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444 South 16th Street Mall
Omaha NE 68102-2247

July 30, 2007
LIC-07-0052

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

References:
1. Docket No. 50-285
2. NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 (NRC-04-0115)
3. Letter from OPPD (H. J. Faulhaber) to NRC (Document Control Desk), Follow-up Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated August 31, 2005 (LIC-05-0101)
4. Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure), "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment, RE: (TAC NO. MC3217)," dated May 20, 2005 (NRC-05-0064)
5. Post-LOCA Water Management Strategies to Optimize Long Term Core Cooling Availability, dated May 30, 2006 (ML061460369)
6. Post-LOCA Water Management Strategies to Optimize Long Term Core Cooling Availability, NRC Meeting – NRC Slides, May 11, 2006 (ML0614603383)
7. Post-LOCA Water Management Strategies to Optimize Long Term Core Cooling Availability, NRC Meeting – Westinghouse Slides, May 11, 2006 (ML0614603378)
8. Letter from NRC (C. Haney) to OPPD (R. T. Ridenoure), "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," Extension Request Approval (TAC No. MD2323), dated August 11, 2006 (NRC-06-0103)

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request (LAR), "Modification of the Containment Spray System Actuation Logic"

Pursuant to 10 CFR 50.90, the Omaha Public Power District (OPPD) requests changes to the Fort Calhoun Station (FCS) Unit No.1 Operating License No. DPR-40 to modify the containment spray (CS) system actuation logic to preclude automatic start of the containment spray pumps for a loss-of-coolant accident (LOCA).

A001

The proposed modification changes containment pressure control during a LOCA from CS to the containment air coolers and mitigates the control room and offsite doses by the use of containment high efficiency particulate air (HEPA) filters. This change will increase the amount of water delivered to the core during the injection phase of a LOCA and will reduce the amount of debris carried to the containment sump strainers during a LOCA. This approach to management and conservation of water is consistent with References 5, 6, and 7 and supports OPPD's resolution of Generic Safety Issue GSI-191 and response to Generic Letter (GL) 2004-02.

Proposed revisions to the Basis of Technical Specifications (TS) 2.4, previously approved in Reference 4, as well as the other revisions are included in this license amendment request. The proposed changes to TS 2.4, 2.14, 2.15, 3.1, 3.5(4), 3.6, as well as the Updated Safety Analysis Report (USAR) Appendix G, are supported by the Significant Hazards Consideration, Attachment 1, Section 4.3.

Credit for operation of the CS system to control containment pressure will be retained for a main steam line break (MSLB) accident; therefore no changes to the CS system operability requirements of TS 2.4 or surveillance requirements of TS 3.6 are proposed. Mitigation of the MSLB accident does not require use of the safety injection (SI) pumps in the recirculation mode; therefore, sump strainer clogging is not a concern for the MSLB accident. Surveillance requirements of CS system containment penetrations, previously considered in service during a LOCA and exempted from containment isolation valve leak rate testing, are now being added to TS 3.5 for containment isolation valve testing.

This amendment is needed to meet the compliance date requirements of OPPD's approved extension to GL 2004-02 (Reference 8). The amendment scope and schedule have been discussed with the NRC staff. In order for the modification to the containment spray system actuation logic to be completed during the next refueling outage, OPPD requests approval of the proposed amendment by April 1, 2008 with the implementation period ending prior to power operation following startup from the 2008 refueling outage.

Pursuant to 10 CFR 2.790, OPPD requests that the proprietary information presented and discussed in the "Summary of FCS Containment Analysis without Containment Spray" (Attachment 6) be withheld from public disclosure. The methodology used in this analysis is proprietary to AREVA NP, Inc., as justified in the supporting affidavit provided in Attachment 5. The methodology was done under exclusive contract to OPPD, using AREVA's 10 CFR 50 Appendix B quality program. Attachment 7 is the non-proprietary version of Attachment 6 with deleted proprietary information enclosed in brackets.

There are no regulatory commitments in this LAR.

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

If you should have any questions regarding this submittal or require additional information, please contact Mr. Thomas C. Matthews at (402) 533-6938.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 30, 2007.

David J. Bannister
Acting Site Director

DJB/CO/DLL/dll

Attachments:

1. OPPD's Evaluation of the Proposed Change
2. Markups of Technical Specifications and USAR pages
3. Proposed Technical Specifications and USAR pages (Retyped)
4. Implementation of GSI-191 Containment Sump Modification, Shaw Stone & Webster Nuclear, June 11, 2007
5. Affidavit for AREVA NP, Inc., Proprietary Information
6. Summary of FCS Containment Analysis without Containment Spray, AREVA NP, Inc. Document 77-9051353P-001, July 2007 - Proprietary
7. Summary of FCS Containment Analysis without Containment Spray, AREVA NP, Inc. Document 77-9051353NP-001, July 2007 - Non-Proprietary

c: B. S. Mallett, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
L. M. Willoughby, NRC Senior Resident Inspector
Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska Health and Human Services, State of Nebraska (w/o proprietary attachment)

ATTACHMENT 1

Omaha Public Power District's
Evaluation of the Proposed Change

Modification of the Containment Spray System Actuation Logic

1.0 SUMMARY DESCRIPTION

The Omaha Public Power District (OPPD) requests changes to the Fort Calhoun Station (FCS), Unit No. 1, Operating License No. DPR-40 to modify the containment spray (CS) system actuation logic to preclude automatic start of the CS pumps for a loss-of-coolant accident (LOCA). The proposed modification changes containment pressure control during a LOCA from containment spray to the containment air coolers, and mitigates the control room (CR) and offsite doses by the use of containment high efficiency particulate air (HEPA) filters. This change will increase the amount of water delivered to the core during the injection phase of the LOCA and will reduce the amount of debris carried to the containment sump strainers during a LOCA. This approach to management and conservation of water is consistent with References 6.12, 6.13, and 6.14 and supports OPPD's resolution of Generic Safety Issue GSI-191 and response to Generic Letter (GL) 2004-02 (Reference 6.8).

Credit for operation of the CS system will be retained for a main steam line break (MSLB) accident; therefore, no changes to the CS system operability requirements of Technical Specification (TS) 2.4 or surveillance requirements of TS 3.6 proposed. Mitigation of the MSLB accident does not require use of the safety injection (SI) pumps in the recirculation mode, therefore sump strainer clogging is not a concern for this accident. Surveillance requirements of CS system containment penetrations, previously considered in service during a LOCA and exempted from containment isolation valve leak rate testing, are now being added to TS 3.5 containment isolation valve testing. Proposed revisions to the Basis of TS 2.4, previously approved in Reference 6.7, as well as other revisions are included in this license amendment request (LAR).

The modification requires revisions to the following documents: TS 2.4, *Containment Cooling*, (1)b *Minimum Requirements* and (2)a, *Modification of Minimum Requirements*, and TS 2.4 Basis; TS 2.14, Table 2-1, *Engineered Safety Features System Initiation Instrument Setting Limits*, for Low Steam Generator Pressure; TS 2.15, Table 2-3, *Instrument Operating Requirements for Engineered Safety Features* for Pressurizer Low/Low Pressure, and TS 2.15, Table 2-4, *Instrument Operating Conditions for Isolation Functions*; TS 3.1, Table 3-2, *Minimum Frequencies for Checks, Calibrations and Testing of Engineered Safety Features, Instrumentation and Controls*; TS 3.5(4), *Surveillance Requirements–Containment Isolation Valves Leak Rate Tests*; TS 3.6, *Safety Injection and Containment Cooling Systems Tests*; and, Updated Safety Analysis Report (USAR) Appendix G, *Responses to 70 Design Basis Criteria* - Criteria 15, 41, and 52.

These proposed changes will support the modification of the CS system actuation logic to prevent automatic CS pump start during a LOCA. The logic change will be physically accomplished by adding the steam generator low signal (SGLS) contact in series with the containment spray actuation signal (CSAS) contact in the containment spray pump sequencer circuit via modification/engineering change (EC) 40070.

2.0 DETAILED DESCRIPTION

2.1 Change Descriptions

In order to utilize the containment air coolers in place of the containment spray system as the means of containment pressure control and the fission product release mitigation during a LOCA, and thereby reduce the amount of debris washed to the containment sump strainers, the following TS and USAR changes are proposed:

2.1.1 TS 2.4, (1)b, *Minimum Requirements* and (2)a, *Modification of Minimum Requirements*, are revised to assure the availability of the containment air cooling and filtering system for responding to a limiting LOCA following the actuation system modification. Specifically, one train of the containment air cooling and filtering system, either (VA-3A and VA-7C) or (VA-3B and VA-7D), must be operable during power operations.

TS 2.4, Basis, is revised to conform with the change in containment spray system and containment air cooler responses to a limiting LOCA, which after modification would not actuate containment spray but rely upon the containment air coolers for pressure control. As a result, the minimum time to recirculation of water from the containment sump increases from 24 minutes to 45 minutes for the LOCA. The Basis is also being updated to delete the allowance for the shutdown of excess CS pumps added by Amendment 235, which is no longer necessary after installation of the modification to the spray pump actuation logic.

2.1.2 TS 2.14, Table 2-1, *Engineered Safety Features System Initiation Instrument Setting Limits*, Low Steam Generator Pressure functions are revised to be consistent with the change in engineered safety features actuation signal (ESFAS) logic as a result of the containment spray system actuation logic change.

2.1.3 TS 2.15, Table 2-3, *Instrument Operating Requirements for Engineered Safety Features* for Pressurizer Low/Low Pressure, and Table 2-4, *Instrument Operating Conditions for Isolation Functions*, are modified to reflect the logic for not starting the spray pumps for a LOCA event.

