tube integrity, and (3) erosion/corrosion programs (RIS 2002-03, Attachment 1, Section IV, Items 1.C through 1.F). The NRC staff's review in this area focuses on the impact of the proposed MUR power uprate on (1) the P-T limits for the reactor vessel and reactor coolant pressure boundary, (2) evaluations for ensuring the integrity of the reactor vessel and reactor coolant pressure boundary against pressurized thermal shock (PTS), (3) evaluations for ensuring that the reactor vessel materials have sufficient levels of upper-shelf energy (USE), (4) surveillance capsule withdrawal schedules, (5) licensee programs for addressing steam generator tube degradation mechanisms, and (6) erosion/corrosion. This review is conducted to verify that the results of the licensee's analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, 10 CFR 50.55a, and 10 CFR Part 50, Appendices G and H, following implementation of the proposed MUR power uprate. Additional guidance for the NRC staff's review of the topics within the materials and chemical engineering area include the guidance contained in SRP Chapters 4, 5, and 6.

3.6.2 Technical Evaluation

The NRC staff has reviewed the licensee's application as related to the materials and chemical engineering areas discussed above and determined that the existing analyses of record bound some of the areas altered by the proposed operation of the plant at the uprated power level. The NRC staff evaluation of the effects of the proposed power uprate on areas not bounded by existing staff analyses or areas with special evaluations and considerations are discussed in Section 3.6.2.1 of this SE. The results of the NRC staff's review for the areas discussed above

within the scope of the materials and chemical engineering, concerning vessel and internals integrity and welding, are summarized in Table 3.6.2 below.

Table 3.6.2
Materials and Chemical Engineering - Summary of NRC Staff Review

Topic	Application Section and Page Number	USAR Section	Bounded by NRC-approved Analysis (Y/N and Reference)	NRC Staff Conclusion
Component Integrity				
Steam Generator Structural Integrity Evaluation	IV.5.1, pg 51	4.3.4	Y, SE Section 3.5	Acceptable
Steam Generator Tube Vibration and Wear and Other Modes of Tube Degradation	VI.5.2, pg 52	4.3.4	Y, SE Section 3.5	Acceptable
Regulatory Guide 1.121 Analysis	VI.5.2, pg 52	n/a	Y, SE Section 3.6.2.1.1	Acceptable
Flow-Accelerated Corrosion	VII.6.4, pg 102	n/a	Y, SE Section 3.6.2.1.2	Acceptable
Structural Integrity and Metallurgy				
10 CFR Part 50 Appendix G – P-T Limits	IV.1.1.2, pg 43	n/a	Y, SE Section 3.6.2.1.4 References 3, 4	Acceptable
10 CFR Part 50 Appendix G - USE	IV.1.1.3, pg 43	n/a	Y, SE Section 3.6.2.1.4 References 2, 4	Acceptable
10 CFR 50.61 PTS Events	IV.1.1, pg 42	n/a	Y, SE Section 3.6.2.1.4 References 2, 4	Acceptable
10 CFR Part 50 Appendix H RPV Surveillance Program	IV.1.1.4, pg 43	n/a	Y, SE Section 3.6.2.1.4 References 2, 3, 4, 5	Acceptable
Leak-Before-Break Analyses	IV.2.3, pg 49	n/a	Y, SE Section 3.6.2.1.3 Reference 1	Acceptable
Structural Integrity of Control Element Drive Mechanism Nozzles	IV.3, pg 50	3.7.2	Y, SE Section 3.5.2.1.3	Acceptable
Structural Integrity of Reactor Vessel Internals	IV.1.2.1, pg 44	3.7.1	Y, SE Section 3.5.1.1.2 References 4, 5	Acceptable
Structural Integrity of the Reactor Coolant Pump Flywheels	IV.4, pg 51	4.3.5	Y, SE Sections 3.5.2.1.5 and 3.6.2.1.4 Reference 6	Acceptable

References to Table 3.6.2

1. Letter to T. L. Patterson, OPPD, from Steven Bloom, "Amendment 165 to Facility Operating License DPR-40 - Revise TS 2.1.4 to Implement RCS Leak-Before-Break Detection Criteria," August 25, 1994.

