September 11, 2006

Mr. R. T. Ridenoure
Vice President - Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: LOSS OF MAIN FEEDWATER EVENT ANALYSES AS DOCUMENTED IN THE UPDATED SAFETY ANALYSIS REPORT, SECTION 14.10 (TAC NO. MC7524)

Dear Mr. Ridenoure:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 242 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment revises the loss of main feedwater (LMFW) event analyses as documented in the Updated Safety Analysis Report (USAR), Section 14.10. This is in response to your application dated July 1, 2005, as supplemented by letters dated September 16, 2005, November 15, 2005, December 14, 2005, February 16, 2006, and July 6, 2006.

In support of the license amendment request, the licensee provided the results of the analyses of the LMFW and feedwater break events to the NRC staff for review and approval. The change will amend the Fort Calhoun Station, Unit 1, design and licensing basis by revising the USAR to describe an existing Emergency Operating Procedure operator action to isolate steam generator blowdown within 15 minutes of reactor trip during a loss of main feedwater event. Changes to the USAR are controlled in accordance with the requirements of Title 10 of the *Code of Federal Regulations*, Section 50.59, “Changes, tests, and experiments.”

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

Sincerely,

/RA/

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 242 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 242
License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee), dated July 1, 2005, as supplemented by letters dated September 16, 2005, November 15, 2005, December 14, 2005, February 16, 2006, and July 6, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, by Amendment No. 242 , changes to the Updated Safety Analysis Report to describe an existing Emergency Operating Procedure operator action to isolate steam generator blowdown within 15 minutes of reactor trip during a loss of main feedwater event as set forth in the application for amendment by the licensee, dated July 1, 2005, as supplemented by letters dated September 16, 2005, November 15, 2005, December 14, 2005, February 16, 2006, and July 6, 2006, are authorized.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: September 11, 2006

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 242 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated July 1, 2005 (Agencywide Documents Access and Management System Accession No. ML051950401), as supplemented by letters dated September 16, 2005, November 15, 2005, December 14, 2005, February 16, 2006, and July 6, 2006 (ADAMS Accession Nos. ML052630443, ML053220536 (Non-Public), ML053540318 (Non-Public), ML060470591, and ML061880425, respectively), Omaha Public Power District (OPPD, the licensee) requested changes to the Updated Safety Analysis Report (USAR) for the Fort Calhoun Station, Unit No. 1 (FCS).

The proposed amendment would revise the analysis of the loss-of-main feedwater (LMFW) event as documented in the USAR , Section 14.10. Specifically, the proposed changes would revise the FCS design and licensing basis by revising the USAR to credit existing emergency operating procedure (EOP) operator action to isolate steam generator (SG) blowdown within 15 minutes of a reactor trip during an LMFW event. The required operator action assures that the auxiliary feedwater (AFW) system performs its design function of maintaining SG water for decay heat removal when the AFW actuation signal (AFAS) is actuated.

The supplemental letters dated September 16, 2005, November 15, 2005, December 14, 2005, February 16, 2006, and July 6, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the Federal Register on August 2, 2005 (70 FR 44403).

2.0 REGULATORY EVALUATION

The licensee re-analyzed the LMFW and feedwater line break (FLB) events with the inclusion of the SG blowdown flow effect to demonstrate that it complied with the acceptance criteria of the analysis of record (AOR). The acceptance criteria, as noted in Reference 1, are: (1) the pressure in the reactor coolant system (RCS) and main steam system must be less than 110 percent of the design values; (2) fuel cladding integrity must be maintained by assuring that the departure from nucleate boiling (DNB) and fuel centerline melting design limits are met; and

(3) the event must not generate a more serious plant condition without other faults occurring independently.

The NRC staff also reviewed the operator manual actions using guidance contained in NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.8, "Time Response Design Criteria for Safety-Related Operator Actions," and NUREG-0800, "Standard Review Plan, Chapter 18.0, Human Factors Engineering," (Revision 1, 2004).