2.1.4 TS 3.1, Table 3-2, *Minimum Frequencies for Checks, Calibrations and Testing of Engineered Safety Features, Instrumentation and Controls*, Item 5, *Containment Spray Actuation Logic*, is revised to include testing of the lockout relays associated with the logic modification.

2.1.5 TS 3.5(4), *Surveillance Requirements – 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. These valves were previously exempted from testing and inspection due to their function during a LOCA.

2.1.6 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 assure the filters are capable of removing radioactive particulates during a LOCA.

2.1.7 USAR Appendix G, *Responses to 70 Criteria*, Criteria 15, 41 and 52, are revised to be consistent with the change in ESFAS logic as a result of the containment spray system actuation logic change.

These proposed TS changes will support the modification of the CS system actuation logic to prevent automatic CS pump start during a LOCA, by crediting the containment air coolers as the only active containment energy removal mechanism during a LOCA. NRC approval of the methodology changes provided in Attachment 6 is required.

Additional TS Bases changes, which are included in the attached mark-ups and "clean" pages, will be processed in accordance with the OPPD TS Bases Change (TSBC) control program. These Bases changes are provided for information purposes only.

2.2 Conditions that the Proposed Change is Intended to Resolve

Evaluations have shown that the net positive suction head (NPSH) margin for the high pressure safety injection (HPSI) pumps during the recirculation phase of a design basis accident (DBA) can be enhanced by reducing the head loss and hydraulic resistance through the containment sump strainers when the HPSI pumps are operating in the recirculation mode (the low pressure safety injection (LPSI) pumps are automatically shut off following a recirculation actuation signal (RAS)). This reduction in flow will be accomplished by modifying the CS system actuation logic to require that both the SGLS and the CSAS are initiated before the CS system is actuated. Thus, CS will not start in response to a LOCA. The enhancement in the NPSH performance will be realized due to reduced transport of debris to the strainer resulting in a reduction in the pressure drop across the strainer and a reduction in piping head loss. This will provide additional margin for the NPSH available ($NPSH_A$) for the HPSI pumps taking suction from the containment sump, increase the amount of water delivered to the core during the injection phase of a LOCA and will increase the time to the initiation of a RAS.

Another benefit of the proposed delay in RAS will be an increased sump pool turnover time during a LOCA. This change is beneficial in that it allows any accident generated debris that is transported to the containment sump pool more time to settle to the floor and thus is less likely to be drawn to the containment sump strainer following RAS. Also, since the containment will no longer be sprayed down, the amount of debris that will be washed into the sump pool will only come from the vicinity of the break.

This approach to management and conservation of water is consistent with References 6.12, 6.13, and 6.14 where the NRC discussed the safety enhancements from extending the post-LOCA injection phase (delaying the onset of the containment recirculation phase). Extending the injection phase would give operators more time to establish a reliable recirculation path, would reduce the debris reaching the containment recirculation sump screen, would reduce downstream effects resulting from containment recirculation, and would extend the time for mitigative actions. It was pointed out that, by minimizing the use of containment spray (possibly by eliminating automatic spray initiation), significantly more water could be delivered to the core during the injection phase of the LOCA.

As a result of these discussions, OPPD considered the opportunity presented by this initiative as a means to address the generic safety issue GSI-191 resolution challenges being faced at FCS. In addition to those measures already taken to address GL 2004-02 and GSI-191 (i.e., removing insulation in containment, changing buffer agents, increasing strainer module size, etc.), OPPD is taking actions that will not only address the GSI-191 resolution challenges but also have the potential for improving the overall safety of the plant.

2.3 Reason for Proposed Change

This change will reduce the amount of debris carried to the containment sump strainers during a LOCA, thereby reducing the probability of strainer plugging and minimizing HPSI pump vulnerability to a reduced NPSH during the recirculation mode when the pumps draw water from the containment sump.

3.0 TECHNICAL EVALUATION

3.1 System Descriptions

3.1.1 Containment Spray (CS) System Description

The CS system is discussed in FCS USAR Sections 6.2, 6.3, and 7.3.2.4 (References 6.1, 6.2, and 6.3, respectively).

The CS system consists of the safety injection and refueling water tank (SIRWT), three spray pumps, two shutdown cooling heat exchangers and all

necessary piping, valves, instruments and accessories. The pumps discharge the borated water through the two heat exchangers to a dual set of spray headers and spray nozzles in the containment. One pump operation is sufficient to meet the capacity requirements in the event of a DBA. CS valve interlocks assure that one CS header valve remains closed if only one spray pump is available, to prevent a pump run-out condition.

Two spray pumps (SI-3B and SI-3C) are located in one engineered safeguards room, along with one HPSI pump (SI-2B) and one LPSI pump (SI-1B). The third spray pump (SI-3A) is located in the second engineered safeguards room with one LPSI pump (SI-1A) and two HPSI pumps (SI-2A and SI-2C). Each engineered safeguards room has separate pump suction from both the SIRWT and the containment recirculation line inlet to ensure that the pumps in one room will have adequate suction if the suction line to the second room fails.

CS pumps SI-3B and SI-3C, LPSI pump SI-1B and HPSI pump SI-2B are powered from diesel generator DG-2 during a loss of offsite power (LOOP). CS pump SI-3A, LPSI pump SI-1A and HPSI pumps SI-2A and SI-2C are powered from diesel generator DG-1 during a LOOP. During a DBA, CS pumps SI-3B and SI-3C and HPSI pump SI-2B are supplied through containment sump strainer SI-12A, and CS pump SI-3A and HPSI pumps SI-2A and SI-2C are supplied through containment sump strainer SI-12B following the recirculation actuation signal (RAS). The RAS is generated when the SIRWT level drops to the SIRWT low signal (STLS) setpoint and a concurrent pressurizer pressure low (PPLS) or containment pressure high (CPHS) is present.

3.1.2 CS System Design Basis Function

The present function of the CS system is to limit the containment pressure rise and reduce the leakage of airborne radioactivity from the containment by providing a means for cooling the containment following a LOCA. This system reduces the leakage of airborne radioactivity by effectively removing radioactive particulates from the containment atmosphere. Removal of radioactive particulates is accomplished by spraying water into the containment atmosphere. The particulates become attached to the water droplets which fall to the floor and are washed into the containment sump. The iodine removal capability during the first 30 days of the DBA (large break LOCA) is discussed in FCS USAR Section 14.15, *Safety Analysis – Loss-of-Coolant Accident* (Reference 6.4).

Pressure reduction is accomplished by spraying cool, borated water into the containment atmosphere which provides a means for cooling the containment atmosphere. Heat removal is accomplished by recirculating

and cooling the water through the shutdown cooling heat exchangers. The CS system is independent of the containment air cooling and filtering system. The CS system is designed for a heat removal capacity that is sufficient to maintain the peak containment pressure below the design limit as discussed in FCS USAR Section 14.16, *Safety Analysis - Containment Pressure Analysis*, (Reference 6.4).

The minimum required hydraulic performance for a CS pump is calculated based on the credited CS flow in the LOCA containment pressure analysis for the one-pump, one-header operating mode as discussed in FCS USAR Section 14.16, *Safety Analysis - Containment Pressure Analysis* (Reference 6.4). The CS system also limits the containment pressure rise during an MSLB event.

3.1.3 CS System Automatic Initiation

Containment spray operation is presently initiated by the same basic signals as safety injection, but in a different logic combination. The 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. Upon reaching low tank level, the RAS is initiated, automatically transferring the pump suction to the containment recirculation line inlet from the containment sump.

3.1.4 Containment Air Cooling and Filtering System (CACFS) Description

The design basis function of the containment air cooling and filtering system (CACFS) is discussed in FCS USAR Section 6.4 (Reference 6.5).

The CACFS consists of four air handling units, each with its own fan, a common plenum discharge system and instrumentation and controls. There are two types of units: 1) two containment air cooling and filtering (CACF) units with filtering capacity; and 2) two containment air cooling (CAC) units with no filtering capacity. Each train of the two-train CACFS consists of one CACF unit and one CAC unit. The arrangement of the equipment is shown in USAR Figure 6.4-1.

Each of the two CACF units is comprised of the following components (listed in order of flow sequence): inlet face dampers, baffle type moisture separators, media type mist eliminators, HEPA filters, charcoal filters and cooling coils, all contained in a single housing. Dampers between the charcoal filters and the cooling coils allow the filter banks to be bypassed during normal (i.e., non-accident) operation. The filter banks of each unit are split in two parallel and separate trains. The common exhaust flows from each train are drawn through coil banks by axial, air-over-motor fans and

discharged into a plenum. Backdraft dampers are installed in the duct sections downstream of the fans. Each unit was designed for an inlet air flow of 110,000 cubic feet per minute (cfm) when cooling the containment atmosphere at the DBA conditions of 60 psig, 288°F and 100 percent relative humidity to remove 140 x 10^6 Btu/hr.

The CAC units are similar in design to the CACF but do not include mist eliminators, face and bypass dampers, HEPA filters, and charcoal filters. Each unit is designed for an inlet air flow of 66,000 cfm to remove 70 x 10^6 Btu/hr at DBA conditions.

The CACF HEPA and charcoal filters are not currently credited for removing radioactive particulates from the containment atmosphere.

It should be noted that in industry practice and FCS documents, the terms "air cooling" and "fan cooling" have both been applied to the "air handling units" described above. For the purposes of this LAR submittal, "air cooling" will be used except when referring to text from the Branch Technical Position (BTP) CSB 6-1, *Minimum Containment Pressure Model for PWR ECCS Performance Evaluation* (Reference 6.19), which uses the term "fan cooling."