2. NRC Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," March 16, 2001.

3. NRC Safety Evaluation, "Fort Calhoun Station, Unit 1 - Reactor Vessel Surveillance Capsule Removal Schedule Change," May 2, 2002.

4. NRC Safety Evaluation, "Staff Evaluation Regarding License Amendments and Exemption Requests Related to Implementation of a Pressure-Temperature Limit Report for Fort Calhoun Station Unit 1," June 10, 2003.

5. ASME Boiler and Pressure Vessel Code, Section III, Article 4, 1965 Edition through and including the 1967 Winter Addenda.

6. Fort Calhoun Updated Safety Analysis Report, Release 4, May 30, 2002.

3.6.2.1 Impact of Power Uprate on Non-Bounding Materials and Chemical Engineering Analyses

3.6.2.1.1 RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," Analysis

RG 1.121 describes an acceptable method for establishing the limiting safe condition of degradation in the tubes, beyond which tubes found defective by the established inservice inspection shall be removed from service. The level of acceptable degradation is referred to as the repair limit. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the resulting structural limit an allowance for continued growth of the flaw and an allowance for eddy current measurement uncertainty. In terms of the MUR power uprate, the structural limit and corrosion rate are affected by parameters such as temperature change and differential pressure (e.g., a change in temperature affects the corrosion rate).

While the licensee did not provide a specific reference to RG 1.121, the intent of RG 1.121 is contained in the licensee’s August 28, 2003, application where the licensee showed that the change in temperature and differential pressure across the steam generator tubes is essentially identical to existing conditions. In the August 28, 2003, application, the licensee indicated that the purpose of the steam generator program is to ensure tube structural and leakage integrity. The licensee’s program includes several program elements, including assessment of existing degradation mechanisms in the reactor coolant boundary, steam generator inspections, assessment of tube integrity, maintenance, plugging, and repairs, primary-to-secondary leakage monitoring, maintenance of steam generator secondary side integrity, water chemistry, foreign material exclusion, and self-assessment of the steam generator program.

The licensee concluded that, as a result of the MUR power uprate, the 0.8°F increase in T_{hot} will marginally increase the stress corrosion rate in the steam generator tubes, and that the existing plugging margin and inspection program elements are sufficient to ensure tube integrity. Given the licensee’s conclusions regarding its steam generator program and the extremely small change in operating conditions resulting from the 1.6 percent MUR power uprate, the NRC staff concludes that the plugging limits are adequate.

3.6.2.1.2 Flow Accelerated Corrosion (FAC)

The purpose of the FAC program is to predict, detect, monitor, and mitigate FAC in plant systems. The licensee utilizes the Electric Power Research Institute (EPRI) CHECWORKS™ program to model these 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 licensee's FAC program was reviewed in support of the 1.6 percent MUR power uprate program. The licensee noted that flow rates and temperatures for piping components within the scope of the FAC program remain within the system design specifications, and that all FAC-susceptible components, suitable for modeling, are modeled using EPRI's CHECWORKS™ program, version 1.0g. The licensee noted that the 1.6 percent MUR power uprate conditions (i.e., changes in operating pressure, temperature, quality, and velocity) do have an effect on FAC wear rates in several piping systems. In order to understand the impact of the 1.6 percent MUR power uprate, the staff asked the licensee to identify the plant component most impacted by the 1.6 percent MUR power uprate and report the expected increase in FAC rate. The licensee responded that the moisture separator drain lines would experience a projected increase in the FAC rate of 7.5 percent due to the 1.6 percent MUR power uprate. The components in these lines with the highest wear rate were the inlet nozzles to the moisture separator drain tanks. The licensee noted that the increases in wear rates were not a significant concern, since (a) many of these lines were replaced (with the highest wearing segments being replaced with a corrosion resistant material), (b) line pressure is relatively low, and (c) the CHECWORKS™ life predictions of these lines are considered to be accurate within the 50 percent tolerance band.

The licensee concluded that changes to piping wear rates at 1.6 percent MUR power uprate conditions were identified, and that the FAC program is adequate to support the 1.6 percent MUR power uprate. Based on the information provided above, the staff agrees with the licensee's conclusions.

