May 2, 2008

Mr. David J. Bannister
Vice President and CNO
Omaha Public Power District
Fort Calhoun Station FC-2-4
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: MODIFICATION OF CONTAINMENT SPRAY ACTUATION LOGIC AND DAMPERS IN CONTAINMENT AIR COOLING AND FILTERING SYSTEM (TAC NOS. MD6204 AND MD7043)

Dear Mr. Bannister:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 255 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 (TSs) and License Conditions in response to your applications dated July 30 and October 19, 2007, and supplemented by letters dated August 31 and December 12, 2007, and February 21, March 28, and April 4 and 10, 2008.

This amendment revises TS Limiting Condition for Operation (LCO) 2.4, "Containment Cooling," LCO 2.14, "Engineered Safety Features System Initiation Instrumentation Settings," and LCO 2.15, "Instrumentation and Control Systems"; TS Surveillance Requirement (SR) 3.1 "Instrumentation and Control," SR 3.5(4), "Containment Isolation Valves Leak Rate Tests (Type C Tests)," and SR 3.6(3), "Containment Recirculating Air Cooling and Filtering System"; and associated TS Basis documents and Updated Safety Analysis Report sections to modify the containment spray system actuation logic to preclude automatic start of the containment spray pumps for a loss-of-coolant accident. This amendment also revises TS SR 3.6(3)a. to delete SRs for testing of the containment air cooling and filtering system (CACFS) emergency mode dampers and replace it with a surveillance to verify that the dampers are in the accident positions in all operating plant modes and deletes the requirement in TS SR 3.6(3)b. to remotely operate dampers.

The amendment added license conditions related to the replacement and testing of containment air cleaning and filtering (CACF) unit HEPA (high-efficiency particulate air) filters and surveillance testing of the CACF unit relief ports. The license conditions require administrative controls pending the completion of detailed analysis and confirm commitments for the licensee to submit TS amendments by October 31, 2008.

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/

Michael T. Markley, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 255 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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/

Michael T. Markley, Senior Project Manager
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Docket No. 50-285

Enclosures: 1. Amendment No. 255 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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(*) Concurrence via SE (**) See previous concurrence

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/SCVB/BC	NRR/EICB/BC	NRR/AADB/BC
NAME	MMarkley (**)	JBurkhardt	RDennig (**)	WKemper (*)	RTaylor (*)
DATE	4/23/08	4/25/08	4/23/08	4/22/08	2/8/08
OFFICE	NRR/SSIB/BC	NRR/ITSB/BC	OGC – No Legal Objections	NRR/LPL4/BC	
NAME	MScott: RArchizel for MLS (**)	RElliott: MHamm for RBE (**)	JBiggins (**)	THiltz:TGH	
DATE	4/23/08	4/24/08	5/1/08	5/2/08	

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cc:
Winston & Strawn
ATTN: James R. Curtiss, Esq.
1700 K Street, N.W.
Washington, DC 20006-3817

Chairman
Washington County Board of Supervisors
P.O. Box 466
Blair, NE 68008

Mr. John Hanna, Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 310
Fort Calhoun, NE 68023

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Ms. Julia Schmitt, Manager
Radiation Control Program
Nebraska Health & Human Services R & L
Public Health Assurance
301 Centennial Mall, South
P.O. Box 95007
Lincoln, NE 68509-5007

Mr. Thomas C. Matthews
Manager - Nuclear Licensing
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

Ms. Melanie Rasmussen
Radiation Control Program Officer
Bureau of Radiological Health
Iowa Department of Public Health
Lucas State Office Building, 5th Floor
321 East 12th Street
Des Moines, IA 50319

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 255
Renewed License No. DPR-40

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

   A. The applications for amendment by the Omaha Public Power District (the licensee), dated July 30 and October 19, 2007, and supplemented by letters dated August 31 and December 12, 2007, and February 21, March 28, and April 4 and 10, 2008, 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 applications, 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 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. 255, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended to revise paragraph 3.F to Renewed Facility Operating License No. DPR-40 to read as follows:

F. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 255, are hereby incorporated into this license. Omaha Public Power District shall operate the facility in accordance with the Additional Conditions.

3. Appendix B, "Additional Conditions," to Renewed Facility Operating License DPR-40 is amended to designate the existing license condition for Amendment No. 181 as (1), and add new license conditions (2) and (3), designated as Amendment No. 255, to read as follows:

(2) This license shall be deemed to contain the following specified conditions and specified actions:

(a) During the 2008 RFO [refueling outage], replace the CACF [containment air cleaning and filtering] unit HEPA [high-efficiency particulate air] filters not previously replaced during 2006;

(b) During the 2008 RFO, ensure all CACF unit HEPA filters meet the 2" wc [water column] differential pressure limitation;

(c) Implement and maintain the newly created HEPA filter replacement criteria in Procedure PE-RR-VA-0209 unless superseded by other criteria via the license amendment process; and

(d) Submit a license amendment request (LAR) by October 31, 2008, that will add the HEPA filter testing and replacement criteria to the FCS [Fort Calhoun Station] Technical Specifications.

(3) This license shall be deemed to contain the following specified conditions and specified actions:

(a) During the 2008 RFO, perform the surveillance operability testing of the containment cooling unit relief ports; and

(b) Submit a LAR by October 31, 2008, that will add surveillance operability and testing of the containment cooling unit relief ports to the FCS Technical Specifications.

4. The license amendment is effective as of its date of issuance and shall be implemented prior to startup from the 2008 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Renewed Facility
Operating License No. DPR-40
and Technical Specifications

Date of Issuance: May 2, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 255

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages of the Renewed Facility Operating License No. DPR-40, the Appendix A Technical Specifications, and Appendix B, Additional Conditions, with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Renewed Operating License

REMOVE	INSERT
- 3 -	- 3 -
- 4 -	- 4 -

Technical Specifications

REMOVE	INSERT
2.4 - Page 2	2.4 - Page 2
2.4 - Page 3	2.4 - Page 3
2.4 - Page 5	2.4 - Page 5
2.14 - Page 2	2.14 - Page 2
2.14 - Page 4	2.14 - Page 4
2.14 - Page 5	2.14 - Page 5
2.15 - Page 9	2.15 - Page 9
2.15 - Page 10	2.15 - Page 10
2.15 - Page 11	2.15 - Page 11
2.15 - Page 12	2.15 - Page 12
3.1 - Page 8	3.1 - Page 8
3.1 - Page 13	3.1 - Page 13
3.5 - Page 2	3.5 - Page 2
3.6 - Page 3	3.6 - Page 3

Appendix B, Additional Conditions

REMOVE	INSERT
Appendix B – Page 1	Appendix B – Page 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 255 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By applications dated July 30 and October 19, 2007 (References 1 and 2, respectively), and supplemented by letters dated August 31 and December 12, 2007, and February 21, March 28, and April 4 and 10, 2008 (References 29, 3, 4, 5, 6, and 7, respectively), Omaha Public Power District (OPPD, the licensee) requested changes to the Technical Specifications (TS, Appendices A and B to Renewed Facility Operating License No. DPR-40) for the Fort Calhoun Station (FCS), Unit No. 1.

The supplemental letters dated August 31 and December 12, 2007, and February 21, March 28, and April 4 and 10, 2008, provided additional information that clarified the applications, did not expand the scope of the applications as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 28, 2007 (72 FR 49581), and January 29, 2008 (73 FR 5227).

The NRC issued requests for additional information (RAIs) by a letter dated January 18, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080100010), and by e-mail dated March 27, 2008 (ADAMS Accession Nos. ML080880009 and ML080880010). The NRC staff held public meetings with representatives of FCS staff on May 11, September 21, and December 12, 2006, to discuss the licensee's approach to using a water management strategy to meet Generic Safety Issue (GSI) No. 191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance" (References 8, 9, and 10, respectively). The NRC staff also held discussions with Mr. Thomas Matthews and others of the FCS staff and its contractors on February 4, March 6, 7, 11, 17, and 27, and April 1 and 3, 2008, to clarify mutual understanding of the NRC requests for additional information and associated licensee responses.

The proposed amendment would revise TS Limiting Condition for Operation (LCO) 2.4, "Containment Cooling," LCO 2.14, "Engineered Safety Features System Initiation Instrumentation Settings," and LCO 2.15, "Instrumentation and Control Systems"; TS Surveillance Requirements (SR) 3.1, "Instrumentation and Control," SR 3.5(4), "Containment Isolation Valves Leak Rate Tests (Type C Tests)," and SR 3.6(3), "Containment Recirculating Air

Cooling and Filtering System"; and associated TS Basis documents and Updated Safety Analysis Report (USAR) Appendix G sections to modify the containment spray system actuation logic to preclude automatic start of the containment spray pumps for a loss-of-coolant accident (LOCA). The amendment would also revise TS SR 3.6(3)a. to delete SR for testing of the containment air cooling and filtering system (CACFS) emergency mode dampers and replace it with a surveillance to verify that the dampers are in the accident positions in all operating plant modes and deletes the requirement in TS SR 3.6(3)b. to remotely operate dampers.

The license amendment supports the licensee's proposed resolution of GSI-191 and response to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors." The proposed amendment changes containment pressure control during a LOCA from the containment spray system to the containment air coolers and mitigates activity release by the use of containment high-efficiency particulate air (HEPA) filters. This change is intended to increase the amount of water available to be delivered to the core during the injection phase of the postulated design-basis LOCA and reduce the amount of debris carried to the containment sump strainers during the accident.

The proposed modification to the containment spray actuation logic relies on the CACFS to maintain containment pressure below the containment design pressure and temperature following a LOCA. This function is currently performed by the containment spray system with no credit for the containment fan coolers. (Credit is taken for the fan coolers as well as the containment spray system for the postulated main steam line break (MSLB) accident.) In addition, the proposed amendment revises the FCS design basis to mitigate the control room and offsite doses following a LOCA by use of the CACFS. This function is also currently performed by the containment spray system.

The purpose of the proposed change to the containment spray actuation logic is to increase the amount of water available to be delivered to the core during the injection phase of the LOCA and to reduce the amount of debris carried to the containment sump strainers during the recirculation phase of the LOCA by eliminating the use of the containment spray system. This approach was previously discussed with the NRC staff in public meetings on May 11, September 21, and December 12, 2006.

The proposed changes involve modification of the containment spray actuation logic to prevent automatic containment spray pump start during a LOCA. The logic change adds the steam generator low signal (SGLS) contact in series with the containment spray actuation signal (CSAS) contact in the containment spray pump sequencer circuit. The NRC evaluation of the containment spray actuation logic and CACFS dampers is provided in Section 2.0 of this safety evaluation. The NRC evaluation of instrumentation aspects are provided in Section 3.0 of this safety evaluation. The NRC evaluation of the capability of the CACFS to mitigate control room and offsite dose is provided in Section 4.0 of this safety evaluation.