The NRC staff's review of the acceptability of the licensee's analyses is based on the acceptance criteria discussed above.

3.0 TECHNICAL EVALUATION

3.1 Background

In the current AOR, the licensee identified that the LMFW event was the most limiting design-basis event used to demonstrate the adequacy of the AFW system for removal of the decay heat during non-loss-of-coolant accident (non-LOCA) events. During an engineering assessment for support of plant operation with the replacement SGs, replacement of reactor vessel head and replacement pressurizer, the licensee evaluated the adequacy of the LMFW AOR in determining the acceptability of the AFAS setpoint and the minimum required AFW flow rate. As a result, the licensee identified that the current LMFW AOR did not include the SG blowdown effect. The current LMFW AOR was performed with the CESEC computer code that was not sufficiently detailed to model the SG blowdown valves. The licensee indicated that if the effect of the SG blowdown were considered in the analysis, the SG water level would reduce to a value lower than that predicted in the AOR, thus, it would result in more challenging conditions with a less effective SG heat transfer area for the AFW to remove decay heat. To adequately include the SG blowdown flow effect, the licensee performed analyses with the NRC-approved S-RELAP5 (References 4 through 7) for two limiting events. The analyses was performed for the USAR LMFW event and the FLB event which is not within the scope of the design-basis events in the USAR. The licensee indicated (Reference 1) that FCS has SG blowdown isolation valves designed to automatically close and isolate the SG blowdown flow following a containment pressure high signal (CPHS), or a pressurizer pressure low signal (PPLS). However, the results of the licensee's analyses showed that the SG blowdown valves would only be isolated by a CPHS during the large FLB event, but not during the small FLB or LMFW event. Therefore, the licensee concluded that it needed an operator action to close the SG isolation valves to mitigate the consequences of LMFW events or small FLB events.

In the analyses of the LMFW and small FLB events supporting the license amendment request, the licensee credited an operator action to isolate SG blowdown flow within 15 minutes following a reactor trip. The licensee provided the results of its analyses in Reference 1 and in responses to the request for additional information (RAI) in References 3, 8, 9, and 10 for the NRC staff review and approval.

The NRC staff has reviewed the licensee's analyses and the RAI responses to confirm that the analyses used the NRC-approved methodology; that the assumptions used in the analyses were adequate; and that the results of the analyses met the applicable acceptance criteria.

3.2 Analytical Methods

The licensee applied the S-RELAP5 code documented in Framatome-ANP (FRA-ANP) topical report (TR) (References 4 through 7) to perform analyses for the LMFW and FLB events. S-RELAP5 code simulates a multi-loop system using a model containing a reactor vessel, hot- and cold-leg piping, SGs, and pressurizer. The code is an updated version of the ANF-RELAP code, which is a modified RELAP5/MOD2 version used by FRA-ANP for pressurized-water reactor plant licensing analyses that include the small-break LOCA analysis, steam line break analysis, and non-LOCA transient analysis. The NRC has generically approved the FRA-ANP TR (References 5 and 6), and specifically approved it for the use of the TR in non-LOCA transient analyses for FCS licensing applications (Reference 7). The NRC staff found that both LMFW and FLB events were listed in Table 1, "Applicable SRP [Standard Review Plan] Chapter 15 Events," of the NRC staff's safety evaluation. Therefore, the NRC staff has concluded that the methods used by the licensee are acceptable.

During the course of the review, the NRC staff requested the licensee to justify that its FLB heat transfer model is adequate as it is compared with the Semiscale test data for modeling FLB events discussed in Section 4.3.3.1 of NUREG/CR-4945. In response to the NRC staff's request, the licensee calculated the heat transfer coefficients (Reference 8) using the S-RELAP5 code for three cases simulating the Semiscale test FLB cases with break sizes of 60, 80, 100 percent break. The licensee's calculations demonstrated that the calculated heat transfer coefficients were consistent with the Semiscale data. Both the licensee's calculated results and the Semiscale data showed that the SG heat transfer capacity remained unchanged until the SG liquid inventory was nearly depleted. This was followed by a rapid reduction to zero percent heat transfer with little further reduction in the SG water inventory. Therefore, the NRC staff concluded the heat transfer model used in the FLB analysis is acceptable.