3.1.5 CACFS Design Basis Function

The design function of the CACFS is to limit the leakage of airborne activity from the containment in the event of a LOCA. This is accomplished by the removal of heat released to the containment atmosphere during the DBA to the extent necessary to initially maintain the structure below the design pressure and then reduce the pressure to near atmospheric. Leakage from containment is thereby restricted to within design limits. FCS presently credits the CACFS in the containment pressure analysis (USAR 14.16) for the MSLB event. The LOCA analysis does not credit the CACFS.

The CACFS also prevents the accumulation of hydrogen pockets by maintaining a continuous flow throughout the containment. The minimum number of air changes in restricted areas of the containment is one per hour which provides adequate mixing and sweeping of hydrogen. A lesser number of air changes are permitted in open top cells in which hydrogen will not tend to accumulate.

3.1.6 Containment Air Cooler Automatic Initiation

The containment air coolers are automatically started for LOCA mitigation. A CPHS and/or a PPLS initiates the following:

- VA-3A and VA-3B are started with PPLS OR CPHS via the sequencer.
- VA-7C and VA-7D are started with PPLS AND CPHS via the sequencer.
- On the cooling and filtering units, the face dampers open and the bypass dampers close.
- The component cooling water (CCW) system valves on the cooling coil supply and return lines receive actuation signals to open.

All four units are then operating in the emergency mode. If all normal power sources are lost and only one emergency diesel generator functions, one cooling unit fan and one cooling and filtering unit fan operate.

3.1.7 Changes to Current Operation of the CS Pumps, CACs and CACFs

Currently, two CS pumps (SI-3A and SI-3B) are automatically started by the CSAS. Containment spray pump SI-3C is connected to the electrical bus associated with diesel generator DG-2; however, it 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 CS system, none of the pumps will automatically start during a LOCA. The change will require that both the SGLS and the CSAS be initiated before the CS system is actuated in response to a MSLB event. An SGLS interlock to each CS pump (SI-3A and SI-3B) sequencer circuit will be added. The change does not impact the overall delay time in initiation of CS assumed in the MSLB containment pressure and temperature response analysis of record (AOR). No changes are being made to the containment air cooler operation or initiation by this proposed change.

3.1.8 Changes to Current Operation of the SGLS Actuation Logic

Currently the logic consists of four channels per steam generator (eight total) feeding one lockout relay for each generator (two total). The logic will be modified such that each channel for each steam generator will input to a matrix to actuate both lockout relays. This change will ensure that a single failure of the lockout relay will not prevent initiation of an SGLS signal during a MSLB accident.

3.2 Description of Analytical Methods, Applicable Standards, Data, and Results Justifying the Amendment

3.2.1 FCS Containment Analysis without Containment Spray

Various containment analyses were performed to assess the impact of eliminating the automatic containment spray actuation following a LOCA at FCS. The analyses were performed assuming the worst single active failure of diesel generator No. 1 (DG-1) to start, which results in only one train of

engineered safeguards and one CCW pump available at the start of the accident. These analyses were performed using GOTHIC 7.2a to evaluate short and long-term containment analysis without CS. The containment air coolers were included in the model as the only active containment energy removal mechanism.

Short-term containment analyses, summarized in Section 3.1 of the attached Summary Report (Reference 6.10), were performed to evaluate peak pressure and to select the limiting hot and cold leg break cases for evaluation of the long-term phenomena. Given the limiting breaks from the short-term analysis, long-term "Mass and Energy Release" (MER) cases were performed to generate data beyond the time of RAS, which is summarized in Section 3.2 of Reference 6.10. The revised MER data was used in various containment analyses that implement the containment analysis methodology changes.

The LOCA containment vapor temperature response without CS is overly conservative using the existing containment analysis methodology defined in Reference 6.11. To generate a smoother code-to-code transition that remains conservative, methodology changes were required. Those changes, used in many of the analyses that support operation without containment spray for FCS, are described in Reference 6.10, along with the results of an assessment of the methodology changes on both hot and cold leg break analyses with and without CS. NRC approval of the methodology changes (Attachment 6) is required prior to implementation of the results of these analyses.

Analyses were performed to evaluate various aspects of the post-LOCA containment response for issues such as containment peak pressure, containment pressure reduced to 50% of the peak in 24 hours, raw water (RW)/CCW peak temperature and RW maximum temperature limit, which were found to be acceptable. Analyses were performed to generate long-term containment vapor and sump liquid temperature data, which are being used to evaluate and to determine the necessary actions to maintain equipment qualification (EQ).

3.2.2 Radiological Consequences Analysis Summary

Amendment No. 201 to the FCS Operating License No. DPR-40 replaced the radiological source term used in the design basis site boundary and control room dose consequence analyses with alternative source terms (AST) pursuant to 10 CFR 50.67, Standard Review Plan (SRP) 15.0.1, and Regulatory Guide (RG) 1.183. This LAR submittal revises the radiological consequences to address the potential impact on the estimated LOCA dose consequences at the exclusion area boundary (EAB), low population zone

(LPZ), control room (CR), and technical support center (TSC) due to modifying the CS system actuation logic to preclude automatic start of the CS pumps for a LOCA. The design changes for the dose consequences of the FCS LOCA associated with this proposed change include the elimination of the use of CS and utilization of the containment HEPA filters for post-LOCA fission product removal.

In accordance with current licensing basis, the acceptance criteria for the EAB and LPZ doses for the LOCA are based on 10 CFR 50.67 and Section 4.4, Table 6 of RG 1.183 (also noted in Table 1 of SRP 15.0.1):

- *An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem TEDE.*

- *An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE.*

The acceptance criterion for the CR dose is based on 10 CFR 50.67:

- *Adequate radiation protection is provided to permit occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.*

The acceptance criterion for the TSC dose is based on NUREG-0737, Supplement 1, Section 8.2.1.f, which requires that plant design "assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body or its dose equivalent to any organ." The 5 rem whole body (or its dose equivalent to any organ) guideline is the same guideline specified for a CR operator in 10 CFR 50, Appendix A, GDC 19. Since 10 CFR 50.67 supersedes GDC 19 when licensing a facility using ASTs for evaluating design basis accidents, the updated guideline for the CR in 10 CFR 50.67 is applied to the TSC for purposes of demonstrating habitability.

The input data was based on the guidance in RG 1.183, with some inputs coming directly from the RG. Additionally, input data came from the FCS TS, USAR, Design Basis Documents, and plant procedures. For added conservatism, input data was adjusted in the direction that resulted in the worst impact on the analyses.

The estimated doses to the public and to the control room reflect the design changes for the dose consequences of the FCS LOCA (i.e., elimination of the use of containment sprays and utilization of the containment HEPA filters for post-LOCA fission product removal) associated with this modification. The radiological consequences of the remaining design basis accidents are not impacted by this application.

The radiological analyses and evaluations demonstrate that modifying the containment spray system actuation logic to preclude automatic start of the containment spray pumps for a LOCA will not impact compliance with applicable regulatory requirements. The LOCA has been evaluated using the guidance provided in RG 1.183. The estimated dose consequences at the EAB, LPZ, and CR remain within the acceptance criteria of 10 CFR 50.67, as supplemented by RG 1.183 and SRP 15.0.1. In addition, the assessment also demonstrates that the dose consequences in the TSC remain compliant with regulatory guidance provided in Supplement 1 of NUREG-0737.

3.3 Results Justifying the Amendment and Technical Details Supporting Safety Arguments

3.3.1 Benefits and Reliability Discussion

Following the implementation of the proposed change, CS pumps will no longer automatically start during a LOCA. Both an SGLS and CSAS will be required for the CS pumps (SI-3A and SI-3B) to start. Changing the logic for actuation of the CS system will not adversely impact its availability, reliability or capability.

The following benefits are also associated with the proposed change:

- Increased NPSH available ($NPSH_A$) for SI system

  Following RAS, if a sufficient amount of debris accumulates on the containment sump strainer, the debris bed would reach a critical thickness at which the head loss across the debris bed could exceed the NPSH margin required to ensure the successful operation of the HPSI pumps in the recirculation mode (the LPSI pumps are shut off following a RAS). A loss of NPSH margin for the pumps as a result of the accumulation of debris on the recirculation sump strainer, referred to as sump clogging, could result in degraded pump performance and eventual pump failure.

  By eliminating CS pump operation during a LOCA, NPSH margin for the high pressure safety injection (HPSI) pumps during the recirculation phase of a LOCA is enhanced by reducing the head loss and hydraulic resistance through the containment sump strainers when the HPSI pumps are

operating in the recirculation mode. The enhancement in the NPSH performance will be realized due to reduced transport of debris to the strainer resulting in a reduction in the pressure drop across the strainer and a reduction in piping head loss. This will provide additional $NPSH_A$ for the HPSI pumps taking suction from the containment sump.

- Delayed time to RAS

The depletion rate of the SIRWT is a direct function of the flow rate through the HPSI, LPSI, and CS pumps. CS pump flow rate is a significant contributor to the total pre-RAS flowrate from the SIRWT. Therefore, not operating CS pumps during a LOCA would be beneficial (i.e., it would result in a delay in RAS actuation). A delay in RAS would allow any accident-generated debris transported to the containment sump pool additional time to settle to the floor and thus be less likely to be drawn to containment sump strainers following RAS. This reduces the probability of sump strainer clogging. In addition, by minimizing the use of CS by eliminating automatic spray initiation, significantly more water can be delivered to the core during the injection phase of the LOCA, which can be viewed as a significant safety enhancement.