2.0 CONTAINMENT SPRAY ACTUATION LOGIC AND CACFS DAMPERS

2.1 System Description and Analysis

As provided in the FCS letter dated July 30, 2007, the licensee relies on containment safety analyses to demonstrate that all acceptance criteria continue to be met for the design-basis

LOCA with no credit for the containment spray system. The analyses use the GOTHIC 7.2a computer code described in AREVA topical report BAW-10252(P), Revision 0, “Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC” (Reference 11). The methodology in BAW-10252(P) was approved by the NRC for large dry containments such as the FCS containment (Reference 12). The licensee developed a FCS containment model using GOTHIC 7.2a, as noted in the FCS letter dated January 27, 2003 (Reference 13). This model was also approved by the NRC in a letter dated November 5, 2003 (Reference 14). The licensee’s GOTHIC model will continue to be used for calculating the containment response to the design-basis MSLB accident and for equipment qualification.

The licensee determined that the LOCA containment vapor temperature response without containment spray is “overly conservative” using the current AREVA containment analysis methodology. Therefore, the licensee proposes a revised methodology for transitioning from the short-term portion of the analysis (during which the RELAP5 computer code (Reference 15) is used to calculate the mass and energy release to the containment) to the long-term portions of the LOCA analysis (during which the mass and energy release to the containment is calculated using GOTHIC). The licensee has concluded that NRC approval of the methodology changes is required prior to implementation of the results of these analyses. The result of the staff review of these methodology changes is included in this safety evaluation report input.

The FCS containment spray system is described in the Updated Safety Analysis Report (USAR) Section 6.3. The containment spray system consists of three spray pumps and two (shutdown cooling) heat exchangers located outside containment and associated piping, valves, and instrumentation. The spray pumps initially take suction from the Safety Injection and Refueling Water Tank (SIRWT). The proposed amendment will result in containment spray actuation only for the MSLB. For this event, the licensee states that the SIRWT inventory is adequate and there is no need for recirculation from the containment sump. With the current licensing basis, during a LOCA, excess containment spray pumps may be secured indefinitely to conserve SIRWT inventory so that only one pump and one heat exchanger remain in service provided the five conditions listed in Section 6.3.4 of the USAR are met. With approval of the proposed license amendment, this option is no longer necessary.

The FCS containment air cooling and filtering (CACF) system is described in USAR Section 6.4. Currently, the operation of the containment fan coolers is credited in the mitigation of the MSLB but not the LOCA although it is automatically started upon indication of a LOCA. The CACF consists of four air handling units, each with a fan and heat exchanger. There is a common plenum discharge. There are two types of units. Two units have a filtering capacity and the other two have no filtering capacity. Each air cooling and filtering unit, according to the USAR, was designed for an inlet air flow of 110,000 cubic feet per minute (cfm) when cooling the containment atmosphere at design-basis accident (DBA) conditions of 60 pounds per square inch gauge (psig), 288 degrees Fahrenheit (°F), and 100 percent relative humidity. The air cooling units are similar in design to the cooling and filtering units but do not include mist eliminators and HEPA and charcoal filters. Each unit was designed for an inlet air flow of 66,000 cfm.

The face and bypass dampers on the CACF units direct air through the HEPA and charcoal filters after a DBA. The dampers are of multi-blade construction with galvanized steel blades, neoprene seals, and fail-safe air piston operators that work against a spring. There are 32 filtered inlet (face) damper assemblies and 2 bypass inlet (bypass) damper assemblies on each CACF unit. A solenoid valve supplies control air pressure to the piston operators located on each damper assembly. The solenoid valve is designed such that an electrical failure removes control air pressure. As a result, on loss of control air pressure or control signal, the dampers assume their post-accident filtration position; the face dampers open and the bypass dampers close. One face damper operator and one bypass damper operator on each CACF unit are equipped with limit switches to provide indication to the control room operators, via indicating lights, the Qualified Safety Parameter Display System, and the emergency response facilities plant computer as to whether these dampers are in their normal or accident positions.

2.2 Regulatory Evaluation

In Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Technical Specifications," the NRC established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) SRs, (4) design features, and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS. As stated in 10 CFR 50.36(d)(2)(i), the "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility." The regulations in 10 CFR 50.36(d)(3) state that "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components will be maintained within safety limits, and that the limiting conditions for operation will be met."

In a memorandum dated September 18, 1992, the Commission approved the staff proposal in SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program," not to apply 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," to plants with construction permits prior to May 21, 1971 (Reference 16). FCS was licensed for construction prior to May 21, 1971, and at that time committed to the draft General Design Criteria (GDC). The draft GDC, which are similar to Appendix A, General Design Criteria for Nuclear Power Plants in 10 CFR Part 50, are contained in Appendix G of the FCS USAR.

In its license amendment requests (LARs) dated July 30 and October 19, 2007, the licensee appropriately identified the applicable regulations in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50.67, "Accident source term." The licensee also identified preliminary design criteria including:

- FCS Design Criterion 10, "Containment." Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

- FCS Design Criterion 15, "Engineered Safety Features Protection System." Protection systems shall be provided for sensing accident situations and initiating operation of necessary engineered safety features."

- FCS Design Criterion 37, "Engineered Safety Features Basis for Design." Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

- FCS Design Criterion 38, “Reliability and Testability of Engineered Safety Features.” All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by systems, including engineered safety features, will be influenced by the known and demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

- FCS Design Criterion 41, “Engineered Safety Features Performance Capability.” Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming the failure of a single active component.

- FCS Design Criterion 42, “Engineered Safety Features Components Capability.” Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

- FCS Design Criterion 43, “Accident Aggravation Prevention.” Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

- FCS Design Criterion 49, “Containment Design Basis.” The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of the emergency core cooling systems.

- FCS Design Criterion 52, “Containment Heat Removal Systems.” Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided. Even though the proposed change would result in the containment spray system not starting automatically for a LOCA event, the containment spray system remains available to the operator for manual action. Therefore, the staff considers that FCS Design Criterion 52 remains satisfied for the LOCA.

- FCS Design Criterion 58, "Inspection of Containment Pressure-Reducing Systems." Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps.

- FCS Design Criterion 59, "Testing of Containment Pressure-Reducing Systems." The containment pressure reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

- FCS Design Criterion 61, "Testing of Operational Sequence of Containment Pressure-Reducing Systems." A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

- FCS Design Criterion 62, "Inspection of Air Clean-up Systems." Design provisions shall be made to facilitate physical inspection of all critical parts of the containment air cleanup systems, such as ducts, filters, fans, and dampers.

- FCS Design Criterion 63, "Testing of Air Cleanup Systems Components." Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

- FCS Design Criterion 64, "Testing of Air Cleanup Systems." A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits. The licensee has proposed modifications to the method of satisfying this criterion.

2.3 Technical Evaluation

The licensee is requesting changes to the FCS TS to modify the containment spray system actuation logic to preclude automatic start of the containment spray pumps for a LOCA. The proposed modification changes the containment pressure control during a LOCA from the containment spray system to the CACFS. The control room and offsite doses would also be controlled by the use of containment HEPA filters included in the containment air coolers.

2.3.1 Revisions to TS LCOs 2.4(1)b. and (2)a.

Credit for operation of the containment spray system to control containment pressure following an MSLB is unchanged by this amendment. Therefore, the LAR proposes no changes to the containment spray system operability requirements of FCS TS 2.4, "Containment Cooling," or the SRs of TS 3.6(2). However, the licensee is proposing changes to the minimum requirements (1)b. and (2)a. of TS 2.4. TS 2.4 (1)a.i. lists equipment powered by Diesel Generator (DG)-1. TS LCO 2.4(1)a.ii. lists equipment powered by DG-2.

Current TS LCO 2.4(1)b. states:

> During power operation one of the components listed in (1)a.i. and ii. may be inoperable.

The proposed LCO TS 2.4(1)b. would state:

> During power operation one of the components listed in (1)a.i. or ii. may be inoperable.

The licensee states the reason for this proposed change is:

> The Technical Specification (TS) 2.4(1)b. limiting conditions for operability (LCO) statement is being revised to clarify that it is applicable to one component in (1)a.i or one component in (1)a.ii, for a total of “one” component.

TS 2.4(2)a. currently allows two items from (1)a.i and ii to be inoperable with some exceptions. One exception is that at least one containment spray pump must remain operable. In addition, the licensee is modifying LCO 2.4(2)a. to specify that both trains of fan coolers cannot be simultaneously inoperable; that is, one train of CACFS must be operable during power operations, either VA-3A and VA-7C , or VA-3B and VA-7D. Conforming changes are also being made to TS LCO 2.4 references.

The NRC staff concludes that the proposed changes are acceptable.

2.3.2 Revisions to TS SR 3.5(4)

TS SR 3.5(4), “Containment Isolation Valves Leak Rate Tests (Type C Tests),” is revised to add valves for penetrations M-86 and M-89 to the list of valves requiring examination and testing. The licensee states that these valves were previously exempted from testing and inspection due to their function during a LOCA. (They were required to remain open to pass the containment spray flow to the containment spray spargers.)

The staff notes that the proposed change is consistent with the safety analyses and is, therefore, acceptable.

2.3.3 Revision to TS SR 3.6(3)

Current TS SR 3.6(3)a. states:

> Emergency mode damper, automatic valve fan, and fusible link automatic damper operation will be checked for operability during each refueling outage.

The proposed TS SR 3.6(3)a. would state:

> The emergency dampers will be verified to be in their accident positions and the automatic valve, fan, and fusible link automatic damper operation will be checked for operability on a refueling surveillance interval.

TS 3.6, “Safety Injection and Containment Cooling Systems Tests,” is revised to include measurement of the pressure drop across the containment HEPA filters on a refueling surveillance interval to ensure the filters are capable of removing radioactive particulates during a LOCA.

Although the calculated peak containment pressure has increased, the value of P_a specified in the FCS TS 5.19, “Containment Leakage Rate Testing Program,” does not have to be revised since the TS value of P_a is the design pressure, 60 psig.

With the face and bypass dampers aligned in their accident positions permanently, these dampers become passive components and the CACF system operates only in filtered air mode. Therefore, OPPD proposes to delete SRs that currently require testing (TS 3.6(3)a) and exercising (TS 3.6(3)b) the face and bypass dampers. OPPD proposes to revise TS 3.6(3)a to verify that the face and bypass dampers are in their accident positions each refueling outage. OPPD also proposes to revise the response to Criterion 64 to Appendix G of the FCS USAR to show that these dampers are maintained in their accident positions and delete the statement indicating that the filters are normally bypassed.

Currently, TS 3.6(3)a requires these dampers to be checked for operability each refueling outage while TS 3.6(3)b requires them to be exercised at intervals not to exceed 3 months. The modification will permanently remove control air to the piston operators causing the valve springs to set the dampers to their post-accident filtration positions. In addition, the modification will remove the ventilation isolation actuation signal currently provided to these dampers. Since the dampers will become passive components not required to change position, the requirements of TS 3.6(3)a and TS 3.6(3)b to test and exercise them, respectively, can be eliminated. TS 3.6(3)a will be revised to require verification that the dampers are in their accident position each refueling outage.

In the FCS letter dated April 4, 2008, the licensee stated that fusible link dampers are a separate set of dampers from the emergency mode dampers and that there is no direct relationship between the fusible link dampers and the emergency mode damper. The modification will permanently remove control air to the piston operators causing the valve springs to set the emergency mode dampers into their post-accident filtration position. Conforming changes to TS SR 3.6(3) include:

Current TS SR 3.6(3)b. states:

> Each fan and remotely operated damper required to function during accident conditions will be exercised at intervals not to exceed three months.