3.3 Analyses of the LMFW and FLB Events

The LMFW event is a design-basis event considered in the current USAR, while the FLB event is not within the scope of the current USAR. Nevertheless, the licensee analyzed the LMFW and FLB events since both events were limiting events in terms of challenge to adequacy of the AFW capacity for decay heat removal during non-LOCA transients. The results of the licensee's analyses are discussed below.

3.3.1 Loss-of-Normal Feedwater Flow

A total loss of feedwater flow, the limiting LMFW event, may be caused by (1) inadvertent closure of main feedwater control or regulating valves, or feedwater isolation valves due to a feedwater controller malfunction or manual position by the operator, or (2) loss of all feedwater or condensate pumps.

The licensee analyzed the LMFW event using the values of initial key plant parameters that are listed in Table 1 of Reference 2. The licensee indicated (Reference 8) that the key parameters

that are sensitive to decay heat removal capability during the LMFW event included the initial core power, reactor protection system and AFAS setpoints, AFW flow and delay time, and SG blowdown flow rate and isolation time. In the analysis, the licensee applied uncertainties to these key parameters to maximize the heat generation rate on the RCS primary side and minimize the capacity of the secondary side system to remove primary side energy.

Some of the specific assumptions used by OPPD are:

1. The initial value of the core power was assumed at the rated power plus measurement uncertainty to maximize the RCS primary side heat generation rate and energy that was removed by the secondary side system.

2. A minimum initial value of the technical specification (TS) RCS flow rate minus measurement uncertainty was used to maximize the calculated RCS peak pressure and minimize the calculated minimum DNB ratio during the transient.

3. Both the assumed setpoints of the SG low water level trip and the AFAS were biased low by the measurement uncertainties to minimize the SG inventory at reactor trip and at AFW actuation. These assumptions maximize the challenge to the RCS secondary side system for decay heat removal.

4. The AFW flow rate was minimized to challenge the decay heat removal capacity of the AFW system. FCS has two safety grade AFW pumps: one electric motor driven pump and one turbine driven pump. Each operates at the same minimum flow rate of 180 gallons per minute (gpm). For a single failure consideration, one AFW pump was assumed to be inoperable, and the minimum flow was assumed from one operable pump to minimize the AFW flow rate.

5. The assumed AFW actuation delay time was biased high with the measurement uncertainty to minimize the SG inventory at AFW actuation and maximize the challenge of the RCS secondary system capability for removing RCS primary side energy.

6. The SG blowdown flow reduces the SG inventory and, therefore, challenges the RCS secondary side heat removal capability. In the analysis, the SG blowdown flow was modeled at a value corresponding to the design flow rate per SG and was assumed to occur in each SG at initiation of the LMFW event until the SG blowdown system was isolated. Continued SG blowdown flow until isolation resulted in a lower SG inventory that delayed the time for the AFW flow to replenish the SG inventory and to provide adequate decay heat removal capability.

7. The initial core inlet temperature was assumed at the maximum TS value to maximize the initial stored energy in the RCS primary side.

Other assumptions for the initial conditions were: (1) the pressurizer pressure was assumed at the nominal value; (2) the pressurizer water level was assumed at the nominal value plus operating band and measurement uncertainty; (3) the SG pressure was assumed at a value consistent with zero percent SG tube plugging; and (4) the main steam safety valve (MSSV) setpoints were biased high by measurement uncertainties. The licensee confirmed that the

values of the initial RCS flow, pressurizer pressure, pressurizer water level, SG pressure and MSSV setpoints had no significant effect on SG inventory or timing of SG blowdown isolation.

Based on the discussion above, the NRC staff concluded that the values of the key parameters used in the analysis were adequate and acceptable. The assumed values maximized the challenge to the RCS secondary side system to remove the RCS primary side energy and resulted in a higher peak RCS pressure and a shorter operator action time allowed for SG blowdown flow isolation.