- Reduced debris transport

The amount of debris collected on the sump strainers is a function of strainer size, flow through the strainers, and overall inflow of debris into the containment sump area. A greater volumetric flow will result in a higher rate and probability of debris deposition on the sump strainers due to higher approach velocity in the sump pool, thereby increasing the strainer head loss. A reduction in flow rate, by not operating the CS pumps, will reduce the rate of sump strainer debris buildup and reduce the probability of strainer blockage, because the CS pump flow rate is a significant contributor to the total flowrate through the sump strainers. Sump pool turnover time is also increased, allowing more time for debris that bypassed the strainer to settle before reaching the strainer for the second time.

- Reduction in debris washed into the containment sump pool

By spraying down containment, debris from all areas is washed into the containment sump pool. By not operating the CS pumps only debris in the vicinity of the break will be washed into the pool. This is advantageous because less debris in the containment sump pool means less debris that can possibly build up on the strainer.

- Limitation of total surface area of containment sump strainers

  One of the planned actions to address GL 2004-02 requirements is increasing the total surface area of the containment sump strainers. Larger strainer modules were installed during the 2006 refueling outage. The reduced flow associated with the proposed change in this LAR will minimize the need to add additional strainer modules by reducing strainer approach velocity and head loss. This is advantageous because the available floor space to expand the strainers is limited.

3.3.2 Impact on Design and Licensing Basis

The following sections discuss the impact of the proposed change on the design and licensing basis for the plant:

- Operator Actions

  The proposed change, which will support the plant modification that will change the logic for actuation of the CS system, will not require any additional operator actions in response to the MSLB and LOCA design basis events.

- Human Factors

  The containment air coolers will auto start on receipt of a CSAS for LOCA mitigation. The CS pumps will auto start on receipt of a CSAS and SGLS for mitigation of high containment pressure and temperature following an MSLB. For both functions, no operator action is required. The systems will respond as required for event mitigation.

  The engineered safeguards systems will operate in the same manner as the current configuration with the exception of the CS system. Under the new configuration, the CS system will only actuate when both an SGLS and a CSAS signal are present. The system will no longer automatically operate for the mitigation of a LOCA event. Manual operation of the Train A and Train B safeguards actuation will no longer initiate CS unless an SGLS signal is present. In the unlikely event that CS is needed and has not been actuated automatically, the operators can start the desired pump via the breaker control switch from the main control room.

- Minimum/Maximum Safeguards

  Engineered safeguards is the designation given to systems and components provided to protect the public and plant personnel by minimizing both the extent and the effects of an accidental release of

radioactive fission products from the reactor coolant system (RCS), particularly those following a LOCA up to and including a double ended rupture of the largest reactor coolant pipe. These safeguards function to localize, control, mitigate, and terminate such accidents and to hold offsite environmental exposure levels within the guidelines of 10 CFR 50.67.

The systems function to cool the core, limit the magnitude and duration of the pressure transient within the containment vessel following a LOCA, and provide long term post-accident cooling. Such an accident and a gross release of fission products could occur only as the result of an incredible series of failures and malfunctions.

Engineered safeguards equipment includes engineered safety features (ESF) systems, essential auxiliary support systems, and engineered safeguards controls and instrumentation.

Full (or maximum) safeguards include all of the equipment defined as engineered safeguards equipment. Full safeguards would be available during a DBA if no single failures were to occur. In most accident analyses for FCS, the limiting single active failure is the loss of an engineered safeguards electrical train. For this single active failure concurrent with a DBA, an entire train of engineered safeguards would be unavailable for mitigation of the initiating event; however, the other redundant train of engineered safeguards would be available. The single train of engineered safeguards is termed as minimum safeguards.

The CS system is an engineered safeguards system. The current plant configuration has two CS pumps for maximum safeguards and one CS pump for minimum safeguards. This system removes heat by spraying cool borated water through the containment atmosphere. Heat is transferred to the CCW system through the shutdown heat exchangers.

The analysis of emergency core cooling system (ECCS) performance following a LOCA is done using the minimum containment pressure that would be present during the accident. Minimizing containment pressure following a LOCA adds conservatism to the ECCS performance since there is a direct dependence of core flooding rate on containment pressure.

Analyses were performed to evaluate various aspects of the post-LOCA containment response for issues such as containment peak pressure, containment pressure reduced to one-half the peak in 24 hours, RW/CCW peak temperature and RW maximum temperature limit, which were found to be acceptable. Analyses were performed to generate long-term

containment vapor and sump liquid temperature data. The impact of modifying the CSAS on the containment pressure and temperature, RW/CCW temperature, methodology changes, and the sump temperature are discussed in detail in the Accident Analysis Impact Evaluation section of this LAR.

Based on these evaluations, it is acceptable to modify the CS system logic to actuate only when an SGLS and CSAS are initiated, therefore, not to operate during a LOCA. Following the modification (EC 40070), the minimum safeguards case for a LOCA will include one train of CACFS and the maximum safeguards case for a LOCA will include two trains of CACFS.

3.3.3 Accident Analysis Impact Evaluation

Summary of Results

Evaluation of the accident analyses determined that:

- One train of CACFS, including a CAC and a CACF unit, is capable of maintaining the containment pressure below containment design pressure and provide cooling to reduce the long-term post-accident containment pressure and temperature. A peak containment pressure of 54.5 psig, peak containment vapor temperature of 277.4°F, and containment pressure at 24 hours of 10.6 psig were obtained for the hot leg break. A peak containment pressure of 53.4 psig, peak containment vapor temperature of 275.8°F, and containment pressure at 24 hours of 9.7 psig were obtained for the cold leg break. These results demonstrate that containment pressure was reduced to less than one-half the peak within 24 hours without CS.

- The LBLOCA analysis for minimum containment pressure impact on ECCS performance remains bounding without CS pumps operating. (See LOCA Minimum Containment Pressure Analysis Impact section.)

- The NPSH margin for ECCS equipment taking suction from the containment sump is improved without CS. Without CS, HPSI is the only system taking suction from the containment sump. Regardless, a LOCA containment analysis was performed to evaluate long-term sump temperature, specifically for use providing input to NPSH calculations for equipment taking suction from the containment sump. The analysis confirms that the 8.99 ft of subcooling/overpressure presently credited for FCS for SI pump NPSH is adequate and available with the proposed modification to the CS system actuation logic.

- The limiting hot leg and cold leg break cases from the CCW design evaluation, USAR Section 9.7.5 (AOR), were run without containment spray. The results of the analysis, documented in Reference 6.10, indicate peak CCW temperature at the containment air cooler secondary inlet produced temperatures of 153°F and 156°F for the limiting hot leg and cold leg pump suction breaks, respectively. These temperatures are only slightly above the same cases with CS and remain less than the maximum allowed temperature of 160°F. The RW system inlet temperature assumed in this analysis was unchanged from the value used in the AOR and remained at 90°F. The MSLB transient results, in the AOR, remain bounding. Therefore, operation without CS is acceptable with respect to RW/CCW operation.

- The radiological analyses and evaluations demonstrate that one train of CACFS is capable of maintaining the LOCA dose consequences at the EAB, LPZ and CR within the acceptance criteria of 10 CFR 50.67 and the dose consequences in the TSC remain compliant with regulatory guidance provided in Supplement 1 of NUREG-0737.

Discussion of Analyses

LOCA Minimum Containment Pressure Analysis Impact

SRP Section 6.2.1.5, Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies, provides the guidelines for conservatively evaluating ECCS performance following a LOCA. As stated in the SRP, following a LOCA, the ECCS supplies water to the reactor vessel to reflood, and thereby cool the reactor core. The SRP continues by stating that the core flooding rate is governed by the capability of the ECCS water to displace the steam generated in the reactor vessel during the core reflooding period and that for pressurized water reactors (PWRs), there is a direct dependence of core flooding rate on containment pressure, (i.e., the core flooding rate will increase with increasing containment pressure). Therefore, to provide a conservative evaluation of ECCS performance following a LOCA, the minimum containment pressure that could exist during the period of time until the core is reflooded following a LOCA must be included in the analysis. Note that this minimum containment pressure analysis done in conjunction with ECCS performance evaluation differs from the containment functional performance analysis, in that the conservatisms and margins are taken in opposite directions in the two cases. Therefore, the minimum containment pressure analysis required for ECCS performance evaluations is not intended to be representative of the peak containment pressure in the event of a LOCA.

SRP Section 6.2.1.5, which is associated with BTP CSB 6-1 (Reference 6.19), delineates the calculational approach that should be followed to assure a conservative prediction of the minimum containment pressure that is used in the LOCA analysis. BTP CSB 6-1 states that for spray and fan cooling systems, the operation of all ESF containment heat removal systems operating at maximum heat removal capacity should be assumed, (i.e., all CS trains operating at maximum flow conditions and all emergency fan cooler units operating).

In general, FCS is not licensed to the SRP. However, the FCS LBLOCA peak cladding temperature (PCT) calculation is performed in accordance with the SRP Section 6.2.1.5 and the BTP (Reference 6.19).

The LBLOCA analysis is based on a realistic LOCA methodology. The RLBLOCA methodology is documented in Framatome ANP, Inc., EMF-2103 (P) (A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003. The ECCS is designed such that its calculated cooling performance following a postulated LOCA conforms to the criteria specified in 10 CFR 50.46. ECCS performance has been calculated for a number of postulated LOCAs of different sizes and other properties sufficient to provide assurance that the entire spectrum of postulated LOCAs is covered. A complete LBLOCA analysis for operation of the FCS at 1525 MWt was performed for Cycle 24 by AREVA. A case set comprising 59 transient calculations was performed by sampling key parameters related to LOCA phenomena and plant operation. The limiting PCT is 1636°F. The worst single failure criterion for LBLOCA is met by assuming either the loss of one LPSI pump or the loss of a diesel generator.