The proposed TS SR 3.6(3)b. would state:

> Each fan required to function during accident conditions will be exercised at intervals not to exceed three months.

Current TS SR 3.6(3)d. states:

> A visual examination of the HEPA and charcoal filters will be made during each refueling outage to insure that leak paths do not exist.

The proposed TS SR 3.6(3)d. would state:

> A visual examination of the HEPA and charcoal filters will be made on a refueling surveillance interval to ensure that leak paths do not exist.

Current TS SR 3.6(3)e. states:

> Measurement of pressure drop across the combined HEPA and charcoal adsorber banks shall be performed at least once per plant operating cycle to verify a pressure drop of less than 6 inches of water at system design flow.

The proposed TS SR 3.6(3)e. would state:

> Measurement of pressure drop across the HEPA filter bank shall be performed on a refueling surveillance interval to verify a pressure drop of less than 2 inches of water at system design flow. Measurement of pressure drop across the combined HEPA and charcoal adsorber banks shall be performed on a refueling surveillance interval to verify a pressure drop of less than 6 inches of water at system design flow.

Current TS SR 3.6(3)f. states:

> Fans shall be shown to operate +/-10% design flow during each refueling outage.

The proposed TS SR 3.6(3)f. would state:

> Fans shall be shown to operate within +/-10% design flow on a refueling surveillance interval.

In the FCS letter dated April 10, 2008, the licensee stated that the revised requirement in TS SR 3.6(3)e. is based on the Maintenance and Instruction Manual for American Air Filters (AAF)-designed Nuclear Ventilation and Cooling equipment which states that the filter should be replaced when the pressure drop across the bank is between 2 inches of water gauge (WG) and 3 inches WG. The licensee's response also describes how the pressure drop will be measured.

In the FCS letter dated April 10, 2008, the licensee agreed to the following "Additional Conditions" to Appendix B to Renewed Facility Operating License:

> (2) This license shall be deemed to contain the following specified conditions and specified actions:
>
> > (a) During the 2008 RFO [refueling outage], replace the CACF [containment air cleaning and filtering] unit HEPA [high-efficiency particulate air] filters not previously replaced during 2006;
> >
> > (b) During the 2008 RFO, ensure all CACF unit HEPA filters meet the 2" wc [water column] differential pressure limitation;

(c) Implement and maintain the newly created HEPA filter replacement criteria in Procedure PE-RR-VA-0209 unless superseded by other criteria via the license amendment process; and

(d) Submit a license amendment request (LAR) by October 31, 2008, that will add the HEPA filter testing and replacement criteria to the FCS [Fort Calhoun Station] Technical Specifications.

(3) This license shall be deemed to contain the following specified conditions and specified actions:

(a) During the 2008 RFO, perform the surveillance operability testing of the containment cooling unit relief ports; and

(b) Submit a LAR by October 31, 2008, that will add surveillance operability and testing of the containment cooling unit relief ports to the FCS Technical Specifications.

Note: An existing license condition for Amendment No. 181 will be designated as license condition (1).

The NRC staff believes these changes are necessary and consistent with the licensee's methodology for calculating post-LOCA HEPA filter loading and the effect on containment fan cooler air flow. Based on the above, the staff concludes that revised TS SR 3.6(3)a., 3.6(3)b., 3.6(3)d., 3.6(3)e., and 3.6(3)f. are acceptable.

2.3.4 FCS Emergency Core Cooling System (ECCS) and Containment Cooling Systems

The FCS ECCS safety-injection system is described in USAR Section 6.2. The safety-injection system consists of a low-pressure safety-injection (LPSI) system, a high-pressure safety-injection (HPSI) system, and four safety-injection tanks, each injecting into a different cold leg. The LPSI system consists of two redundant trains. Each train contains one centrifugal pump. The HPSI system contains three centrifugal pumps.

The FCS USAR states that the requirements during a design-basis large-break LOCA are met with the assumption of three of the four safety-injection tanks delivering borated water to the core and with one HPSI pump delivering approximately 75 percent of its rated flow to the core and one LPSI pump delivering approximately 75 percent of its rated flow to the core during the injection phase. For the recirculation phase, one HPSI pump has sufficient capacity with 25 percent spillage to maintain the core water level at the start of recirculation and during long-term cooling.

The containment spray system is described in Section 6.3 of the FCS USAR. It consists of three centrifugal pumps and two heat exchangers (shutdown cooling heat exchangers). These heat exchangers are cooled by the component cooling water system. The USAR states that one spray pump meets the capacity requirements in the event of a DBA.

For long-term cooling following a LOCA and depletion of the SIRWT, a recirculation actuation signal (RAS) shuts down the LPSI pumps. The HPSI pumps take suction from the containment sump. The USAR states that at the discretion of the operator, cooled water from the containment spray system may be diverted to the suction of the HPSI pumps. The USAR states that this is the preferred method of operation but is not necessary to meet core cooling requirements. Since the containment spray pumps would not be in operation after converting to the new containment spray actuation logic, this mode would no longer be available. In response to an NRC staff RAI, the licensee stated that the system alignment for this mode of operation is currently described in the Emergency Operating Procedures/Abnormal Operating Procedures (EOP/AOP) attachments. The steps in the attachments require that valves be configured so that containment spray flow is inhibited prior to initiating cooled safety-injection flow. The licensee states that this procedural guidance will be maintained under the proposed change.

The licensee proposes to rely on the CACFS as the sole active system to cool the containment atmosphere and to limit the leakage of airborne activity from the containment following a LOCA. Section 6.4 of the FCS USAR states that the system was designed to maintain the containment pressure below its design pressure of 60 psig by removing heat from moisture-saturated air at a design pressure of 60 psig and a temperature of 288 °F.

The CACFS consists of four air handling units, each with its own fan, and a common plenum discharge system. Each train of the CACFS is made up of one air filtering and cooling unit (CAFC) and one air cooling unit (CAC). The air cooling and filtering units comprise, in flow sequence, inlet face dampers, moisture separators, mist eliminators, HEPA filters, charcoal filters, and cooling coils. The air cooling units are similar in design to the air cooling and filtering units but do not include mist eliminators, face and bypass dampers, HEPA filters, and charcoal filters.

The FCS USAR describes the HEPA filters in the containment air cooling and filtering units as arranged in filter banks which consist of individual cells 24-inches-wide by 24-inches-high by 12-inches-deep (24 in $\times$ 24 in $\times$ 12 in), supported in a holding frame. The filter medium is pleated fiberglass separated by aluminum spacers. The USAR states that these filters are "suitable for the DBA environmental operating conditions" (Reference 17).

The current FCS licensing basis credits the containment air coolers in determining the peak containment pressure for an MSLB, but not for a LOCA. The licensee proposes to revise the licensing basis to also credit the containment air coolers for a LOCA (in place of the containment spray system).

The LOCA analyses assume 100 x 10^6 British thermal units per hour (BTU/hr) from one train of fan coolers for the first 30 minutes of the accident and 150 x 10^6 BTU/hr after 30 minutes when the analysis assumes a second component cooling water (CCW) pump is started to remove more heat from the fan coolers. The licensee states that the design capacity of one train of fan coolers is 210 x 10^6 BTU/hr, 140 x 10^6 BTU/hr for the CAFC and 70 x 10^6 BTU/hr for the CAC. The licensee verifies that the fan cooler air flows meet or exceed the acceptance criteria every refueling outage. The licensee states that the air sides of the coolers are visually inspected every outage and cleaned as needed. In addition, as noted in the FCS letter dated February 21, 2008, a preventive maintenance procedure is performed each outage to flush the coils and to verify that post-DBA flows through the cooling coils can be achieved. In response to an NRC staff RAI concerning verification of these heat removal rates, the licensee stated that,

> The assumed heat removal rates were not specifically benchmarked, however, the CCW flow rates assumed in the CCW system model were benchmarked against plant operating data with various system alignments. Benchmarking of the CCW system confirmed that the CCW system model provides accurate results for various systems alignments such as LOCAs and MSLB [main steam line break] conditions.

The staff requested that the licensee address the effect on the emergency DGs of starting a second CCW pump. In its letter dated March 28, 2008, the licensee indicated that the second CCW pump is powered by DG-2 which has "excess capacity and can handle the additional load." In addition, the EOP contain caution statements warning the operators not to exceed DG power and current limits.

For dose calculations, the licensee assumes 50 percent filter efficiency. The justification for this efficiency is provided in the licensee's February 21, 2008, response to an NRC staff RAI. The 50 percent filter efficiency is based on a calculation that estimates the potential bypass leakage following a DBA with a factor of 2 applied.

The 50 percent filter efficiency is a highly conservative assumption used for the dose consequence analysis. The 50 percent filter efficiency is supported by a calculation that estimates that the maximum potential bypass leakage following a DBA does not exceed 25 percent (then a conservative factor of 2 is applied). In calculating the 25 percent bypass value, both the leakage around/through the HEPA filters at LOCA conditions, and the reduction of flow due to the increased head requirements across the fan were factored into the analysis. The calculation concludes that a filter efficiency greater than 50 percent can be achieved with an equivalent hole diameter of approximately 1.5 inches in each of the filter media or half of the gasket missing on all the filter elements in the HEPA filter, at filter differential pressure as high as 5.5" wc (see Reference 2, Attachment 4, Page 11 of the licensee's July 30, 2007, letter to the NRC). HEPA filter elements, dampers, pressure relief ports, and other associated components are visually inspected during each refueling outage. Therefore, any source of significant bypass would be identified and corrected prior to start-up from the outage.

The staff concluded that the licensee's approach to determining filter efficiency is reasonable and concluded that the proposed method is acceptable.

2.3.5 Calculation Methods

The licensee refers to the starting point for the analyses supporting the licensing amendment to eliminate automatic actuation of the containment spray system following a LOCA as the AOR. The AOR is described in Revision 11 of Section 14.16 of the FCS USAR. The AOR calculates the containment response to a LOCA using GOTHIC 7.0 (References 11 and 12). The mass and energy release into the containment from the postulated pipe break is calculated in the AOR with the NRC-approved AREVA computer code RELAP5/MOD2-B&W (Reference 18).

The licensee is proposing to use the containment analysis methods of the AREVA topical report BAW-10252-P-A (Reference 19). BAW-10252-P-A describes AREVA's approach to using the GOTHIC code for containment calculations. GOTHIC 7.1 was used to perform the sample calculations in this report. However, Section 2.0 of BAW-10252-P-A states that AREVA's containment analysis methods are not restricted to a specific GOTHIC computer code version, "but rather to the code options and modeling approach described in the topical report." The NRC staff found this approach acceptable in approving BAW-10252-P-A.

The previously approved FCS GOTHIC model used GOTHIC 7.0. The proposed application of GOTHIC to support the change in containment spray logic uses GOTHIC 7.2a. The NRC staff has reviewed the differences between GOTHIC 7.0 and GOTHIC 7.2a as described in the GOTHIC 7.2a documentation, and concludes that they do not affect the proposed use of GOTHIC.