In the analysis, the licensee assumed that the operator would isolate the SG blowdown valves which would terminate the blowdown flow within 15 minutes following a reactor trip. In its response to the NRC staff's RAI (Reference 8), the licensee indicated that the SG blowdown isolation system is a safety grade system with redundant blowdown lines and isolation valves in series for each SG. The closure time of the four isolation valves was tested by the licensee per its Surveillance Test Program procedures, OP-ST-BD-3000, "Blowdown System Category A and B Valve Exercise Test," and OP-ST-ESF-0011, "Channel A and B Automatic and Manual Engineered Safeguard Actuation Signal Test." This testing is in accordance with TS 3.3, "Reactor Coolant System and Other Components Subject to ASME XI Boiler and Pressure Vessel Code Inspection and Testing Surveillance." The NRC staff determined that the requirements of the design and testing of the valves would provide reasonable assurance that the SG blowdown valves were reliable to isolate the blowdown flow on demand. Therefore, the NRC staff concluded that the use of the valves for flow isolation in the LMFW analysis was acceptable.

The licensee presented the LMFW event response in Figures 1 through 8 and the associated sequence of events in Table 2 of Reference 2. Following an LMFW event, the SG water inventory decreases as a consequence of the continuous steam supply to the turbine. The SG water level decreases to the low SG water level setpoint that trips the reactor at 25 seconds. A turbine trip follows the reactor trip. The RCS primary-to-secondary side heat transfer causes the SG pressure to increase. The SG pressure increases following the turbine trip causing the MSSVs to lift at 28 seconds and mitigate the increase in SG pressure. The SG continues to boil off its water inventory through the MSSVs. The boil off continues until the SG water level reaches the AFAS setpoint at 525 seconds. The licensee assumed a delay time of 51 seconds for the AFW flow to reach the SGs. Therefore, at 576 seconds, the AFW system delivers water to each SG at a flow at a rate of 180 gpm. With the assumed delay time of 15 minutes after the reactor trip, the operator would isolate the SG blowdown flow at about 925 seconds. The AFW flow is initially not sufficient to remove the RCS primary side energy until 3,054 seconds when the decay heat level had decreased sufficiently. After 3,054 seconds, the AFW is sufficient to remove the decay heat and the RCS temperature and pressure began to decrease.

As indicated in the AOR, the challenge to peak RCS primary and secondary overpressure limits was less severe for the LMFW event (with the reactor trip and the secondary-system side isolation nearly coincident) than for the loss-of-load (LOL) to both SGs event (with secondary system side isolation at event initiation and continued operation at power for considerable period of time). Since the licensee confirmed that the AOR for the LOL event remained valid, the NRC staff agreed with the licensee that the peak primary and secondary pressures during the LMFW event would be bounded by that of the LOL event and, thus, would be below the 110 percent of the design pressure acceptance criteria. The licensee also calculated the

maximum post-trip hot-leg temperature to be 567 °F and the subcooling margin to be significantly greater than 20 °F. These results assure that boiling and DNB will not occur as result of the LMFW event. Therefore, the DNB and fuel centerline melting limits will be met. In addition, the analysis in Reference 2 determined that the maximum pressurizer level was 77.6 percent of span, assuring that the pressurizer does not fill and therefore, liquid will not be discharged through the power-operated relief valve (PORVs) or pressurizer safety valve (PSVs).

3.3.1.1 Operator Actions

Isolation of SG blowdown flow ensures that the AFW system will be able to perform its design function to maintain adequate SG water level for decay heat removal once the AFAS is actuated. The purpose of the AFAS setpoint is to start at least one of the two available safety-related AFW pumps during a design-basis event. The AFW pumps' operation satisfies the acceptance criteria to ensure adequate removal of decay heat and to maintain the primary and secondary systems within overpressure limits.