The failure of one LPSI pump scenario is impacted by the proposed change, since it currently assumes that all three CS pumps are operating. However, as discussed above, the more containment cooling components that are assumed operating, the more conservative the evaluation is. Therefore, if fewer containment cooling components are operating, (i.e., no CS pumps), then the performance of the ECCS is actually enhanced due to lower containment cooling capacity which results in higher containment pressure which equates to higher ECCS core reflood rate in response to a LOCA. Therefore, following the implementation of the proposed change, the current ECCS performance analysis will remain bounding and conservative.

Containment Pressure Analysis

The containment building and associated penetrations are designed to withstand an internal pressure of 60 psig at 305°F, including all thermal loads resulting from the temperature associated with this pressure, with a leakage rate of 0.1 percent by weight or less of the contained volume per 24

hours. To maintain containment pressure and temperatures within design limitations, and assure that the release of fission products to the environment following a design basis accident will not exceed regulatory guidelines, the following engineered safety systems are incorporated into the FCS design:

a. CS system with redundant features to remove heat from the containment and to provide for reduction of airborne particulate radioactivity following a LOCA; and

b. CACFS with redundant features to remove heat from the containment atmosphere.

The CACFS operates independently of the CS system to remove heat from the containment atmosphere. The CACFS consists of two redundant trains; each train with one air cooling and filtering unit and one air cooling unit, for a total of four cooling units. If all normal power sources are lost and one diesel generator fails to function, one train of CACFS will operate.

The FCS licensing basis assumes two CS pumps operating during a MSLB. Operation of the CACFS with a heat removal capacity of 200×10^6 BTU/hr is credited in the MSLB containment pressure analysis. The existing MSLB analysis remains bounding for the changes in the logic for the actuation of the CS system.

The present FCS licensing basis credits only one of the three CS pumps to limit the containment pressure to below the design value for a LOCA.

The proposed modification will eliminate the automatic CS actuation for a LOCA and will credit the one train of containment air coolers as the only active containment energy removal mechanism.

LOCA Containment Pressure Analysis Impact

Containment analyses were performed to assess the short-term impact of the LOCA blowdown and performance of containment cooling systems. All analyses were performed assuming loss of offsite power and a single active failure minimizing containment heat removal. This assumed single active failure of DG-1 results in only one CCW pump operating at the beginning of the accident. The second CCW pump is loaded onto the swing bus, as described in USAR Section 9.7.4.3, per direction provided in the emergency operating procedures within 30 minutes of the accident initiation. The acceptance criterion for this analysis was the containment design pressure of 60 psig. Limiting cases were identified for evaluation of long-term issues. The results of that short-term analysis established the limiting hot and cold leg break configurations for subsequent long-term analyses. Because of

possible changes in the peak pressure when crediting only containment air coolers, short-term containment analysis cases without CS and with containment air coolers were analyzed. (Reference 6.10) The containment analysis used the short-term MER data generated with RELAP5/MOD2-B&W computer code. The short-term containment analysis without CS demonstrated an acceptable containment peak pressure response.

The limiting peak containment pressure is 54.5 psig at a time of 13.2 seconds using the MER of a double-ended hot leg break with a discharge coefficient of 0.6, LOOP, a single active failure resulting in minimum ECCS injection, and a minimum containment backpressure. The limiting peak containment vapor temperature is 277.4°F at a time of 13.0 seconds, from the same hot leg break case. A limiting peak containment pressure for cold leg breaks of 53.4 psig at a time of 165.9 seconds using the MER for a double-ended pump suction break with a discharge coefficient of 0.8, LOOP, a single active failure resulting in minimum ECCS injection, and a minimum containment backpressure. The corresponding peak containment vapor temperature of 275.8°F at a time of 20.1 seconds was observed for the same case. A complete set of results for the short-term LOCA containment analysis is shown in Table 1 of Reference 6.10.

The short-term analysis demonstrates that the peak containment pressure following a LOCA without CS meets the acceptance criterion of 60 psig. A majority of the peak pressures and temperatures occur prior to initiation of cooling to the containment building. Thus, the spectrum analysis confirmed, as anticipated, that the hierarchy of limiting hot and cold leg breaks remained unchanged regardless of whether containment cooling was modeled or not.

Containment Analysis Methodology Changes

The approved containment analysis methodology defined in Reference 6.11 implements a code-to-code transition for MER from RELAP5 to GOTHIC after core quench. A significant quantity of energy is stored in the primary and secondary heat structures and secondary fluid that dissipates over time to the RCS liquid and vapor and ultimately to the containment through the break as liquid and/or vapor. The methodology in Reference 6.11 stipulates that the stored energy is transferred to the RCS liquid and containment vapor at rates calculated as discussed in Section 3.3.2 of Reference 6.10.

Preliminary analyses without CS demonstrated that containment vapor temperature increased dramatically after "Transition" as a result of non-mechanistic direct heat addition to the containment vapor space without spray drops to absorb latent heat of vaporization. This behavior, although conservative, is unrealistic for many long-term containment analyses. Therefore, methods were pursued that would give a conservative, yet more

reasonable containment vapor temperature response. Two method changes were evaluated separately and in combination in the areas of mass and energy release (MER) and stored energy dissipation.

The first approach was to delay transition to the time of RAS, which would delay the time and reduce the magnitude of the post-transition peak vapor temperature. The RELAP5 simulation would be extended to a time beyond RAS. The MER rates from this system analysis would be used in the containment analysis up to RAS. After transition at RAS, the GOTHIC code would perform the MER rates.

The second approach was to calculate the stored energy dissipation based on the rate of heat transfer to the vapor and liquid phases instead of a change in stored energy and control volume void fraction. Further discussions of the existing and alternate methods of performing these calculations are included in Sections 3.3.2 and 3.3.3 of Reference 6.10, respectively.

These methodology changes were evaluated with and without CS as discussed in Section 3.3.4 of Reference 6.10.

Long-Term Containment Pressure and Vapor Temperature Analysis

A LOCA containment analysis was performed to evaluate long-term pressure and vapor temperature response. Specifically, the results of the analysis are evaluated to ensure that the containment pressure decreases to less than half of the peak pressure within 24 hours. Additionally, the containment vapor temperature is generated for evaluation of electrical equipment qualification (EEQ). The limiting hot leg and cold leg break long-term cases with CS documented were analyzed without CS. The analysis was performed with the time of transition at RAS and the stored energy dissipation rates calculated using both the approved and alternate methods.

A peak containment pressure of 54.5 psig, peak containment vapor temperature of 277.4°F, and containment pressure at 24 hours of 10.6 psig were obtained for the hot leg break. A peak containment pressure of 53.4 psig, peak containment vapor temperature of 275.8°F, and containment pressure at 24 hours of 9.7 psig were obtained for the cold leg break.

These results demonstrate that containment pressure was reduced to less than half of the peak within 24 hours without CS. The containment vapor temperature results are bounded by the existing temperature profile in the AOR (USAR Figure 14.16-10).

Long-Term Containment Sump Temperature Analysis

In general, the NPSH margin for ECCS equipment taking suction from the containment sump is improved without CS. Without CS, HPSI is the only system taking suction from the containment sump. Regardless, a LOCA containment analysis was performed to evaluate long-term sump temperature, specifically for use providing input to the NPSH calculations for equipment taking suction from the containment sump. The sump temperature data generated by this analysis will be used for other evaluations. The limiting hot leg and cold leg break long-term cases with CS documented in the AOR were analyzed without CS. The analysis was performed with the time of transition at RAS and the stored energy dissipation rates calculated using the existing methods. The resulting sump temperature, subcooling head, and overpressure head as a function of time after RAS from the long-term containment sump temperature analysis are included in Reference 6.10 for the limiting hot leg and cold leg breaks. The analysis confirms that the 8.99 feet of subcooling/overpressure presently credited by FCS for SI pump NPSH is adequate and available with the proposed change to the plant operation.

CCW/RW Peak Temperature Analysis

The peak temperature of the CCW system and limit on the RW system maximum temperature are a function of the containment vapor temperature and humidity for the containment air coolers throughout the transient and sump temperature for the shutdown cooling (SDC) system after RAS. The post-LOCA transient behavior generates two peaks in load on the CCW and RW systems with the first occurring after containment peak vapor temperature and the second occurring after RAS. The pre-RAS peak CCW temperature with CS is more limiting for FCS than the post-RAS peak based on the analysis documented in the AOR.

A change in containment response occurs because of the proposed change to eliminate actuation of CS following a LOCA and the unavailability of SDC after RAS without CS. Without CS to remove energy from the containment vapor space, the vapor temperature could potentially be higher for a longer period of time which would place a greater heat load on the CCW system. That change could cause a higher pre-RAS peak CCW temperature. The post-RAS peak in containment vapor temperature is lower than the pre-RAS peak regardless of spray operation. Therefore, the containment air cooler heat load is lower post-RAS. Additionally, the post-RAS heat load is reduced by the lack of heat removal from SDC. Based on the aforementioned system response, the pre-RAS CCW temperature peak will be limiting regardless of CS operation.

Based on the results of the methodology change evaluation discussed in Reference 6.10, the post-transition containment vapor temperature response, which is the major contributor to containment air cooler heat load, decreases smoothly after transition with either transition at RAS or the alternate method of calculating the stored energy dissipation rates. Therefore, only the pre-transition response need be evaluated for CCW heat load.

The limiting hot leg and cold leg break cases from the RW/CCW AOR were run without CS. The results of the analysis indicate peak CCW temperature at the containment air cooler secondary inlet produced temperatures of 153°F and 156°F for the limiting hot leg and cold leg pump suction breaks, respectively. These temperatures are only slightly above the same cases with CS and remain less than the maximum allowed temperature of 160°F. The RW system inlet temperature assumed in this analysis was unchanged from the value used in the AOR and remained at 90°F. The CCW peak temperature of 159.95°F established in the MSLB AOR remains bounding.