After a certain point in the calculation of the containment response to a LOCA, called the transition, the mass and energy release calculation is performed using the GOTHIC computer code. Information must be transferred from RELAP5/MOD2-B&W to GOTHIC to affect this change in computer codes.

The use of RELAP5/MOD2-B&W for containment analysis is described in Section 5.1.2.3 of BAW-10252-P-A. Section 5.1.2.3 states that the initial and boundary conditions for the mass and energy release calculations are chosen to maximize the stored energy in the primary and secondary coolant systems and to maximize the removal of this energy to the containment. This is conservative for the calculation of peak containment pressure and temperature. Cases in which the containment pressure is to be minimized are discussed later in this safety evaluation. The sources of energy included in these mass and energy calculations are the same as those considered in LOCA analyses done to demonstrate compliance with the criteria of 10 CFR 50.46 using the required and acceptable features of 10 CFR Part 50 Appendix K. This is consistent with the guidance of Section 6.2.1.3.II.1 of the Standard Review Plan (SRP) (Reference 20). The use of RELAP5/MOD2-B&W includes the reduction in the decay heat uncertainty from 20 percent to 10 percent after 1000 seconds. Therefore, this treatment of decay heat uncertainty is included in the LOCA calculations supporting the FCS change in containment spray logic and remains acceptable. It is also consistent with the guidance of NRC Branch Technical Position ASB 9-2 (Reference 21).

The proposed FCS GOTHIC model uses a lumped parameter, single-control volume approximation which has been found acceptable for the analysis of the containment atmospheric response to a postulated high-energy fluid pipe break with a large-break area because the turbulence caused by the large jet momentum mixes the containment atmosphere. The steam and noncondensable components of the containment atmosphere can be considered to be homogeneously mixed and in thermal equilibrium with each other. In addition, containment spray provides large-scale mixing of the containment atmosphere (Reference 22).

The staff questioned the use of the lumped parameter, single-control volume, approximation for conditions after the blowdown with no containment spray in operation. In discussing the design bases of the CACFS, Section 6.4 of the FCS USAR (Revision 7) states that a design basis of this system is the ability to prevent accumulation of hydrogen pockets by maintaining a continuous flow of air throughout the containment. This flow of air throughout the containment can be considered adequate to support the lumped parameter approximation.

The GOTHIC conservation equations are solved for three primary fields. These are the steam/gas mixture, continuous liquid, and liquid drops. The steam/gas mixture is referred to as the vapor phase and consists of steam and any non-condensable gases (air, nitrogen released from the safety-injection tanks, and any hydrogen generated as a result of the LOCA).

As discussed in Section 2 of Attachment 6 to the licensee's July 30, 2007, letter (Reference 23), the licensee found that, with containment sprays operating, there is a small increase in containment vapor temperature after transition from RELAP5/MOD2-B&W to GOTHIC. However, without containment spray, the licensee found a marked increase in containment vapor temperature which the licensee judged to be non-mechanistic. The licensee corrected this non-mechanistic behavior with a change to the calculation method. This is explained and evaluated in more detail in a supplement to this safety evaluation report input which contains AREVA proprietary information (ADAMS Accession No. ML081120493). The main report can be found in ADAMS Accession No. ML081080321). A nonproprietary version of this supplement is available (ADAMS Accession No. ML081080358). The licensee terms the currently approved method of transition between methods of calculating the mass and energy release the "existing" method and the revised method, which more accurately models the containment vapor temperature response, is the "alternate" method.

The licensee also revised the timing of the transition. The licensee states that in the existing methodology, the transition is modeled at the end of the RELAP5 mass and energy release analysis, which occurs after core quench, typically between 500 to 1000 seconds. The licensee now proposes the option of the transition taking place at the time of the RAS. Delaying the transition provides a smoother, continuous, and therefore more realistic prediction of containment conditions.

The table below summarizes the licensee's proposal for determining the energy partition between the vapor and liquid phases (existing or alternate) and the timing of the transition (early or RAS).

Table
Applications of Transition Timing and Energy Distribution Models

	Transition (Early: approximately 500 to 1000 seconds/Late: RAS)	**Energy Dissipation (Existing/Alternate)**
LOCA Peak Containment Pressure	N/A. Time to peak pressure is too short. Do not require transition.	N/A. Time to peak pressure is too short.
LOCA Peak Containment Temperature	N/A. Time to peak temperature is too short. Do not require transition.	N/A. Time to peak temperature is too short.
LOCA Long-term (24-hour) Pressure	RAS	Existing/Alternate
LOCA Long-term Temperature (EQ)	RAS	Alternate
LOCA Sump Water Temperature	RAS	Existing/Alternate
Peak Component Cooling Water (CCW) and Raw Water (RW)Temperature	N/A. The analysis of the LOCA CCW and RW temperature response are short-term analyses that do not require transition from RELAP5/MOD2-B&W to GOTHIC.	N/A. The analysis of the LOCA CCW and RW temperature response are short-term analyses that do not require transition from RELAP5/MOD2-B&W to GOTHIC.

Extending the transition to RAS involves a change in the decay heat uncertainty at 1000 seconds. As discussed above, this is consistent with NRC guidance and previous reviews and is acceptable. Also, as discussed in Sections 3.3.1 and 3.3.3 of Attachment 6 to the licensee's July 30, 2007, letter, changes are required to several RELAP5 edit variables to support the alternate method of energy dissipation.

The licensee has revised the safety analyses affected by the proposed change. The analysis results are provided in Attachment 6 to the FCS letter dated July 30, 2007.

Table 1 of the licensee's February 21, 2008, letter lists the major differences between the new containment analysis in the licensee's July 30, 2007 letter, and the AOR given in Chapter 14.16 of the FCS USAR. In addition to those already discussed in this safety evaluation (change in the timing of the transition, the energy partition between the vapor and liquid phases, the change in the uncertainty of the value of the decay heat), the licensee has proposed other changes to the AOR.

Several of these changes are changes in calculation models. These changes include phase separation of the break flow in the containment atmosphere, revaporization fraction, and liquid vapor interface area. The separation of the break flow into liquid and vapor affects the containment pressure and temperature. The license uses the BAW-10252-P-A model which is conservative for determining a peak containment pressure. For containment pressure calculations in which the pressure is to be minimized, this approach would not be conservative. However, the effect is small and there is sufficient conservatism in the licensee's available net positive suction head (NPSH) calculations (see Section 2.3.8 of this safety evaluation). The revaporization fraction is only significant with superheated steam and is therefore not important for LOCA analyses. The liquid vapor interface for both the short-term and long-term analyses is determined using the methods of BAW-10252-P-A. The liquid vapor interface area is set to zero for short-term analyses only. In the FCS letter dated February 21, 2008, the licensee states that by not allowing heat and mass transfer between the vapor and liquid, the vapor region temperature remains slightly higher since the steam in the atmosphere is saturated and the pool is subcooled which produces a slightly higher pressure.

Another change is operator action at 30 minutes after the LOCA to start a second component cooling water pump. This is discussed later in Section 2.3.9 of this safety evaluation.

Based on the above, the NRC staff finds the licensee's calculation methods acceptable.

2.3.6 Short-term LOCA Analysis

The licensee considered the effect of eliminating the use of containment spray on short-term LOCA containment analyses. The purpose of these analyses is to demonstrate that the peak containment pressure and temperature remain below their respective design values of 60 psig and 305 °F (USAR Section 5.1, Reference 24). Also, these analyses served to identify limiting cases for the long-term containment analyses without containment spray.

The conservative assumptions used in the short-term analyses are listed in Section 14.16 (Revision 11) of the FCS USAR. In addition to these the licensee's February 21, 2008, letter, in response to an NRC staff RAI, listed additional conservative assumptions that include:

- A single train of the CACFS is credited in addition to any single failures assumed in the mass and energy release calculations.

- The containment air cooling unit heat removal rates are set conservatively low.

Table 1 of Attachment 6 of the licensee's July 30, 2007, letter provides the results of the licensee's calculations. The peak pressure is determined to be 54.47 psig at 13.22 seconds. This limiting case is a double-ended hot-leg break with a discharge coefficient of 0.6, loss of offsite power, a single failure of emergency DG-1. The mass and energy release was determined using a minimum containment backpressure. A lower containment back pressure results in a lower reflood rate and consequently a lower release of energy to the containment. However, for the peak pressure case, the peak pressure and temperature occur before reflood commences and, therefore, the mass and energy release has the dominant effect on peak containment pressure. The licensee points out that for the cases listed in Table 1 of Attachment 6, the peak pressures are within 2 pounds per square inch (psi) for the entire spectrum of cases analyzed.

The peak containment vapor temperature is determined to be 277.4 °F at 13.04 seconds from the same hot-leg break case.

The licensee also analyzed cold-leg breaks. These resulted in lower peak pressure and temperature. The peak containment pressure for the pump suction line break is 53.43 psig at 165.88 seconds. The predicted peak containment temperature is 275.8 °F. These results were obtained with a low containment backpressure.

10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," defines the peak calculated containment pressure as a result of a LOCA as P_a. The FCS TS specify a value of P_a equal to the containment design pressure of 60 psig. Therefore, no change in the TS value of P_a is required as a result of this analysis.

Based on the above, the NRC staff concludes that the short-term calculations are acceptable to support the proposed change in containment spray actuation logic since they are done with acceptable methods, conservative assumptions and the calculated pressure and temperature values are less than the design pressure and temperature values.

2.3.7 Long-term LOCA Analysis

The licensee also analyzed the long-term peak containment pressure and vapor temperature. For the long-term period, the transition is made from RELAP5/MOD2-B&W to GOTHIC for calculating the mass and energy release to the containment. The transition is either at the time of core quench (500 to 1000 seconds) or at the time of the RAS. The long-term mass and energy release rates are based on conditions at the time of the recirculation actuation signal (RAS) obtained from the RELAP5 calculations. The associated rates of energy released at the time are maintained constant until all energy is depleted from the reactor coolant system. This releases the energy earlier than a more realistic but less conservative calculation.

The long-term sources of energy are the stored energy in the reactor coolant system and the steam generator metal as well as the decay heat.

The licensee also performed analyses to ensure that the guidance of SRP Section 6.2.1.1.A (Reference 25) is satisfied in that the containment pressure decreases to less than half the peak pressure within 24 hours. This SRP criterion is considered to satisfy the requirement of 10 CFR Part 50, Appendix A, General Design Criterion 38, "Containment Heat Removal," to rapidly reduce the containment pressure and temperature following a LOCA. Although neither the regulatory guidance of the SRP nor the requirement of GDC 38 applies to FCS, the licensee demonstrated that these criteria are met. The licensee analyzed the limiting hot-leg and cold-leg cases without containment spray. The peak containment pressures at 24 hours for the hot-leg and cold-leg breaks are 10.61 psig and 9.73 psig, respectively. The peak containment pressures for these two breaks, as stated above, are 54.47 psig and 53.43 psig, respectively. Therefore, GDC 38 is satisfied.