The analyses for the FLB and LMFW events were revised using the AREVA computer codes. As a result of the revised analyses, the licensee concluded that isolation of the SG blowdown valves were needed to meet the acceptance criteria. In the FLB event, the SG blowdown valves isolate as a result of containment pressure high signal or a pressurizer pressure low signal. However, in the event of an LMFW, the operator must take manual actions to isolate the SG blowdown valves. This operator action is performed by closing the SG blowdown valves within a specific time to isolate the SG blowdown flow and provide adequate SG water level for decay heat removal.

The licensee has determined that the operator action to isolate the SG blowdown flow using the SG blowdown valves is acceptable because the action is performed from within the control room and occurs soon after a reactor trip associated with LMFW. The licensee used the ANSI/ANS-58.8-1984 standard, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," to calculate the minimum operator action time to perform the SG blowdown isolation. The NRC staff reviewed the calculation and confirmed that the licensee's assumptions for the LMFW event would produce the resulting operator action time of 8 minutes. This operator action time is less than the 15 minutes assumed by the licensee's revised LMFW event analysis. However, the licensee has indicated that the operators have been able to complete this action within 8 minutes following reactor trip during numerous simulator training sessions that involved LMFW events. Therefore, the licensee concluded that the operator should take a minimum of 8 minutes, but no longer than 15 minutes, to isolate SG blowdown for LMFW events.

In its RAI, the NRC staff inquired about the licensee's specific actions required to be performed by the operator to isolate the SG blowdown valves as well as any additional operator actions. The licensee responded that Step 13.1 of EOP-00, "Standard Post Trip Actions," directs the operator to isolate SG blowdown in the control room by closing the two sets of blowdown isolation valves, HCV-1387A/B for SG RC-2A and HCV01388A/B for SG RC-2B. The manipulations of both valves, one for each steam generator, are required for SG blowdown isolation. The operators are required in an LMFW event to verify that the turbine as well as the reactor have been tripped, using the annunciators and panels. If the operators observe that the

main feedwater is unavailable, the expected contingency would be to isolate SG blowdown, which could occur before EOP-00 is completed. If the SG blowdown isolation is not completed immediately, the operators will arrive at step 13 of EOP-00 within 8 minutes to perform the operator action. The NRC staff also reviewed the licensee's additional analysis of the LMFW event that provides a timeline of the event without any operator actions and no additional feedwater. The licensee identifies that by 18.3 minutes, the SGs will reach their minimum water inventory to provide adequate heat sink for decay heat removal and will be completely dry by 30 minutes after the event. With the operator performing the SG blowdown isolation no later than 15 minutes, the NRC staff agrees that the operator action would prevent the SGs from reaching the minimum water levels needed for decay heat removal.

The NRC staff also asked how the simulator was modeled to simulate the LMFW event and whether the 8 minutes for completion of the operator actions and subsequent system response were representative of an actual LMFW event, which would include the operating crew response. The licensee referred to their analysis model of the LMFW event, which used the AREVA codes, that established the bounding time within which the operator action is required to isolate SG blowdown in order to protect the reactor core limits. The simulator model used for the LMFW event was a best estimate model of the plant response that remains within the bounds of the analysis provided that the operator action to isolate SG blowdown is performed in less than 15 minutes after a reactor trip. Operator training for all training crews reinforced the performance of EOP-00 actions, which include SG blowdown isolation when the main feedwater is unavailable. During these training sessions, the licensee observed that the operators consistently performed this operator action within 8 minutes between the different crews. The NRC staff agrees that the licensee's operator training and revised simulator model provides reasonable assurance that the operators can perform the SG blowdown isolation within 15 minutes.

The NRC staff has reviewed the licensee's proposal to credit an operator action time of 15 minutes for SG blowdown isolation after a reactor trip in the LMFW analysis. With the operator action delay time, the NRC staff concluded that the analysis of the LMFW event was acceptable based on: (1) the licensee performed the analysis using the NRC-approved methods; (2) the analysis used adequate values for key plant parameters; (3) the operator action would prevent the SGs from reaching the minimum water levels needed for decay heat removal; (4) the licensee's operator training and revised simulator model provides reasonable assurance that the operators can perform the SG blowdown isolation within 15 minutes; and (5) the results of the analysis demonstrated that the acceptance criteria used in the AOR were satisfied in meeting the RCS pressure boundary limits, fuel integrity safety limits, and the maximum pressurizer water level limit.