Therefore, operation without CS is acceptable with respect to RW/CCW operation.

Radiological Consequences

LOCA

The FCS LOCA analysis, summarized in USAR Section 14.15, demonstrates that, in the event of a postulated LB LOCA, the limits of 10 CFR 50.67, as supplemented by RG 1.183, are met assuming a worst-case single failure.

OPPD had identified three activity release paths following a LOCA: (1) containment pressure relief line release, (2) containment leakage, and (3) ESF system leakage (including SIRWT back leakage).

(1) In accordance with the current licensing basis, it is assumed that the containment pressure 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, assumed to be at TS levels, is released to the containment at T = 0 hours. It is conservatively assumed that 100% of the volatiles are instantaneously and homogeneously mixed in containment atmosphere. Containment pressurization (due to the RCS mass and energy release), combined with the pressure relief line cross-sectional area, results in a 600 scfm release of containment atmosphere to the environment over a period of 5 seconds (i.e., prior to containment isolation) via the auxiliary building ventilation stack. Since the release is isolated

within 5 seconds after the LOCA, (i.e., before the onset of the gap phase release assumed to be at 30 seconds), no fuel damage releases are postulated. The chemical form of the iodine released from the RCS is assumed to be 97% elemental and 3% organic.

(2) The inventory of fission products in the reactor core available for release via containment leakage represents a conservative equilibrium reactor core inventory of dose significant isotopes, assuming maximum full power operation at 1.02 times the current licensed thermal power, and taking into consideration fuel enrichment and burnup.

In accordance with RG 1.183 and the current licensing basis, the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. Two fuel release phases are considered, the gap release, which begins 30 seconds after the LOCA and continues for 30 minutes, and the early in-vessel release, which begins 30 minutes into the accident and continues for 1.3 hours, at which time the release into the containment is assumed to terminate.

The activity transport model takes credit for aerosol removal by the HEPA filters in the containment air cooling and filtering (CACF) units and credits plateout on the surface of the water in the sump for elemental iodine removal. The maximum decontamination factor (DF) for elemental iodine is based on SRP 6.5.2 and is limited to a DF of 200. There is no regulatory based maximum DF for aerosol removal via use of HEPA filters.

(3) With the exception of noble gases, all the fission products released from the core in the gap and early in-vessel release phases are assumed to be instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the fuel. Consistent with current licensing basis, a minimum sump volume is utilized in this analysis. With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. Due to their similar characteristics, the ESF and SIRWT leakage are modeled together as one release. The combined ESF and SIRWT leakage is set by TS to 3800 cc/hr. The analysis uses 7600 cc/hr as the ESF/SIRWT leakage rate, which includes a factor of 2. The subsequent environmental radioactivity release is discussed below:

In accordance with current licensing basis, equipment carrying sump fluids and located outside containment is postulated to leak at twice the expected value into the auxiliary building. Leakage is expected starting at initiation of the recirculation mode, which is based on maximum ESF. (Note that due to the long term nature of this release, minor variations in the start time of this release will not significantly impact the resultant doses.) Since the temperature of the recirculation fluid is less than 212°F, 10% of the halogens associated with this leakage become airborne and are exhausted (without mixing and without holdup) to the environment via a release point in the auxiliary building, with the most unfavorable dispersion characteristics relative to the control room and the TSC intakes (i.e., the auxiliary building ventilation stack and the auxiliary building fresh air intake, respectively). The chemical form of the iodine released from the sump water is 97% elemental and 3% organic. No credit is taken for the ESF filter system.

Aerosol Removal by CACF HEPA Filters

The maximum potential bypass leakage following a DBA (with a conservative safety factor of 2 applied to bound filter performance) is utilized to support the 50% filter efficiency used in the dose consequence analysis. The potential for bypass leakage would be primarily dominated by leakage through cracks in the gasketing material or through holes in the filter media itself. Estimates of maximum bypass leakage due to gasket deterioration is made assuming maximum credible failures of the gasket (to establish worst case crack leakage), and the equivalent size of a hole (orifice) in filter media is calculated which confirms that a hole of this size in the filter media can be detected by visual inspection.

Elemental Iodine Plateout

The surface of the pool water is utilized as the renewable wet surface. The issue of surface renewal for the pool water has to be addressed only up to the time a sump water pH of 7 or greater can be established. Prior to that time frame, surface renewal occurs based on the following: (1) the ECCS injection water from the break location entering the pool as a stream (due to temperature differences) and entraining the surface pool water setting up convection and momentum driven currents promoting surface renewal, and (2) the containment is structurally designed to direct water to the sump, thereby avoiding water holdups and isolated water sections, which extends the area of influence for the convective and momentum driven flow even though obstructions exist. As added conservatism, half the calculated surface area is utilized for elemental iodine plateout.

Mixing between regions above and below the operating floor is assumed for the duration of the accident and is based on the minimum CACFS flow rate and the maximum containment free volumes above and below the operating floor. Radioactivity is assumed to leak from the area below and above the operating floor to the environment through cracks and penetrations in the containment wall/steel liner, at the containment TS leak rate for the first day and half that leakage rate for the remaining twenty-nine days.

Equipment Qualification

GOTHIC computer code is used to determine the bounding temperature profile associated with environmental equipment qualifications (EEQ) as discussed in USAR Section 14.16.1. The limiting hot leg and cold leg break long-term cases with CS documented were analyzed without CS to determine the impact on the LOCA EEQ peak pressure and temperature and time dependent plot for temperature. The results show that the AOR documented in USAR Section 14.16 remains bounding.

In addition, a preliminary analysis of the SI pump-rooms' heat-up calculations, which are used for evaluation of EEQ of auxiliary building components, shows that post-LOCA temperatures with the proposed plant configuration remain below the 164°F limiting temperature in the current calculation. The final results will be used to evaluate the electrical equipment qualification and ensure that the existing analyses are bounding or that appropriate actions are taken to maintain the equipment qualification for the new conditions. This final analysis is scheduled to be completed by October 1, 2007.

Therefore, the equipment will remain qualified to operate in the accident environment.

Supporting Analyses

Additionally, a review of other analyses that might be impacted by the elimination of the automatic CS actuation was performed. This review determined that none of the other analyses are adversely impacted. Some of these analyses will be updated as part of the modification package (EC 40070) that implements the proposed change. These updates are anticipated to produce results which are acceptable under 10 CFR 50.59 screening and will not require NRC review/approval.

Summary

Evaluation of the accident analyses determined that:

- One train of CACFS (one CAC unit and one CACF unit) is capable of maintaining the containment pressure below containment design pressure and provides cooling to reduce the long-term post-accident containment pressure and temperature. A peak containment pressure of 54.5 psig, peak containment vapor temperature of 277.4°F, and containment pressure at 24 hours of 10.6 psig were obtained for the hot leg break. A peak containment pressure of 53.4 psig, peak containment vapor temperature of 275.8°F, and containment pressure at 24 hours of 9.7 psig were obtained for the cold leg break. These results demonstrate that containment pressure was maintained below the allowable 60 psig pressure and was reduced to less than half the peak within 24 hours without CS.

- The LBLOCA analysis for minimum containment pressure impact on ECCS performance remains bounding without CS pumps operating.

- The NPSH margin for ECCS equipment taking suction from the containment sump is improved without CS. Without CS, HPSI is the only system taking suction from the containment sump. Regardless, a LOCA containment analysis was performed to evaluate long-term sump temperature, specifically for use providing input to NPSH calculations for equipment taking suction from the containment sump. The analysis confirms the 8.99 feet of subcooling/overpressure presently credited for FCS is available with the new design configuration in this modification to the CS system activation logic.

- The limiting hot leg and cold leg break cases from the CCW/RW AOR were run without CS. The results of the analysis documented in Reference 6.10 indicate peak CCW temperature at the containment air cooler secondary inlet produced temperatures of 153°F and 156°F for the limiting hot leg and cold leg pump suction breaks, respectively. These temperatures are only slightly above the same cases with CS and remain less than the maximum allowed temperature of 160°F. The RW system inlet temperature was unchanged and remained at 90°F. The MSLB transient results, in the AOR, remain bounding. Therefore, operation without CS is acceptable with respect to RW/CCW operation.

- The radiological analyses and evaluations demonstrate that one train of CACFS is capable of maintaining the LOCA dose consequences at the EAB, LPZ and CR within the acceptance criteria of 10 CFR 50.67 and the

dose consequences in the TSC remain compliant with regulatory guidance provided in Supplement 1 of NUREG-0737.

As discussed above, changes to the CSAS logic do not adversely affect the operation of the plant and do not result in unbounded conditions for the plant. The change will enhance the NPSH margins during a DBA by reducing the head loss and hydraulic resistance through the containment sump strainers when the HPSI pumps are operating in the recirculation mode. The enhancement in the NPSH performance will be realized due to reduced transport of debris to the strainer resulting in a reduction in the pressure drop across the strainer and a reduction in piping head loss. This will provide additional margin for the NPSH available ($NPSH_A$) for the HPSI pumps taking suction from the containment sump, increase the amount of water delivered to the core during the injection phase of a LOCA, and will increase the time to the initiation of a recirculation actuation signal (RAS).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 Regulations

The following regulations are directly applicable to this License Amendment Request (LAR):

- 10 CFR 50.46, "Acceptance Criteria For Emergency Core Cooling Systems For Light –Water Nuclear Power Reactors" which requires in part that the calculated maximum fuel element cladding temperature shall not exceed 2200°F following a postulated LOCA.