The staff questioned the licensee's assumption that the limiting long-term cases with spray would result in the limiting long-term results without spray. In the FCS letter dated February 21, 2008, the licensee indicated that the post-peak pressure trends for the cases analyzed were comparable with and without sprays. The licensee illustrated this with calculations. These calculations are given in Figures 7 to 12 included in the licensee's February 21, 2008, letter. The staff finds this explanation acceptable.

The long-term LOCA calculations are acceptable to support the proposed change in containment spray actuation logic since they are done with acceptable methods, conservative assumptions, and the calculated pressure is less than half the peak calculated pressure after 24 hours.

2.3.8 Long-Term Sump Water Temperature Analysis and Net Positive Suction Head (NPSH)

The sump water temperature is an important input to the calculation of pump available net positive suction head (NPSHA). The only pumps taking suction from the sump when the containment spray pumps are not in operation are the HPSI pumps.

The licensee states that, in general, the NPSH requirements for ECCS equipment taking suction from the containment sump are less without containment spray than with containment spray. This is due to the lower decay heat because the RAS occurs later in the postulated accident.

FCS currently credits containment accident pressure in determining NPSHA for the HPSI pumps. The NRC approved credit for containment accident pressure based on the licensee's response (Reference 27) to NRC Generic Letter 97-04 (Reference 26).

In performing these analyses it is conservative to maximize the sump water temperature and minimize the containment pressure.

The licensee uses the following terminology in assessing the NPSH margin of the ECCS and containment spray pumps.

Subcooling Head (ft) = $[P_{atmosphere} - P_{saturation}(T_{sump})] / \rho$

Overpressure Head (ft) = $[P_{containment} - P_{atmosphere}] / \rho$

Total Available Head (ft) = Subcooling Head + Overpressure Head,

where

$P_{atmosphere}$ is 2044.8 lbf/ft^2 or 14.2 psia

$P_{saturation}(T_{sump})$ is the saturation pressure at the sump water temperature, lbf/ft^2

$P_{containment}$ is the containment accident pressure, lbf/ft^2, and

ρ is the weight density of water at the sump water temperature, lbf/ft^3.

The licensee re-analyzed the limiting hot-leg and cold-leg long-term sump water temperature cases with containment spray assuming that containment spray was not operating. The licensee used the RAS as the transition time and the stored energy dissipation rates using the existing method. In the FCS letter dated February 21, 2008, the licensee stated that, in general, either the existing or alternate energy dissipation methods could be used “given the expectation that either method would produce comparable results.” Figures 11 and 14 of Attachment 6 to the licensee’s July 30, 2007, letter show the comparison of the sump liquid temperature calculated with the different options: early-existing, early-alternate, extended-existing, and extended-alternate. As would be expected, the figures show no difference in sump liquid temperature up to the time of RAS for the cases with transition at RAS (extended).

The post-RAS sump liquid temperatures are very close using the existing and the alternate energy dissipation methods for the limiting hot-leg break while for the limiting cold-leg break, the alternate method gives an approximately 3 °F higher temperature at the post-RAS time of maximum sump water temperature which would result in a slightly lower NPSH margin. In Figure 6 of the licensee’s February 21, 2008, letter to the NRC, the licensee has demonstrated significant margin between the containment pressure available and the pressure credited in determining NPSHA. The licensee’s calculations show a maximum NPSH deficit of 4.15 feet at 1.7 hours after the RAS. The total time that credit for containment accident pressure is required is approximately 9 hours after the RAS. This is for a cold-leg break case with maximum ECCS (3 HPSI pumps post-RAS). The calculated overpressure head at the time of the maximum NPSH deficit is approximately 25 feet. Therefore, the licensee’s proposal to use either the existing or alternate method of energy dissipation is acceptable.

The licensee adds 10 percent of the total stored energy directly to the sump liquid as an added conservatism for sump water temperature calculations. The remaining 90 percent is transferred to the containment vapor space and the reactor coolant system liquid according to the existing or alternate energy dissipation models. In the FCS letter dated February 21, 2008, the licensee identified the following other conservative assumptions used in the calculation of sump water temperature and minimum containment pressure:

- Nominal initial pressure was used.
- High initial humidity was used. This minimizes the amount of noncondensable gas which results in reducing the calculated containment accident pressure.
- The addition of the nitrogen gas to the containment atmosphere from the safety injection tanks after they discharge their water inventory is neglected.
- A multiplier of 1.2 was applied to the Uchida condensation heat transfer correlation. This increases the heat loss from the containment atmosphere to the internal containment structures and reduces the containment pressure.

The values of required NPSH for the HPSI pumps vary with the configuration of the operating pumps. The most conservative cases (lowest margin) were used in the development of the curves. In the FCS letter dated March 28, 2008, the licensee indicated that it verified that no adjustment is made to the pump-required NPSH due to the temperature of the sump water being significantly higher than the temperature at which the required NPSH value was measured. This satisfies Position 1.3.1.5 of NRC RG 1.82, Revision 3, which states that no adjustment to the required NPSH due to temperature should be made, and is conservative and acceptable (Reference 28).

The staff finds the determination of sump water temperature for available NPSH calculations to be acceptable since the calculation is done with acceptable methods and a sufficient degree of conservatism.

2.3.9 Component Cooling Water (CCW) and Raw Water (RW) Peak Temperature Analysis

The peak CCW temperature and the limit on the RW system maximum temperature are a function of the containment vapor temperature and humidity as these factors affect the heat-removal capability of the fan coolers.

Since the shutdown cooling heat exchangers are not in operation following a LOCA without containment spray, the heat load on the CCW system is different than with spray operation. The licensee restricted the evaluation of the CCW heat load to the pre-RAS impact on the CCW system following a LOCA with containment air coolers as the only active means of containment energy removal. The post-RAS peak in containment vapor temperature is lower than the pre-RAS peak due to the fact that the energy transport to the containment is lower from lower break flows and lower decay heat, with the manual start of a second CCW pump. The limiting hot-leg and cold-leg break cases from the CCW/RW analysis with containment spray were run without containment spray. The peak CCW temperatures at the containment air cooler secondary inlet were 152.6 °F for the limiting hot-leg break and 156.4 °F for the limiting cold-leg pump suction break. The raw water system inlet temperature remained at 90 °F.

The licensee states that the MSLB, for which the containment spray system is still credited, remains bounding.

2.3.10 Equipment Qualification

In response to an NRC staff RAI, the licensee addressed the qualification of safety-related equipment within the containment. The licensee states,

> Current equipment qualification is based on a bounding curve enveloping the analysis of record. The results of the long-term containment response shown in FC07247 [Attachment 6 to the licensee's July 30, 2007, letter] Figures 15 and 16 demonstrate that the containment vapor temperature is lower than the AOR in all areas of concern, including the short-term and the post-RAS peak. Therefore, elimination of the automatic containment spray actuation following a LOCA has no effect on the existing equipment qualification.

Based on the results of Figures 15 and 16 of Attachment 6 to the licensee's July 30, 2007, letter, the staff concurs. Therefore, the requirements of 10 CFR 50.49 for equipment qualification remain satisfied with respect to containment temperature and pressure.

2.3.11 Evaluation of Capability of CACFS for Containment Cooling and Dose

In response to the NRC staff's RAI, by letter dated April 10, 2008, the licensee provided the following methodology for calculating post-LOCA HEPA filter loading and effect on containment fan cooler air flow:

> The containment air cooling and filtering (CACF) units are safety-related engineered safeguards systems. The CACF units have been credited with the mitigation of post-LOCA radiation dose effects in containment since the unit was licensed in 1973, as a part of a system fully redundant in function to the containment spray system until 2001 (approval of Alternate Source Term), when fission product removal was conservatively credited only by the containment spray system. However, accumulation of fission products in the filters was acknowledged for purposes of estimating direct shine contribution to the control room in the Alternate Source Term calculations.
>
> The original design analysis for the HEPA filters did not determine a debris accumulation vs. pressure drop analysis to predict how much debris would be generated or would be collected by the HEPA filters. Instead, the HEPA filters were designed, fabricated and subjected to the quality assurance inspection tests of Atomic Energy Commission (AEC) Health and Safety Information Issue No. 212, dated June 25, 1965, "Minimal Specifications for the Fire Resistant High Efficiency Filter Unit," which is obsolete and was replaced by ASME [American Society of Mechanical Engineers] N509, "Nuclear Power Plant Air Cleaning Units and Components." Per Regulatory Guide (RG) 1.52, ASME N509 is the acceptable design and testing standard. Testing of the HEPA filters consisted of verification of the following attributes:
>
> - Design air flow test can be achieved with less than 1" water column (wc) pressure drop.
>
> - The filter element was required to pass a National Bureau of Standards (NBS) dust capacity test with 4 lbs. of dust loading without exceeding 2" wc pressure drop.

- The ultimate strength of the filter was verified by maintaining a 6" wc pressure drop across the dust loaded filter for 10 minutes.

- After passing all previous tests, the filter was required to pass a dioctyl phthalate (DOP) penetration test with no more than 0.03% penetration for 0.3 micron diameter particles.

OPPD has recently performed a HEPA filter loading engineering evaluation, which models the accumulation of non-radioactive debris from a LOCA event and an analysis of the fan and filter performance post-LOCA accounting for all debris accumulated on the filters.

The HEPA filter loading study was based upon the existing Generic Letter (GL) 2004-02 debris generation calculation for Fort Calhoun Station (FCS). Only those debris sources that were considered fines were calculated to be released to the upper containment atmosphere where the intakes to the CACF units are located. The amount instantaneously released to the upper containment is 69% of the total fines calculated in the debris generation calculation. Releasing all of these fines instantaneously to the upper containment atmosphere (i.e., above 1060' elevation) is considered extremely conservative, as the break location would be roughly near the 1013' elevation.

The total amount of entrained debris on the HEPA filters will depend on the RCS [reactor coolant system] break location and the "jet time," which is the time the high-velocity jet flows into the containment building before stopping. For this event, 440 pounds mass (lbm) of debris would be entrained. Fully saturated debris-laden air enters the CACF units at 288 °F and 60 psig. To maximize debris loading on the HEPA filters, only one of the two CACF units was assumed to run throughout the LOCA scenario.

The calculation was broken down into two parts for each type of debris: entrainment and settling. Entrainment is the time between the start of the jet-flow and the time the overall turbulent kinetic energy (TKE) drops below the entrainment TKE of the debris type. Settling is the time after debris begins to settle. During entrainment, the debris concentrations are conservatively treated as equally distributed throughout the upper containment building, and any debris removed by the HEPA filters results in a decrease in the overall debris concentration. During settling, only debris in the HEPA filters' zone of influence (ZOI) will be removed, while the rest will settle on surfaces in the containment building. The ZOI was calculated using a force balance on the debris and a momentum calculation on the HEPA filters. The force balance determines the debris terminal velocity, which is then used to determine the ZOI using a momentum balance.

Based on this conservative model, it was estimated that most of the TempMat© and Nukon© fibers (~93%) will settle to the containment building surfaces. This occurs because their respective terminal velocities are large, so they settle quickly. On the other hand, none of the CalSil or paint particulate is predicted to settle because their ZOIs are larger than the upper containment building volume.