3.6.2.1.3 Leak-Before-Break Analysis

FCS's leak-before-break (LBB) methodology is based on WCAP-9558, Revision 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," Westinghouse Energy Systems, dated May 1981. The NRC approved FCS's methodology on August 25, 1994. The LBB analyses justified the elimination of large primary loop pipe rupture from the structural design basis for FCS. To demonstrate the continued acceptability of the elimination of the RCS primary loop pipe rupture from the structural design basis for the MUR power uprate program, the following objectives must be achieved:

1. Demonstrate that margin exists between the critical crack size and a postulated crack that yields a detectable leak rate.

2. Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.

3. Demonstrate margin on applied load.

4. Demonstrate that fatigue crack growth is negligible.

These objectives were met by the analyses discussed in WCAP-9558, Revision 2, and Calculation "FC-05462, Rev. 8, 'Response Time of Containment Air Monitoring System'."

The licensee stated that there is no change in loads on the primary loop piping due to the uprate parameters. The licensee stated that the effects of material properties due to the changes in temperature were bounded 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. WCAP-9558, Revision 2 was based on enveloped design loads of axial tension-1800 kips, bending moment 45,600 inch-kips, RCS pressure of 2250 psi, and RCS temperature of 600°F. The licensee stated the enveloped nozzle loads for FCS were reported as axial load-1650 kips, and bending moment 9800 inch-kips. The licensee concluded that since FCS operates at a much lower RCS temperature and pressure, the LBB-evaluated enveloped design loads still provide adequate margin in regards to crack stability conclusions with an increase in RCS temperature of 0.8°F.

The licensee stated that previous LBB leak detection capability for radiation monitoring was based on a 1500 MWt core inventory, and total integrated dose source term. With the implementation of an alternate source term (AST) at 1530 MWt, an assessment was made to ensure that leak detection margins were still met. The licensee concluded that LBB leak detection capability was not impacted by the power uprate condition, and therefore the existing LBB analyses and revised radiation monitoring analysis conclusions remain applicable for the FCS MUR uprate program.

The NRC staff has reviewed the information provided by the licensee and concurs with their conclusion. This is based on the fact that the NRC staff considers the materials used for the RCS will be adequate for the 1.6 percent power uprate since the pressure and temperatures at FCS are less than the licensee's original design evaluation.

Furthermore, the design basis LOCA forces due to postulated primary loop guillotine breaks have been eliminated using the loop LBB methodology. With the use of LBB technology, LOCA forces for the power uprate condition were derived based on postulation of breaks in three branch lines at the surge line nozzle on the hot leg, the accumulator line nozzle at the cold leg, and the residual heat removal line nozzle on the hot leg. As such, the design basis LOCA hydraulic forcing functions are bounding for the LOCA loads at the uprated power condition. Therefore, the licensee concluded that the existing stresses, fatigue usage factors and loads remain bounding for the power uprate for the NSSS components including the reactor coolant loop piping, the primary equipment nozzles, the primary equipment supports, pipe supports and the auxiliary equipment (i.e., heat exchangers, pumps, valves and tanks). As a result, these components will continue to be in compliance with the Code of record at FCS.

3.6.2.1.4 Structural Integrity and Metallurgy

Regulatory Requirements

Requirements for Generating P-T Limits

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50, provides the requirements for generating P-T limits curves. Section IV.A.2 of Appendix G requires that P-T limits for reactor vessels of light water reactors must be at least as conservative as the P-T limit generation methods of Appendix G to Section IX of the ASME Boiler and Pressure Vessel Code (ASME Code). For materials in the beltline region of the reactor vessel, the rule requires that the calculations of P-T limits take into account the effects of neutron irradiation on the reference temperatures for nil ductility (i.e., RT_{NDT} values) for the materials used to fabricate the reactor vessel and to incorporate any relevant reactor vessel surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," reactor vessel material surveillance program.

Additional guidance for the staff's review of the reactor vessel P-T limit curves is provided in RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," SRP Chapter 5.3.2, "Pressure-Temperature Limits," and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements."