- 10 CFR 50.67, "Accident Source Term," December 1999, as clarified per the additional guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," USNRC, Office of Nuclear Regulatory Research, July 2000. 10 CFR 50.67 requires that the evaluated radiological consequences for design basis accidents demonstrate with reasonable assurance that any individual located at the Exclusion Area Boundary or Low Population Zone would not receive a radiation dose in excess of 0.25 Sv total effective dose equivalent, and that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv total effective dose equivalent for the duration of the accident.

FCS was licensed for construction prior to May 21, 1971 and is committed to the draft General Design Criteria (GDC). The draft general design criteria are contained in Appendix G of the FCS USAR. The preliminary design criteria which relate to this LAR are:

1. FCS Design Criterion 10, "Containment." This criterion is similar to 10 CFR Part 50, Appendix A, GDC 16, "Containment Design." FCS Design Criterion 10 states, "Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a 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." This request does not impact the compliance with FCS Design Criterion 10, as no change to the containment structure or its supporting systems were proposed as part of this LAR. The increase in surveillance requirements to test penetrations previous active during a LOCA is not considered a change to the containment structure.

2. FCS Design Criterion 12, "Instrumentation and Control Systems." This criterion is similar to 10 CFR Part 50, Appendix A, GDC 13, "Instrumentation and Control." FCS Design Criterion 12 states that instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges." This request does not impact the compliance with FCS Design Criterion 12, as no changes to instrumentation or controls were proposed as part of this LAR.

3. FCS Design Criterion 15, "Engineered Safety Features Protection System." This criterion is similar to 10 CFR Part 50 Appendix A, GDC 20, "Protection System Functions." FCS Design Criterion 15 states that "protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features." Steam generator low pressure was added to the protective systems sensing accident conditions in the response to this criterion. FCS Design Criterion 15 is met by the proposed change. The revised response to FCS Design Criterion 15 requires NRC approval.

4. FCS Design Criterion 17, "Monitoring Radioactivity Releases." This criterion is similar to 10 CFR Part 50, Appendix A, GDC 64, "Monitoring Radioactivity Releases." FCS Design Criterion 17 states, "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." This request does not impact compliance with FCS Design Criterion 17, as no changes to the means used to monitor radioactive releases were proposed as part of this LAR.

5. FCS Design Criterion 41, "Engineered Safety Features Performance Capability." This criterion is similar to 10 CFR Part 50, Appendix A, GDC 50, "Containment Design Basis." FCS Design Criterion 41 states, "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 a failure of a single active component." Clarification to the redundancy between and within systems was added in the response to this criterion. The FCS Design Criterion 41 is met by this LAR change. The revised response to FCS Design Criterion 41 requires NRC approval.

6. FCS Design Criterion 49, "Containment Design Basis." This criterion is similar to 10 CFR Part 50, Appendix A, GDC 50, "Containment Design Basis." FCS Design Criterion 49 states, "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 emergency core cooling systems." This request does not impact the compliance with FCS Design Criterion 49, as no changes to containment design basis were proposed as part of this LAR.

7. FCS Design Criterion 52, "Containment Heat Removal Systems." This criterion is similar to 10 CFR Part 50, Appendix A, GDC 38, "Containment Heat Removal." FCS Design Criterion 52 states, "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." The response was revised to indicate that the criterion is met for a LOCA using the CACFS.

   The basis of the revised response is the use of one of the two redundant, 100% capacity trains. Although the two trains are not of different principles, there is suitable redundancy in components and features, and suitable interconnections and power supplies are provided to assure the safety system function can be accomplished assuming a single active failure. Analysis show that containment pressure and temperature can be maintained below design values assuming the single active failure of one train of cooling.

The containment pressure and temperature requirements following an MSLB are met crediting two 100% capacity containment spray trains (or systems) and two 100% capacity containment air recirculation and cooling trains (or systems) for cooling the containment building atmosphere as described in USAR Section 14.16. The containment air recirculation and cooling system utilizes a different operating principle than the containment spray system.

For an MSLB, for the maximum MER analysis, the limiting conditions occur with maximum RCS flow. A LOOP will result in the reactor coolant pumps tripping and coasting down. Additionally, the main feedwater pumps will trip, which results in less main feedwater (MFW) flow injection to the SGs. With forced circulation in the RCS lost and loss of MFW flow, it is expected a LOOP will lessen the severity of the MER to the containment building; therefore, less cooling would be required if LOOP was assumed. The limiting failure for this accident is a failure of one containment fan cooler or failure of a MFW isolation valve to close.

This submittal changes the mitigation of the containment pressure response of a LOCA from CS to CACFS to enhance core cooling by increasing the amount of water delivered to the core during the injection phase of a LOCA. This change also significantly reduces the debris that could be transported to ECCS sump strainers during a LOCA and the resulting increase in strainer head loss. The supporting analysis demonstrates that the containment pressure response of a LOCA can be successfully mitigated by the CACFS. The change in the initiation logic for the CS was added and the source of cooling water to the containment coolers was clarified in the response to the criterion. The FCS Design Criterion 52 is met as a result of the modification to the CS system actuation logic. The revised response to FCS Design Criterion 52 requires NRC approval.

In summary, based upon the results of reanalysis of the containment response and radiological consequences, OPPD proposes to modify the CS system actuation logic and the associated FCS licensing and design bases, including the responses to FCS Design Criteria 15, 41 and 52 in the FCS USAR, Appendix G.

4.1.2 Design Basis (USAR)

The following USAR sections are applicable to this License Amendment Request:

- USAR 5.9, Containment Penetrations (Reference 6.20)

- USAR 6.2, Engineered Safeguards, Safety Injection System (Reference 6.1)

- USAR 6.3, Containment Spray System (Reference 6.2)
- USAR 6.4, Containment Air Cooling and Filtering System (Reference 6.5)
- USAR 7.3.2.4, Instrumentation and Control, Engineered Safeguards Controls and Instrumentation, Containment Spray Actuation System, (Reference 6.3)
- USAR 14.15, Safety Analysis-Loss-of-Coolant Accident (Reference 6.4)
- USAR 14.16, Safety Analysis Containment Pressure Analysis (Reference 6.4)
- USAR Appendix G, Responses to 70 Criteria (Section 4.1.1 of this LAR)

4.1.3 Approved Methodologies

See Attachments 4 and 6 of this LAR.

4.1.4 Analysis

The proposed change will enhance the performance of the containment sump strainers during a DBA. The CS system response to the MSLB is not affected by the proposed change. The containment pressure and temperature response to a LOCA were reanalyzed with the containment air coolers in operation.

One train of the containment air cooling and filtering system (CACFS), consisting of one containment air cooling (CAC) unit and one containment air cooling and filtering (CACF) unit, is capable of maintaining the containment pressure below containment design pressure and provide cooling to reduce the post-accident containment pressure and temperature.

The NPSH margin for ECCS equipment taking suction from the containment sump is improved without CS. Without CS, HPSI is the only system taking suction from the containment sump. Regardless, a LOCA containment analysis was performed to evaluate long-term sump temperature, specifically for use providing input to NPSH calculations for equipment taking suction from the containment sump. The analysis confirms the 8.99 feet of subcooling/overpressure presently credited is available with the new design configuration in this modification to the CS system actuation logic.

The limiting hot leg and cold leg break cases from the CCW/RW analysis in Reference 6.10 were run without CS. The results of the analysis documented

in Reference 6.10 indicate peak CCW temperature at the containment air cooler secondary inlet produced temperatures of 153°F and 156°F for the limiting hot leg and cold leg pump suction breaks, respectively. These temperatures are only slightly above the same cases with containment spray and remain less than the maximum allowed temperature of 160°F. The RW system inlet temperature was unchanged from Reference 6.10 and remained at 90°F. The MSLB transient results, in Reference 6.10, remain bounding. Therefore, operation without CS is acceptable with respect to RW/CCW operation.

The FCS large break (LB) LOCA analysis includes a minimum containment pressure analysis for ECCS performance following a LOCA; it currently assumes that all three CS pumps are operating. This is conservative since the more containment cooling capacity that is available following a LOCA, the lower the performance of the ECCS will be due to the lowered containment pressure, following the guidance in SRP 6.2.1.5 and BTP CSB 6-1. In general, FCS is not licensed to the SRP; however, the FCS LBLOCA peak cladding temperature (PCT) calculation is performed in accordance with the SRP Section 6.2.1.5 and BTP CSB 6-1. The reduction in the number of CS pumps operating will result in a lower calculated PCT due to the increase in ECCS performance caused by the resultant higher containment pressure. Therefore, the existing analysis continues to be bounding and the requirements of 10 CFR 50.46 will continue to be met.

The FCS radiological consequences of a LOCA are analyzed in accordance with RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and containment HEPA filters post-LOCA fission product removal is credited.

10 CFR 50.67, "Accident Source Term," requires that the evaluated radiological consequences for design basis accidents demonstrate with reasonable assurance that any individual located at the Exclusion Area Boundary or Low Population Zone would not receive a radiation dose in excess of 0.25 Sv total effective dose equivalent, and that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv total effective dose equivalent for the duration of the accident. The analysis results are within the regulatory limits of 10 CFR 50.67 and RG 1.183; therefore, the requirements of 10 CFR 50.67 and RG 1.183 will continue to be met.

Therefore, based on the considerations discussed above, (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.