This analysis shows that 116 lbm or 26.4% of the total debris will settle on surfaces in the containment building. All of the settled debris is TempMat and Nukon fibers along with

> dirt and dust. None of the smaller debris (CaiSil and paint) is predicted to settle, so the HEPA filters will collect all debris of these types. Therefore, it was conservatively estimated, with the entrainment and settling model, that 324 Ibm of non-radioactive material debris would be trapped by the HEPA filters. If consideration were given for addressing potential agglomeration from condensation in the containment building, the amount of debris that would settle out in containment would be significantly larger. However, 324 Ibm is conservatively assumed to be trapped by the HEPA filters.
>
> The total accumulation of debris on the HEPA filters post-LOCA is the combination of the debris loading at the beginning of the event plus the radioactive and non radioactive debris generated during the event that is transported to the filters.
>
> The analysis of the CACF unit performance assumed the combination of the revised surveillance test, IC-ST-VA-0013, "Verification Of Containment Air Cooling and Filtering Units Flow and Pressure Drop," limit (i.e., 2" wc for the HEPA filters) for loading plus the maximum accumulation of debris over the course of an operating cycle based on historical data as the starting point for the event, which is equivalent to 342.7 pounds. The fan and filter performance analysis adds the loading from the LOCA debris generation analysis described above and 52.3 pounds of radioactive debris from a conversion of the core inventory of Reference 2 [of the licensee's April 10, 2008, letter]. The FCS core inventory source term was provided in the [same] Reference 2, Table 4.1-1. In this reference, Table 7.1-1 identified the fuel activity release fractions per RG 1.183. The RG 1.183 release fractions were applied to the core equilibrium table values (in curies) to obtain the amount released into the containment atmosphere. This value is then converted into a mass value. The HEPA loading uses the full core inventory with RG 1.183 release fractions and no credits for any plateout, depositions, etc. The total accumulation equals 719 pounds of debris trapped by the HEPA filters, which includes the 324 pounds of LOCA generated non-radioactive debris, 52.3 pounds of radioactive debris, and 342.7 pounds of filter preload.
>
> The post-LOCA differential pressure was calculated to be 4.15" wc across the HEPA filters. This pressure drop was then included in the system performance analysis to determine the fan operating conditions at 24 hours post-LOCA and 30 days post-LOCA. The fan is not predicted to stall under post-LOCA conditions. The predicted minimum fan flow post-LOCA is approximately 118% of the stall air flow rate of 65,000 CFM for the duration of the event.

Finally, the licensee stated that,

> In summary, OPPD has performed a conservative analysis of the particulate accumulation on the HEPA filters and determined they will perform their design basis function post-LOCA, and the containment cooling function of the CACFs will be maintained.

The NRC staff held extensive discussions with the licensee and reviewed the supporting information described above concerning this methodology and found that USAR Section 6.4 "CACFS" and specifically that Section 6.4.1.2, "Design Criteria and Performance Objectives," states that the heat-removal capability of the system is based upon the DBA. The system was designed to remove heat from air saturated with moisture at a design pressure of 60 psig and temperature of 288 °F to maintain the containment below its design pressure of 60 psig. It is the NRC staff's judgment that this methodology is acceptable because with it the design basis of the system will continue to be met.

In the FCS letter dated October 19, 2007, the licensee requested a change in the way the licensee meets the "design criteria."

The licensee requested that the first paragraph demonstrating how Criterion 64 is met be changed from:

> This criterion is met. Two of the four containment fan coolers are equipped with the following components in order of flow sequence: louvers (spring loaded to fail open), mist eliminator, absolute filter, iodine filter, bypass louvers (designed to fail closed), cooling coil and fan.

to:

> This criterion is met. Two of the four containment fan coolers are equipped with the following components in order of flow sequence: open face dampers, mist eliminator, absolute filter, iodine filter, closed bypass dampers, cooling coil and fan.

The licensee also requested that the second paragraph demonstrating how Criterion 64 is met be changed from:

> The filters will normally be bypassed. As stated in "Criterion 62" the filters can be inspected during normal operation of the reactor. Installed instrumentation will indicate abnormal changes in filter pressure drops during routine operations. In place testing and out-of-containment testing of removed filter modules is also possible. Containment filters and the particulate air purge filters will be periodically evaluated to ensure they are operating within acceptable limits.

to:

> As stated in "Criterion 62" the filters can be inspected during normal operation of the reactor. Installed instrumentation will indicate abnormal changes in filter pressure drops during routine operations. In place testing and out-of-containment testing of removed filter modules is also possible. Containment filters and the particulate air purge filters will be periodically evaluated to ensure they are operating within acceptable limits.

The licensee stated that the proposed changes are necessary because of problems encountered during the 2006 RFO with the operation and positioning of these dampers while undergoing TS 3.6(3)a. surveillance testing. An operability justification (Safety Analysis for Operability 06-04) is in place documenting the failure to meet the SRs of TS 3.6(3)a. and 3.6(3)b. These dampers are being maintained in their accident positions; therefore, the CACFS which is operating in filtered air mode is operable. These dampers referred to in TS 3.6(3) as "emergency mode" or "remotely operated" are the CACFS face and bypass dampers. The accident positions for the face dampers are open and the bypass dampers are closed to allow air flow through the CACFS HEPA and charcoal filters.

Based on the above, the NRC staff finds these changes to how Criterion 64 is met are acceptable because it is consistent with the licensee's methodology for calculating post-LOCA HEPA filter loading and effect on containment fan cooler air flow.

2.3.12 Conclusion

The licensee has proposed mitigating the consequences of a design-basis LOCA without reliance on the containment spray system which was intended for that purpose. The licensee demonstrated that this was acceptable by demonstrating that the FCS containment acceptance criteria remain satisfied, i.e., the peak containment pressure and temperature remain below their design limits.

The licensee has also proposed to remove the FCS TS SR to exercise the emergency mode (remotely operated) dampers in the CACFS. The NRC staff finds this acceptable since the dampers will be maintained in the emergency position and therefore capable of responding to the DBA without changing position.

3.0 INSTRUMENTATION AND CONTROLS

By letter dated July 30, 2007 (Reference 1), OPPD requested an LAR for changes to FCS TS 2.14, "Engineered Safety Features System Initiation Instruments Setting Limits," and TS 3.1 "Instrumentation and Control Surveillance Requirements."

3.1 Regulatory Evaluation

In 10 CFR 50.36, the NRC established its regulatory requirements related to the content of TS. FCS was licensed for construction prior to May 21, 1971, and at that time committed to the draft General Design Criteria (GDC). The draft GDC, which are similar to Appendix A, General Design Criteria for Nuclear Power Plants in 10 CFR Part 50, are contained in Appendix G of the FCS USAR.

In its LAR, the licensee appropriately identified the applicable regulatory requirements that govern instrumentation and control systems at FCS including:

- FCS Design Criterion 12, "Instrumentation and Control Systems." Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

- FCS Design Criterion 15, "Engineered Safety Features Protection System." Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

3.2 Technical Evaluation

In response to GL 2004-02, the licensee submitted an amendment request to modify FCS TS LCO 2.14, "Engineered Safety Features System Initiation Instruments Settings," and TS SR 3.1, "Instrumentation and Control." In its letter dated February 21, 2008 (Reference 4), the licensee responded to an NRC RAI related to modification of containment spray system actuation logic. The licensee submitted a logic diagram clarifying the containment spray pump actuation logic.

The logic diagram indicated that the existing Steam Generator Low Pressure Signal (SGLS) contact is taken from the SGLS Matrix and input into the containment spray auto-start logic. The containment spray auto-start logic will now start on Pressurizer Pressure Low Signal (PPLS) and Containment Pressure High Signal (CPHS) coincident with SGLS.

The SGLS signal is taken from the SGLS logic matrix. This is the same signal as the existing Steam Generator Isolation Signal (SGIS) contact output from the SGLS matrix. Since the signal is taken from an existing contact output from the SGLS matrix, no additional components are affected or added. The associated analog instrument loops are unaffected and, therefore, the loop uncertainty is not affected by this change.

3.3 TS Changes for Containment Spray Logic

3.3.1 TS 2.14, “Engineered Safety Features System Initiation Instruments Setting,” Table 2-1, Function Unit 4, “Low Steam Generator Pressure,” would be revised by adding a new channel. The new channel would be (c), “Containment Spray” with a setting limit of ≥ 500 pounds per square inch absolute [psia). The staff concludes this is acceptable since ≥ 500 psia is used for channel (a), “Steam Line Isolation” for Functional Unit 4, “Low Steam Generator Pressure” and is fed from the same steam generator pressure contact.

3.3.2 TS 2.14, “Engineered Safety Features System Initiation Instruments Setting,” Table 2-1, current Note (3), “Simultaneous high containment pressure and pressurizer low/low pressure” would be revised, to new Note (3), “Simultaneous containment high pressure, pressurizer low/low pressure, and steam generator low pressure.” The staff concludes that the revising of Note (3) is acceptable since containment spray auto-start logic will now start on PPLS and CPHS coincident with SGLS.

3.3.3 TS 2.15, "Instrumentation and Control Systems," Table 2-3, “Instrument Operating Requirements for Engineered Safety Features," Functional Unit 2A, "Containment Spray, Manual," would be revised by adding a new Note m, "Steam Generator Low Pressure permissive is required for actuation." The staff concludes this is acceptable as it conforms to the new containment spray logic configuration.

3.3.4 TS 2.15, "Instrumentation and Control Systems," Table 2-3, “Instrument Operating Requirements for Engineered Safety Features," Functional Unit 2D, "Containment Spray, Steam Generator Low Pressure," would be added to reflect the minimum operable channels, minimum degree of redundancy, permissible bypass conditions, and test, maintenance and inoperable bypasses. Additionally, Note n, "Auto removal of bypass prior to exceeding 600 psia," would be added. The staff concludes this is acceptable as it conforms to the new containment spray logic configuration.

3.3.5 TS 2.15, "Instrumentation and Control Systems," Table 2-3, “Instrument Operating Requirements for Engineered Safety Features," Note C would be revised to read, "Coincident containment high pressure, pressurizer pressure, and steam generator low pressure are required for initiation of containment spray." The staff concludes this is acceptable as it reflects the addition of steam generator low pressure to the containment spray logic.

3.3.6 TS 2.15, "Instrumentation and Control Systems," Table 2-4, "Instrument Operating Conditions for Isolation Functions," Note a, "Circuits on ESF Logic Subsystems A and B each have 4 channels," would be applicable to Functional Unit 2B(i) "Steam Generator Isolation, Steam Generator Low Pressure." The staff concludes this is acceptable as it reflects the number of channels in the steam generator low pressure logic subsystem.

3.3.7 TS 3.1, “Instrumentation and Control”, Table 3-2, Channel Description 5, “Containment Spray Actuation Logic,” would be revised by adding a new signal to the channel functional test. The new item would be, “SGLS 2/4 Logic.” The staff concludes the addition of the new signal is acceptable since this signal is part of the containment spray pump auto-start logic and would provide for the testing of this new signal.