Requirements for Reactor Vessel USE

Section IV.A.2 of Appendix G of 10 CFR Part 50, requires that the reactor vessel beltline materials have a minimum USE value of 75 ft-lb in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analytical engineering analyses (i.e., through equivalent margins analyses) that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI to the ASME Code. For materials in the beltline region of the vessel, the rule requires USE calculations to account for the effects of neutron irradiation on the USE values for the materials and to incorporate any relevant reactor vessel surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, reactor vessel material surveillance program.

Additional guidance for the NRC staff's review of USE analyses is provided in RG 1.99, Revision 2, SRP Chapter 5.3.2, and Branch Technical Position MTEB 5-2. Appendix K to Section XI of the ASME Code and RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 ft-lb," may also be used as guidance when USE equivalent margins analyses are required.

Requirements for Ensuring Reactor Vessel Integrity Against PTS Events

The requirements for protecting the reactor vessels of pressurized water reactors against PTS events are stated in 10 CFR 50.61. The rule requires RV materials made of carbon or low-alloy steel materials to meet a maximum screening criterion for nil-ductility reference temperatures (i.e., RT_{PTS} values). The rule's screening criteria are 270°F for axial weld materials and base metal materials (i.e., plates or forging materials) and 300°F for circumferential weld materials.

The rule provides methods for calculating these RT_{PTS} values. For materials in the beltline region of the vessel, the rule requires the calculations to take into account the effects of neutron irradiation on the RT_{PTS} values for the materials and to incorporate any relevant reactor vessel surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, reactor vessel material surveillance program.

Requirements for Reactor Vessel Surveillance Program

Regulatory requirements related to the establishment of a facility's reactor vessel surveillance capsule program and withdrawal schedule are given in Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50, which also references the guidance in the American Society for Materials and Testing Standard Practice E 185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Additional guidance regarding the evaluation of the materials surveillance program may be found in SRP Chapter 5.3.1, "Reactor Vessel Materials."

Structural Integrity Requirements for Reactor Vessel Internal Components (Reactor Vessel Internals)

Structural integrity maintenance of the reactor vessel internals is required in order to demonstrate that the functional requirements of the reactor vessel internals are met. These functional requirements include core support and ECCS performance aspects. As such, the structural integrity of the reactor vessel internals is linked to regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry.

Additional guidance regarding the evaluation of the structural integrity of RV internals may be found in SRP Chapter 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and in ASME Code Sections III and XI.

Structural Integrity Requirements for RCP Flywheels

Structural integrity maintenance of the RCP flywheels is important to ensure the RCP flywheels can maintain a continuous coastdown from 120 percent of the design rotor speed for the RCP impellers and to preclude the potential for missile generation as a result of their failure. The evaluation of RCP flywheel integrity is related to Appendix A to 10 CFR Part 50, GDC 1 and 4, or for those plants licensed prior to the development of Appendix A to 10 CFR Part 50, similar requirements which were imposed during the NRC staff's review of the facility's operating license.

Technical Evaluation

Upper Shelf Energy and PTS Analyses

The licensee assessed the effect that the proposed MUR power uprate would have on the structural assessments for the reactor vessel in Section IV.1.1 of their August 28, 2003, application. These structural integrity assessments included evaluations of the reactor vessel materials relative to USE and PTS screening criteria in 10 CFR Part 50, Appendix G and 10 CFR 50.61, respectively. The licensee concluded that the proposed 1.6 percent power uprate will not have a significant effect on the structural integrity evaluations. The projected end-of-life (EOL) fluence for the reactor vesssel is based on 48 EFPYs of operation and a core thermal power level of 1524 MWt.

For the evaluation of USE, the licensee indicated that the power uprate will cause an insignificant increase to the fluence at the 1/4T location for the limiting weld. The predicted USE decrease in accordance with Figure 2 of RG 1.99, Rev. 2, remains essentially unchanged and will remain above the limit of 50 ft-lb. The NRC staff performed an independent calculation of the USE values for the reactor vessel beltline material using the uprated neutron fluences for the reactor vessel 1/4T location at EOL. The beltline of the reactor vessel is limited by the USE drop that is projected to occur in weld 2-410. The NRC staff projected the EOL USE value for the limiting material to be 54.6 ft-lb. This USE value meets the Appendix G to 10 CFR Part 50 screening criterion of 50 ft-lb. Based on the above, the staff concludes that RV beltline materials for the reactor vessel will continue to comply with the USE requirements in Appendix G to 10 CFR Part 50.