4.2 Precedent

To minimize the amount of debris washed into the containment sump and address the requirements of NRC Generic Letter 2004-02, (Reference 6.8), OPPD had previously requested and received NRC approval of an amendment to manually shutdown unneeded CS pumps during a LOCA (References 6.6 and 6.7). Amendment 235 (Reference 6.7) is valid through the end of the current fuel cycle, April 2008. This proposed amendment provides proactive measures for reducing the risk associated with the potentially degraded or non-conforming ECCS and CS recirculation functions due to adverse post accident debris blockage. This proposed license amendment change will supersede those changes approved by Amendment 235 in that this change will modify the CS system actuation logic to preclude automatic start of the CS pumps for a LOCA and credit the containment air coolers for containment pressure control and the containment HEPA filters for dose reduction.

No specific industry precedence has been identified where any other licensee has requested to utilize post-LOCA water management strategies to optimize long term core cooling availability by precluding sump clogging via the license amendment process. However, this issue was discussed between the NRC and the industry owners group on May 11, 2006, as documented in References 6.12, 6.13, and 6.14.

4.3 Significant Hazards Consideration

The Omaha Public Power District (OPPD) proposes to modify the containment spray (CS) system actuation logic to change the means of containment pressure control during a loss-of-coolant accident (LOCA) from CS to the containment air coolers. This change reduces the amount of debris washed to the containment sump strainers during a LOCA, thereby reducing the vulnerability of the low and high pressure safety injection (HPSI) and containment spray pumps to a reduction of net positive suction head (NPSH). The change also increases the amount of water delivered to the core during the injection phase of a LOCA, thereby enabling the plant to cope with "larger" small break LOCAs without relying on recirculation for long term cooling.

The proposed modification will preclude automatic start of the CS pumps during a LOCA. The logic change will be accomplished by adding 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 proposed change also eliminates the use of CS and credits utilization of high efficiency particulate air (HEPA) filters for post-LOCA fission product removal.

This license amendment request (LAR) represents a portion of the actions required to satisfy the applicable regulatory requirements with respect to NRC Generic Letter (GL) 2004-02 as it relates to the CS system. OPPD's response to NRC Request 2e in GL 2004-02 indicated OPPD's intentions to update the design and licensing basis to reflect the results of the modifications and analyses.

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

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The containment spray (CS) system and the containment air cooling and filtering system (CACFS) are not initiators of any accident previously evaluated at the Fort Calhoun Station (FCS). Both systems are accident mitigation systems. Their licensing basis functions are to limit the containment pressure rise and reduce the leakage of airborne radioactivity from the containment by providing a means for cooling the containment following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) inside containment. The proposed modification to the CS system logic shifts the function of containment pressure and temperature control during a LOCA from the containment spray system to the equally capable and reliable containment air coolers. The change in the CS actuation logic does not impact the containment response to the MSLB analysis of record (AOR).

   The CACFS provides the design heat removal capabilities for the containment during the postulated LOCA. The system is operated to remove atmospheric heat loads from the containment during normal plant operation. Since system components are only lightly loaded during normal operation, system availability and reliability are enhanced. In the unlikely event that normal power sources are lost and one emergency diesel generator fails to operate, one containment air cooling and filtering unit and one containment air cooling unit will operate.

   The component cooling water (CCW) system, on which the CACFS is dependent, has sufficient capacity for all normal and shutdown operating modes. In addition, the system is capable of satisfying the design criteria under post-design basis accident (DBA) conditions with the single failure of an active component and a loss of instrument air. Analyses demonstrate that CCW flowrates to essential equipment would be adequate for removing post accident design basis heat loads.

Following implementation of the proposed change, at least one train of containment air coolers will be available to mitigate a LOCA. Analyses show that one train of coolers can maintain the containment pressure and temperature below the design values; therefore, the proposed change will have no adverse effect on the containment pressure analysis following a LOCA.

Analyses have also shown that one train of containment high efficiency particulate air (HEPA) filters maintains the radiological consequences doses within regulatory limits; therefore, the proposed change will have no adverse effect on the radiological consequences following a LOCA.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The CACFS was designed to remove heat released to containment atmosphere during the design basis accident to the extent necessary to maintain the structure below the design pressure. The proposed modification to the CS system logic shifts the function of containment pressure and temperature control from the containment spray system to the equally capable and reliable containment air coolers. The use of CACFS, as a means of containment pressure control, has been evaluated for the LOCA event and found to result in an acceptable peak containment pressure (peak pressure less than 60 psig). Radiological consequences were evaluated for the use of CACFS in this application using the guidance provided in Regulatory Guide (RG) 1.183. This radiological analysis demonstrates that the dose consequences are in compliance with applicable regulatory requirements. The estimated dose consequences at the exclusion area boundary (EAB), low population zone (LPZ), and control room (CR) remain within the acceptance criteria of 10 CFR 50.67 as supplemented by RG 1.183 and the standard review plan (SRP) 15.0.1. The assessment also demonstrates that the dose consequences in the technical support center (TSC) remain compliant with regulatory guidance provided in Supplement 1 of NUREG-0737.

No credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis have been created and none of the initial condition assumptions of any accident evaluated in the safety analysis are impacted.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

Response: No.

The containment building and associated penetrations are designed to withstand an internal pressure of 60 psig at 305°F, including all thermal loads resulting from the temperature associated with this pressure, with a leakage rate of 0.1 percent by weight or less of the contained volume per 24 hours. The containment air coolers are credited for maintaining containment pressure and temperatures within design limitations, and assure that the release of fission products to the environment following a design basis accident will not exceed regulatory guidelines for a large break (LB) LOCA.

The containment spray system and containment air coolers continue to be credited for limiting peak containment pressure for an MSLB.

Adequate NPSH margin is maintained for the HPSI pumps during the recirculation phase of a LBLOCA due to the reduction in ECCS sump strainer pressure drop.

The CACFS operates independently of the CS system to remove heat from the containment atmosphere. The CACFS consists of two redundant trains, each train with one air cooling and filtering unit and one air cooling unit, for a total of four cooling units. Operation of the CACFS, in accordance with analyses completed for the 2006 steam generator replacement, is and will continue to be credited in the MSLB containment pressure analysis. The operation and maintenance of the CACFS are not impacted by this proposed change. Therefore, the containment heat removal licensing basis is not adversely affected by the proposed change. The ability to maintain containment peak pressure and temperature, as well as long-term containment pressure and temperature, is maintained.

The LBLOCA 10 CFR 50.46 analysis assumes that there will be three CS pumps operating when evaluating the effects of containment pressure on ECCS performance. This assumption minimizes containment pressure, to conservatively evaluate ECCS performance in response to a LOCA. Eliminating operation of the CS pumps improves ECCS performance and thus increases margin to 10 CFR 50.46 limits on peak clad temperature, therefore, the existing analysis remains bounding as is.

In summary, following implementation of the proposed change:

- Peak containment pressure for analyzed DBAs remains within design limits;
- Radiological releases remain within the limits of 10 CFR 50.67; and
- The currently calculated peak clad temperature following a LOCA remains bounding.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, OPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (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.

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 REFERENCES

6.1 FCS Updated Safety Analysis Report (USAR) Section 6.2, Engineered Safeguards, Safety Injection System
6.2 FCS USAR Section 6.3, Engineered Safeguards, Containment Spray System
6.3 FCS USAR Section 7.3.2.4, Instrumentation and Control, Engineered Safeguards Controls and Instrumentation, Containment Spray Actuation Signal (CSAS)
6.4 FCS USAR Chapter 14, Safety Analysis
6.5 FCS USAR Section 6.4, Engineered Safeguards, Containment Spray System

6.6 Letter from OPPD (R. L. Phelps) to NRC (Document Control Desk), Fort Calhoun Station Unit No. 1, License Amendment Request, "Incorporation of Allowance to Secure Containment Spray Pumps During a Loss-of-Coolant-Accident to Minimize the Potential for Containment Sump Clogging," dated May 21, 2004 (LIC-04-0050)

6.7 Amendment 235 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1, (TAC MC3217) dated May 23, 2005, (NRC-05-0064)

6.8 NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 (NRC-04-0115)

6.9 Letter from OPPD (R. P. Clemens) to NRC (Document Control Desk), Fort Calhoun Station Unit No. 1, 60-Day Response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," dated August 8, 2003 (LIC-03-0105)

6.10 AREVA NP Document 77-9051353P-001 – "Summary of FCS Containment Analysis without Containment Spray," dated July 2007 (Proprietary version)

6.11 AREVA Document 43-10252PA-00 "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC" - Proprietary

6.12 NRC Meeting Summary for the May 11, 2006, "regarding Water Management Post-LOCA," dated May 30, 2006 (ML061460369)

6.13 Post-LOCA Water Management Strategies to Optimize Long Term Core Cooling Availability, NRC Meeting – NRC Slides, May 11, 2006 (ML061460338)

6.14 Post-LOCA Water Management Strategies to Optimize Long Term Core Cooling Availability, NRC Meeting – Westinghouse Slides, May 11, 2006 (ML0614603378)

6.15 Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), Application for Amendment of Operating License, dated February 7, 2001 (LIC-01-0010)

6.16 Letter from (A. B. Wang) to OPPD (S. K. Gambhir), Fort Calhoun Station, Unit No. 1 - Issuance of Amendment 201 (TAC No. MB1221) dated December 5, 2001 (NRC-01-112)

6.17 Implementation of GSI-191 Containment Sump Modification, Shaw Stone & Webster Nuclear, June 11, 2007

6.18 Generic Letter 84-04, Safety Evaluation of Westinghouse Topical Dealing With Elimination of Postulated Pipe Breaks Reports in Pressure Water Reactor Primary Main Loops, dated 2/1/1984, (NRC-84-0031)

6.19 Branch Technical Position (BTP) CSB 6-1, "Minimum containment Pressure Model for PWR ECCS Performance Evaluation"

6.20 FCS USAR Section 5.9, Containment Penetrations