3.3.8 TS 3.1, “Instrumentation and Control,” Table 3-2, new Note (8), “SGLS is required for containment spray pump actuation only. SGLS lockout relays are not actuated for this test.” The staff concludes the addition of Note (8) is acceptable since containment spray actuation is not desired during power operation.

3.4 Conclusion

The NRC staff reviewed the licensee’s technical analysis of the proposed changes to the “Engineered Safety Features System Initiation Instruments Settings”, and “Instrumentation and Control Surveillance Requirements” sections of the TS against the requirements in 10 CFR Part 50, Appendix A, GDCs 13 and 20. Based on the above, the staff concludes that the addition of a new function to TS 2.14, Table 2-1 is not a new setpoint and the addition of a new signal to TS 3.1, Table 3-2 is acceptable. Since an existing setpoint is being used for the new function, the staff did not review the licensee’s setpoint methodology.

4.0 RADIOLOGICAL EVALUATION

4.1 Regulatory Evaluation

Amendment 201 to FCS Renewed Facility Operating License, issued December 5, 2001 (Reference 30), allowed the current alternative source term (AST) used in design-basis radiological accidents for control room habitability to be replaced with an AST in accordance with 10 CFR 50.67, and following the guidance of RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” Therefore, this evaluation has been conducted to verify that the results of the licensee’s affected DBA radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67 for offsite doses and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, “Control Room Habitability” (or equivalent for plants licensed before the GDC were in existence). The applicable acceptance criteria are 5 rem [roentgen equivalent man] Total Effective Dose Equivalent (TEDE) in the control room (CR), 25 rem TEDE at the exclusion area boundary (EAB), and 25 rem TEDE at the outer boundary of the low population zone (LPZ). The dose acceptance criterion in the Technical Support Center (TSC) is accepted to be 5 rem TEDE for the duration of the accident to show compliance with the regulatory requirements of NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

FCS Design Criterion 17, “Monitoring Radioactivity Releases.” Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients and from accident conditions.

Except where the licensee proposes a suitable alternative, the staff used the regulatory guidance provided in applicable sections of RG 1.183 and NUREG-0800, SRP Chapter 15, for DBAs, and SRP Chapter 6.4, for control room habitability, in performing this review. Also, the staff referenced the FCS TS and USAR for verification of design-basis input, as the licensee relied on these documents for input to their analyses.

4.2 Technical Evaluation

The licensee states that currently, containment spray operation is initiated by the same basic signals as safety injection, but in a different logic combination. The containment spray actuation signal (CSAS) results from coincidence of PPLS and CPHS, both on a two-out-of-four basis; the CSAS brings the system to full operation. Initially, the pumps take suction from the SIRWT, and upon reaching low-tank level, the RAS is initiated, automatically transferring the pump suction to the containment recirculation line inlet from the containment sump. Two containment spray pumps (SI-3A and SI-3B) are automatically started by the CSAS, and containment spray pump SI-3C is connected to the electrical bus associated with DG-2. However, containment spray pump SI-3C only has manual-start capabilities and is intended to be used to replace SI-3A when SI-3B is not running or replace SI-3B at any time.

Following the proposed modification to the containment spray system, none of the containment spray pumps will automatically start during a LOCA. A physical modification, or engineering change (EC 40070), to the plant will require that both the SGLS and the CSAS be present before the containment spray system is actuated in response to a design-basis LOCA. The licensee will also add an SGLS interlock to each containment spray pump, SI-3A and SI-3B, sequencer circuit. Though this change prevents automatic actuation of the containment spray system during a LOCA, it does not impact the overall containment spray initiation time delay assumed in the licensee's MSLB containment pressure and temperature response AOR, where containment spray is still credited for post-accident containment pressure and temperature reduction. In response to a design-basis MSLB accident, it is logical that the affected FCS steam generator will drain and lead to the SGLS required for containment spray actuation.

Following the proposed change, the containment air coolers are now relied upon as the primary source of containment pressure suppression and temperature reduction following the postulated design-basis LOCA. For this LAR, the licensee has verified the capability of the containment air coolers to perform this function without the need for containment sprays. The licensee has proposed no changes to the containment air cooler operation or initiation in the submitted LAR.

The staff reviewed the licensee's regulatory and technical assessments, as they relate to the radiological consequences of DBA analyses, in support of the proposed changes to the FCS license. The licensee determined, and the staff agrees, that the only DBA analysis which is affected by the new containment spray system actuation logic is the design-basis LOCA. This is because, of the FCS DBAs, only the postulated MSLB accident and LOCA analyses credit the containment spray system for accident mitigation, and, as discussed above, the proposed change does not affect the current relied upon containment spray system response during an MSLB accident. Further, this evaluation focuses on the specific effect of the proposed license changes and acknowledges FCS's current and recently approved AST design-basis assumptions. Information regarding the effect of the licensee's proposed changes on the design basis LOCA dose consequence analysis was provided by the licensee in Attachment 4 to the licensee's July 30, 2007, LAR. The findings of this safety evaluation are based upon the descriptions and results of the licensee's assessment and other supporting information docketed by the licensee.

The licensee has determined, and the staff agrees, that there are three affected activity release paths associated with the DBA analysis:

- Containment pressure relief line release
- Containment leakage
- Engineered Safety Feature (ESF) system leakage

The following sections address the impact of the licensee's proposed changes on these affected activity release paths.

4.2.1 Containment Pressure Relief Line Release

For the containment pressure relief line release pathway, the licensee assumed that the relief line is operational at the initiation of the LOCA and that the release is terminated as part of containment isolation. The entire RCS inventory was assumed to be at TS levels and released to the containment at T = 0 hours. It is also assumed, in accordance with the guidance of RG 1.183, that 100 percent of the airborne activity is instantaneously and homogeneously mixed in the containment atmosphere. Over a period of 5 seconds prior to containment isolation, the licensee calculated a 600 standard cubic feet per minute (scfm) activity release rate. This release rate was based on containment pressurization (due to the RCS mass and energy release) and the pressure relief line cross-sectional area through which the release occurs. This release to the environment is postulated to occur via the auxiliary building ventilation stack. Because the release from this pathway is isolated within 5 seconds after the initiation of the LOCA, only activity present in the coolant is available for release, as the onset of fuel gap activity release is assumed to begin at 30 seconds following accident initiation. This AST assumption is consistent with the guidance of RG 1.183. Also consistent with the guidance of RG 1.183 for gaseous coolant releases, the chemical form of the iodine released from the RCS is assumed to be 97 percent elemental and 3 percent organic. This is acceptable to the staff.

4.2.2 Containment Leakage

For the containment leakage pathway, the licensee assumed the activity available for release from the reactor is based on in an equilibrium reactor core inventory of dose-significant isotopes, as suggested by the guidance of RG 1.183. The isotopic activity of this equilibrium core inventory was calculated by the licensee assuming maximum full power operation at 1.02 times the current licensed thermal power or 1530 megawatts thermal (MWt), to account for ECCS evaluation uncertainty, and taking into consideration characteristic fuel enrichment and burnup. For the assumed release fractions, timing, and radionuclide grouping, the licensee uses the guidance shown in Tables 2, 4, and 5 of RG 1.183, respectively. This is acceptable to the staff.

In accordance with RG 1.183 and the current licensing basis, the licensee assumed that the activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of containment as it leaves the core. Also consistent with RG 1.183, fuel was assumed to be linearly released in two sequential phases: (1) the gap release phase, beginning at 30 seconds following the initiation of the LOCA and continuing for 30 minutes, and (2) the early in-vessel release phase, which begins 30 minutes into the accident and continues for 1.3 hours, at which time the release into containment is assumed to terminate. Consistent with the guidance of RG 1.183, the licensee assumed that activity leaks through containment to the environment at the design-basis leak rate (L_a) of 0.1 percent per day for the first 24 hours, then at a reduced rate of 0.05 percent per day for the remaining accident duration. The licensee credited the containment recirculation fan coolers (CRFC) for containment mixing between regions above and below the operating floor for the duration of the accident, and for providing temperature and pressure reduction to support the reduced post-LOCA leak rate after a day. This credit taken for the fan coolers is based on the minimum CRFC flowrate and the maximum containment free volumes above and below the operating floor. This is acceptable to the staff.

4.2.3 Containment Release Filtration

The LAR proposes to no longer credit containment spray in mitigating LOCA activity releases; however, in its new AOR and model of activity transport, the licensee credited the CRFC for aerosol removal by HEPA filters. For FCS, the licensee used the existing HEPA installation and maintenance inspection procedures, in conjunction with additional requirements relative to filter replacement, and a calculation that estimates the maximum potential bypass leakage following a DBA (with a conservative safety factor of 2 applied to bound filter performance) to support the 50 percent particulate filter efficiency used in the dose consequence analysis. The assumed 50 percent particulate removal efficiency attributed to the HEPA filter is generally conservative. This is acceptable to the staff.

4.2.4 Iodine Plateout

The licensee also credited the surface of sump water for removal of elemental iodine by plateout. This is consistent with the SRP 6.5.2 allowance for elemental iodine plateout on wetted surfaces. The licensee states that, because the containment spray system is no longer immediately actuated in the event of a design-basis LOCA, neither condensation nor sprays can be demonstrated on the containment walls. Therefore, the surface of the sump water is credited as the renewable wetted surface. The licensee supports the assumption that the sump water surface is renewable based on the thermally convective dynamic established by cooler ECCS injection water entering the warmer, and therefore denser, sump water as a stream, thereby establishing momentum-driven currents to promote surface renewal. The licensee also states, and the staff agrees, that structurally, the containment directs water to the sump, which avoids water holdups and isolated water sections, and works to extend the area of influence for the convective and momentum-driven flow. For additional conservatism, and to address uncertainty, the licensee only credits half of the available containment sump surface area (3000 square feet) until the recirculation phase for core injection is initiated at 3.25 hours. The licensee assumed a maximum decontamination factor (DF) for elemental iodine of 200, based on SRP 6.5.2. This approach is conservative, consistent with applicable regulatory guidance, and therefore, acceptable to the staff.

4.2.5 ESF System Leakage

With the exception of noble gases, the licensee assumed that all activity released from the core during the gap and early in-vessel release phases is instantaneously and homogeneously mixed in the containment sump water at the time of release from the fuel. This is a conservative model of the activity transport and consistent with the guidance of RG 1.183. Also consistent with their current licensing basis, the licensee used a minimum sump volume in the analysis of activity transport through this pathway. This assumption maximizes the activity concentration in the sump water. With the exception of iodine, all radioactive materials in the recirculating coolant are assumed to be retained in the liquid phase.

As a design basis at FCS, the total post-LOCA ESF leakage is defined as the maximum allowed leakage from the equipment carrying sump fluids and located outside containment, and back-leakage of sump water into the SIRWT. The licensee postulated both of these leakage sources to release into the FCS auxiliary building and begin following the initiation of recirculation mode at 20.4 minutes into the accident. The TS limited combined leakage value of 3800 cubic centimeters per hour (cm^3/hr) assumed in the DBA analysis is doubled, for conservatism in accordance with the guidance of RG 1.183, to 7600 cm^3/hr. The coolant that leaks into the auxiliary building is assumed to flash and exhaust 10 percent of the available activity and exhaust it directly to the environment without credit for mixing, holdup, or filtration. The assumed 10 percent flashing fraction is conservative because of calculated sump coolant temperatures below 212 °F. The licensee assumed that the chemical form of the iodine released from the sump water is 97 percent elemental and 3 percent organic. The licensee's assumptions used to model this ESF system activity release pathway are all reasonable, conservative, and consistent with the guidance of RG 1.183; therefore, they are acceptable to the staff.