The NRC staff performed an independent calculation of the material property values (i.e., RT_{PTS} values) for the reactor vessel beltline materials, in order to assess the effects that the uprated condition would have on the PTS evaluations for the facility and also to validate the licensee's conclusions. For the evaluation of the PTS, the beltline of the reactor vesssel is limited by weld 3-410 comprised of weld wire heats 13253/12008, which was projected to be the closest to the PTS screening criteria at the EOL. The NRC staff projected the RT_{PTS} value for weld 3-410 to be 269.80°F. For the reactor vessel, the NRC staff wants to emphasize that, for the limiting material, the RT_{PTS} value (269.80°F) is 0.20°F within the screening criteria in 10 CFR 50.61 for axial welds and base metal materials, which is 270°F. Section 50.61(b) requires licensees to update the PTS assessment whenever there is a significant change (changes to the PTS values are considered significant if either the previous value or current value, or both values, exceed the screening criterion prior to the expiration of the operating license) in the projected values of the PTS, or upon request for a change in the expiration date for operation of the facility. Section 4.2.2.2 of Reference 6 to Table 3.6.2 of this SE, describes the licensee's reactor vessel integrity program, which will monitor future core loadings to ensure that no beltline material will exceed the PTS screening criteria. Based on the above, the NRC staff finds that the reactor vessel beltline materials for the FCS reactor vessel will continue to have a safety margin against the impacts of PTS events.

3.6.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on reactor vessel integrity, steam generator tube integrity, and

erosion/corrosion programs. The technical areas reviewed by the NRC staff are those discussed in Section 3.6.1 of this SE. Based on the above, the NRC staff concludes that the licensee has adequately addressed these impacts and has demonstrated that the plant will continue to meet the applicable requirements following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the materials and chemical engineering issues discussed above.

3.7 Dose Consequences Analysis

3.7.1 Regulatory Evaluation

The NRC staff's review covers the impact of the proposed MUR power uprate on the results of dose consequences analyses (RIS-2002-03). The review is conducted to verify that the results of the licensee's dose consequences analyses continue to meet the acceptance criteria in 10 CFR 50.67 and GDC 19, as applicable, following implementation of the proposed MUR power uprate.

3.7.2 Technical Evaluation

The NRC staff reviewed the impact of the proposed MUR power uprate changes on DBA radiological analyses, as documented in Chapter 14 of the FCS USAR. In its August 28, 2003, application, the licensee stated that the Chapter 14 radiological consequences calculations were recently updated to reflect implementation of AST methodology (RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"). As such, the licensee re-analyzed all accident radiological consequences to the new methodology and determined that criteria as specified in 10 CFR 50.67 and GDC 19 as stated in RG 1.183 were met. 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, MSLB, steam generator tube rupture, waste gas decay tank and liquid waste tank radiological dose consequence assessments. Thus, the analyses of record for all offsite radiological consequences and control room doses bound the conditions for the proposed 1.6 percent power uprate. The NRC staff verified that the existing FCS USAR Chapter 14 radiological analyses source term and steam release assumptions bound the proposed 1.6 percent power uprate conditions for analyses of the offsite radiological consequences of DBAs.

The NRC approved License Amendment 201 for implementation of RG 1.183 (AST) in the NRC letter dated December 5, 2001, "Fort Calhoun Station, Unit No. 1 Issuance of Amendment (TAC NO. MB1221)." The NRC staff found the licensee's analyses to be acceptable, as stated in the SE for License Amendment 201. Section 2.2.1 of the SE addresses use of 102 percent core power in accordance with RG 1.49, "Power Levels of Nuclear Power Plants."

Based on the above discussion, the NRC staff finds that the USAR Chapter 14 radiological analyses remain bounding for the proposed 1.6 percent power uprate to 1524 MWt. These analyses of record show that, for the proposed power uprate, the radiological consequences of postulated DBAs continue to meet the dose limits given in 10 CFR 50.67 and 10 CFR Part 50, Appendix A, GDC 19.

3.7.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on dose consequences analyses. As set forth above, the NRC staff has determined that the results of the licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to dose consequences analyses.

3.8 Human Factors

3.8.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions (RIS 2002-03, Attachment 1, Section VII, Items 1 through 4). The NRC staff's human factors evaluation is conducted to confirm that operator performance will not be adversely affected as a result of system changes necessary for the proposed MUR power uprate. The NRC staff's review covers the licensee's plans for addressing changes to operator actions, human-system interfaces, and procedures and training necessary for the proposed MUR power uprate. The NRC's acceptance criteria for human factors are based on 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR 55.59, and GDC-19.

3.8.2 Technical Evaluation

Items 1 through 4 in Section VII of Attachment 1 to RIS 2002-03 define the scope of the NRC staff's review for the human factors area. The licensee addressed these items in its August 28, 2003, application. The following is a summary of the NRC staff's evaluation related to the human factors area.

3.8.2.1 Operator Actions

The proposed MUR power uprate is not expected to have any significant effect on the manner in which the operators control the plant during normal operations or transient conditions. All operator actions taken credit for by previous safety evaluations will still be valid following the implementation of the MUR power uprate. Operator actions that would be required by the failure of the CROSSFLOW system or one of its inputs will be provided in procedures that are being revised and/or developed to support the MUR power uprate.

3.8.2.2 Emergency and Abnormal Operating Procedures

There are no emergency operating procedures (EOPs) or abnormal operating procedures (AOPs) that currently reference the use of the Emergency Response Facility Computer System (ERFCS) for the purpose of determining plant power level. Several procedures within the EOP/AOP program were evaluated for impact based on the proposed MUR power uprate plant parameters. Two AOPs were identified for revision. These revisions will incorporate guidance on required power reduction in the event of a loss-of-power to CROSSFLOW, failure of the ERFCS, CROSSFLOW system or certain inputs to the secondary calorimetric calculation.

These revisions, including revisions to the AOP Technical Basis Document will be processed under the EOP/AOP control program.

3.8.2.3 Control Room Controls, Displays, and Alarms

System status and alarms associated with the CROSSFLOW system will be provided through the ERFCS. Two new CROSSFLOW system status pages will be available for monitoring CROSSFLOW system performance, and the secondary calorimetric status page is being modified to include the value of the CROSSFLOW correction factor being applied to feedwater flow. Three alarms associated with the CROSSFLOW system will be available for display on the ERFCS alarm summary display screen. No new annunciators or control board controls are introduced by the MUR power uprate. Response to the CROSSFLOW system alarms is being proceduralized in the annunciator response procedure. Training is being conducted on the changes to the ERFCS alarms and procedures associated with the MUR power uprate and will be completed prior to implementation.

3.8.2.4 Control Room Plant Reference Simulator

The FCS simulator certification renewal was submitted in a letter from W. G. Gates to T. E. Murley dated February 13, 1991, pursuant to 10 CFR 55.45(b)(5). The proposed MUR power uprate is not expected to have a significant effect on any simulated systems, and the simulator is not expected to be modified, except that provisions will be made to allow for the simulation of one or more malfunctions to the CROSSFLOW system. The malfunction(s) will be used to exercise the capability of the operators to respond to the failure of the system and utilize the appropriate procedural guidance to maintain plant power level within required limits. Incorporation of the revised secondary calorimetric software into the simulator ERFCS will be accomplished under the simulator’s configuration management system.

3.8.2.5 Operator Training Program

Training will be required prior to the implementation of the MUR power uprate and revised procedures and training materials. This training is being conducted in the classroom with supporting activities in the control room simulator. It includes discussions of the MUR process, changes to the secondary calorimetric (XC105) calculation program, including the addition of the CROSSFLOW system, and procedures associated with normal operation and failures of the CROSSFLOW system. Revisions to procedures are being completed under the engineering change process, and have been incorporated into the current training package supporting the power uprate. Revision of existing training materials will be accomplished using the training program configuration management system, which will track action items associated with the MUR power uprate through to completion.