4.2.6 Control Room Habitability and Modeling

The design basis LOCA dose consequence analysis modeling of the FCS control room design and operation remains unchanged from that which was accepted for the approved AST amendment of December 5, 2001.

4.2.7 Atmospheric Dispersion

The licensee used atmospheric dispersion factors (χ/Q values) from their current licensing basis. For the CR, EAB, and LPZ dose estimates, the χ/Q values implemented in this proposed LAR were previously reviewed and approved by the staff for AST Amendment 201 on December 5, 2001.

4.3 Conclusion

As described above, the staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impact of the proposed licensing basis changes on the postulated design-basis LOCA. The staff finds that OPPD's new design-basis LOCA analysis conservatively accounts for the impact of the proposed changes. Based on the above, the NRC staff has concluded that the licensee will continue to meet the applicable dose acceptance criteria, as identified in Section 4.2 of this evaluation, following implementation of these changes. The staff further finds reasonable assurance that FCS Unit No. 1, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to mitigate unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological dose consequences of the DBAs.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 findings published in the *Federal Register* on August 28, 2007 (72 FR 49581), and January 29, 2008 (73 FR 5227). 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.

7.0 COMMITMENTS

In the FCS letter dated April 10, 2008, the licensee made commitments and agreed to licensing actions specified in Sections 2.3 and 8.0 of this safety evaluation.

8.0 ADDITIONAL CONDITIONS

In the FCS letter dated April 10, 2008, the licensee agreed to added license conditions related to the replacement and testing of CACF unit HEPA filters and surveillance testing of the CACF unit relief ports. The license conditions require administrative controls pending the completion of detailed analysis and confirm commitments for the licensee to submit TS amendments by October 31, 2008. Appendix B, “Additional Conditions,” to Renewed Facility Operating License DPR-40 is amended to designate the existing license condition for Amendment No. 181 as (1), and add new license conditions (2) and (3), designated as Amendment No. 255, to read as follows:

(2) This license shall be deemed to contain the following specified conditions and specified actions:

   (a) During the 2008 RFO [refueling outage], replace the CACF [containment air cleaning and filtering] unit HEPA [high-efficiency particulate air] filters not previously replaced during 2006;

(b) During the 2008 RFO, ensure all CACF unit HEPA filters meet the 2” wc [water column] differential pressure limitation;

(c) Implement and maintain the newly created HEPA filter replacement criteria in Procedure PE-RR-VA-0209 unless superseded by other criteria via the license amendment process; and

(d) Submit a license amendment request (LAR) by October 31, 2008, that will add the HEPA filter testing and replacement criteria to the FCS [Fort Calhoun Station] Technical Specifications.

(3) This license shall be deemed to contain the following specified conditions and specified actions:

(a) During the 2008 RFO, perform the surveillance operability testing of the containment cooling unit relief ports; and

(b) Submit a LAR by October 31, 2008, that will add surveillance operability and testing of the containment cooling unit relief ports to the FCS Technical Specifications.

9.0 CONCLUSION

The Commission 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.

10.0 REFERENCES

1. Letter dated July 30, 2007, from David J. Bannister, Acting Site Director, Omaha Public Power District, to NRC, Subject: Fort Calhoun Station, Unit No. 1, License Amendment Request (LAR), Modification of Containment Spray System Actuation Logic (ADAMS Accession No. ML072150293).

2. Letter dated October 19, 2007, from D.J. Bannister, Site Director, Fort Calhoun Station, to NRC, Subject: Fort Calhoun Station, Unit No. 1 License Amendment request (LAR) “Modification of Surveillance Requirements for Emergency Mode (Remotely Operated) Dampers in CACFS (ADAMS Accession No. ML072950082).

3. Letter dated December 12, 2007, from D.J. Bannister, Site Director, Omaha Public Power District, to NRC, Subject: Revision to Technical Evaluation of Fort Calhoun Station, Unit No. 1, License Amendment Request (LAR) “Modification of Surveillance Requirements for Emergency Mode (Remotely Operated) Dampers in CACFS (ADAMS Accession No. ML073461006).

4. Letter dated February 21, 2008, from David J. Bannister, Vice President, Omaha Public Power District, to NRC, Subject: Response to Request for Additional Information Regarding License Amendment Request for Proposed Technical Specification Changes for Modification of Containment Spray System Actuation Logic (ADAMS Accession No. ML080580407).

5. Letter dated March 28, 2008, from R.P. Clemens, Division Manager – Nuclear Engineering, Omaha Public Power District, to NRC, Subject: Response to Second Round Request for Additional information Regarding Containment Analysis for Fort Calhoun Station Water Management License Amendment Request (TAC No. MD6204) (ADAMS Accession No. ML080910060).

6. Letter dated April 4, 2008, from R.P. Clemens, Division Manager – Nuclear Engineering, Omaha Public Power District, to NRC, Subject: Fort Calhoun Station Response to Request for Additional Information License Amendment Request Re: Dampers in Containment Air Cooling (MD7043) (ADAMS Accession No. ML080950468).

7. Letter dated April 10, 2008, from R.P. Clemens, Division Manager – Nuclear Engineering, Omaha Public Power District, to NRC, Subject: Fort Calhoun Station Response to Request for Additional Information License Amendment Request Re: Modification of Containment Spray Actuation Logic (MD6204) (ADAMS Accession No. ML081010122).

8. NRC Meeting Summary dated April 18, 2008, for meeting held in Rockville, MD, December 12, 2006, Subject: Resolution of GSI-191 (ADAMS Accession No. ML081000002).

9. NRC Meeting Summary dated October 20, 2006, for meeting held in Rockville, MD, September 21, 2006, Subject: Water Management Strategies Program and Completion of GSI-191 Commitments (ADAMS Accession No. ML060270064).

10. NRC Meeting Summary dated May 30, 2006, for meeting held in Rockville, MD, May 11, 2006, Subject: Water Management Post-LOCA (ADAMS Accession No. ML061460369).

11. Letter dated July, 13, 2004, from James F. Mallay, Director, Regulatory Affairs, Framatome ANP, Inc., to NRC, Subject: Request for Approval of BAW-10252(P), Revision 0, "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC," (ADAMS Accession No. ML041980470).

12. Letter dated September 6, 2005, from Mohan C. Thadani, NRC, to Ronnie L. Gardner, Framatome ANP, Subject: Correction to Letter Forwarding the Final Safety Evaluation for Framatome ANP Topical Report BAW 10252-P Revision 0, "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC" (TAC No. MC3783) (ADAMS Accession No. ML052450297).

13. Letter dated January 27, 2003, from D.J. Bannister, Plant Manager, Fort Calhoun Station, to NRC, Subject: Fort Calhoun Station Unit 1 License Amendment

Request, "Containment Pressure Analysis using the GOTHIC Computer Code (ADAMS Accession No. ML030360205).

14. Letter dated November 5, 2003, from Alan B Wang, NRC, to R.T. Ridenoure, Division Manager – Nuclear Operations, Omaha Public Power District, Subject: Fort Calhoun Station Unit 1 – Issuance of Amendment (MD7496) (ADAMS Accession No. ML033100290).

15. Letter dated April 9, 2002, from Leslie W. Barnett, NRC, to James F. Mallay, Director, Regulatory Affairs, Framatome ANP, Richland, Inc., Subject: Safety Evaluation of Framatome Technologies Topical Report BAW-10164-P, Revision 4, "RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses" (ADAMS Accession No. ML013390204).

16. Memorandum dated September 18, 1992, from Samuel J. Chilk, Secretary, NRC, for James M. Taylor, Executive Director for Operations, Subject: "Resolution of Deviations Identified during the Systematic Evaluation Program" (ADAMS Legacy Library Accession No. AN9210060362).

17. Fort Calhoun Station Updated Final Safety Analysis Report Section 6.4, Engineered Safeguards: CACFS, Revision 7 July 5, 2005 (ADAMS Accession No. ML003717543).

18. Letter dated January 10, 2003, from James F. Mallay, Director, Regulatory Affairs, Framatome ANP, Inc., to NRC, Subject: Publication of BAW-10164-P-A, Revision 4, "RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis" (ADAMS Accession No. ML030220113).

19. Letter dated August 31, 2005, from Herbert N. Berkow NRC, to Ronnie L. Gardner, Manager, Site Operations and Regulatory Affairs, Framatome ANP, Inc., Subject: Final Safety Evaluation for Framatome ANP Topical Report BAW-10252(P), Revision 0, "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC" (TAC No. MC3783) (ADAMS Accession Nos. ML041980470 and ML052240307 – proprietary).

20. U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, March 2007 (ADAMS Accession No. ML070810350).

21. U.S. Nuclear Regulatory Commission, NUREG-0800 (SRP), Section 9.2.5, Ultimate Heat Sink, Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," July 1981 (ADAMS Accession No. ML052350549).

22. U.S. Nuclear Regulatory Commission, NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings," by K.D. Marx, May 1986 (ADAMS Legacy Library Accession No. AN8607240299).

23. Same as reference 1. Attachment 6, Summary of FCS Containment Analysis Without Containment Spray, July 2007 (AREVA Proprietary) (ADAMS Accession No. ML072150309).

24. Fort Calhoun Station, Updated Safety Analysis Report, Section 5.1, Structures - Containment Structure, November 8, 2006 (ADAMS Accession No. ML003717543).

25. U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Section 6.2.1.1.A, “PWR Dry Containments, Including Subatmospheric Containments, July 1981 (ADAMS Accession No. ML033580033).

26. U.S. Nuclear Regulatory Commission, Generic Letter 97-04, “Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,” October 7, 1997 (NRC website http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1997/gl97004.html).

27. Letter dated March 7, 2000, from L. Raynard Wharton, NRC, to S. K. Gambhir, Division Manager – Nuclear Operations, Omaha Public Power District, Subject: Fort Calhoun Station Unit No. 1 – Generic Letter 97-04, “Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps (ADAMS Accession No. ML003689023).

28. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.82, “Water Sources for Long-Term Containment Cooling Following a Loss-of-Coolant Accident,” November 2003 (NRC website http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-082/01-082.pdf).

29. Letter dated August 31, 2007, from D.J. Bannister, Omaha Public Power District, to NRC, Subject: Fort Calhoun Station Unit No. 1, Response to Request for Additional Information for License Amendment Request (LAR), "Modification of the Containment Spray System Actuation Logic" (ADAMS Accession No. ML072480667).

30. Letter dated December 5, 2001, from A. B. Wang, NRC, Subject: Fort Calhoun Station, Unit No. 1, Issuance of Amendment [201] (Tac No. MB1221) (ADAMS Accession No. ML013030027).

Principal Contributors: R. Lobel
H. Walker
B. Heida
B. Marcus
A. Boatright

Date: May 2, 2008