3.8.3 Summary

The NRC staff has reviewed the licensee’s planned actions related to the human factors area and has determined that the licensee has adequately considered the impact of the proposed MUR power uprate on changes to operator actions, procedures, plant hardware, and associated training programs to ensure that operators’ performance is not adversely affected by the proposed MUR power uprate. Accordingly, the NRC staff concludes that the licensee will

continue to meet the requirements of 10 CFR 50.54(i) and (m), 10 CFR 50.120, and 10 CFR 55.59 following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the human factors aspects of required system changes.

4.0 LICENSE AND TECHNICAL SPECIFICATION CHANGES

4.1 Change to Renewed Facility Operating License No. DPR-40

The licensee proposes to revise paragraph 3.A in Renewed Facility Operating License DPR-40 to authorize operation at a steady-state reactor core power level not in excess of 1524 MWt (100-percent power).

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.2 Change to TS 1.0 – Rated Power

The licensee proposes to revise the definition of "Rated Power" in TS 1.0 to reflect the increase from 1500 MWt to 1524 MWt.

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.3 Changes to the Bases Sections

The Bases sections of TS 2.1.6 and TS 3.5 have been revised to reflect the proposed increase in power. TS 5.20, "Technical Specification (TS) Bases Control Program," ensures the continuing accuracy and adequacy of the Bases. Therefore, the Bases changes have had the appropriate reviews performed and have the administrative controls to ensure the accuracy and adequacy of the changes. The NRC staff has reviewed these changes and has no objections to them.

5.0 REGULATORY COMMITMENTS

To support the proposed MUR power uprate, the licensee made the following commitments (as stated):

1. Modifications associated with the MUR power uprate will be completed prior to implementation. This includes implementation of control room alarm functions.

2. Figure 2-1 of the PTLR will be revised prior to the reactor vessel reaching 39.9 EFPYs of operation.

The NRC staff considered the above commitments as part of its evaluation in Section 3.0 above and finds the commitments appropriate for the proposed MUR power uprate. The NRC staff has conditioned the implementation of the proposed MUR power uprate on the above commitments.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 54751). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: List of Abbreviations

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LIST OF ABBREVIATIONS

AC	alternating current
AFW	auxiliary feedwater
AL	analytical limit
AOP	abnormal operating procedure
ASME	American Society of Mechanical Engineers
AST	alternate source term
ATWS	anticipated transient without scram
AV	allowable value
BOP	balance of plant
CCW	component cooling water
CEA	control element assembly
CEDM	control element drive mechanism
COT/CFT	channel operational test/channel functional test
CUF	cumulative fatigue usage factor
DBA	design basis accident
DC	direct current
DNBR	departure from nucleate boiling ratio
DSS	diverse shutdown system
ECCS	emergency core cooling system
EOL	end-of-life
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ERFCS	emergency response facility computer system
EQ	environmental qualification
ESF	engineered safety feature
FAC	flow accelerated corrosion
FCS	Fort Calhoun Station
GDC	General Design Criterion
GL	Generic Letter
HVAC	heating, ventilating, and air conditioning
ISA	Instrument Society of America
LBB	leak-before-break
LHR	linear heat rate
LOCA	loss-of-coolant accident
LSSS	limiting safety system setpoint
LTOP	low temperature overpressure protection
MOV	motor operated valve
MSLB	main steam line break
MUR	measurement uncertainty recapture
MWt	megawatts thermal
NSSS	nuclear steam system supplier
OL	Operating License
OPPD	Omaha Public Power District
P-T	pressure-temperature
PTLR	pressure temperature limits report

PTS	pressurized thermal shock
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RIS	Regulatory Issue Summary
RPS	reactor protection system
RPV	reactor pressure vessel
RTD	resistance temperature detector
RTP	rated thermal power
SBO	station blackout
SE	safety evaluation
SFPC	spent fuel pool cooling
SRP	Standard Review Plan
TLU	total loop uncertainty
TR	Topical Report
TS	Technical Specifications
UFM	ultrasonic flow measurement
USAR	Updated Safety Analysis Report
USE	upper shelf energy