3.3.2 Feedwater Line Break (FLB)

An FLB is an event with a break in a feedwater line large enough to prevent the addition of sufficient feedwater to maintain shell-side water inventory in the SGs. The FLB may reduce the ability to remove heat generated by the reactor core from the RCS, because fluid in the SG is discharged through the break at a rate that may be large enough to exceed the addition of MFW after the reactor trip.

The licensee performed a break spectrum analysis for the FLB event including break sizes of 60 percent, 70 percent, 80 percent, 90 percent, and 100 percent of the largest feasible break area. The largest feasible break is 0.9 ft^2 and that occurs at the inlet to the feedring. In the analysis, the licensee applied uncertainties to key parameters to maximize the heat generation rate on the RCS primary side and minimize the capacity of the RCS secondary side system to remove primary side energy. The licensee listed the values of initial plant operating conditions and key plant parameters in Table 3 of Reference 2 for the FLB analysis.

The NRC staff compared the values used in the FLB analysis with that used in the LMFW analysis discussed in Section 3.3.1 above, and found that the values were identical for all key plant parameters, except for the following assumptions:

1. The SG low water trip setpoint was 0.0 percent of narrow range (NR) span versus 25.5 percent NR used for the LMFW analysis, and

2. The MSSV lifting setpoints were the nominal sepoints minus 3 percent uncertainty versus the nominal setpoints plus 3 percent uncertainty used in the LMFW analysis.

The NRC staff concluded that: (1) the assumption in item 1 of using a lower SG water level setpoint would delay the reactor trip time, thus, maximizing the heat generated in the core; and (2) the assumption in item 2 of using lower MSSV opening setpoints would actuate the MSSVs in the unaffected SG to release steam earlier, thus, minimizing the SG inventory for decay heat removal. Therefore, the NRC staff concluded that the values were adequate and acceptable.

Also, in the large FLB analysis, a containment isolation actuation signal (CIAS) was credited (Reference 9) to automatically isolate the SG blowdown system, while an operator action time of 15 minutes after reactor trip was assumed in the small FLB analysis for SG blowdown isolation.

In the FCS design, a CIAS would isolate SG blowdown. The CIAS could occur when either a CPHS or a PPLS was initiated. A CPHS would occur when containment pressure reached 5.4 pounds per square inch gauge (psig) (including uncertainty) and a PPLS would occur when pressurizer pressure decreased to 1578 pounds per square inch absolute (psia) (including uncertainty).

In addition, the licensee made the following assumptions (Reference 3) for the AFW model used in the FLB analysis:

1. For conditions with both SG pressures greater than 435 psia, the AFW system was actuated to deliver flow to any SG that had water level of less than 15 percent of the wide range (WR) span,

2. For conditions with one SG pressure less than 435 psia, the AFW system was actuated to deliver flow to only the SG that had a pressure of equal to or greater than 435 psia and a water level of less than 15 percent WR, and

3. For conditions with both SG pressures less than 435 psia, the AFW system was actuated to deliver flow to the SG that had a differential pressure of greater than

135 psia than the other, and a water level of less than 15 percent WR. If the neither SG had a differential pressure of greater than the other by at least 135 psia, and a water level less than 15 percent WR, the AFW system was not actuated to deliver flow to any SG.

The NRC staff found that the AFW actuation setpoints of the SG pressure, SG differential pressure, and SG water level used in the analysis were bounded by the values specified in the FCS Technical Specifications. The bounding septoints would delay delivery of the AFW flow to the SGs, thus minimizing the SG inventory for decay heat removal. Therefore, the NRC staff concluded the assumptions were adequate and acceptable.

In the original submittal (Reference 2), the licensee provided the results of analysis of break sizes of 60 percent, 70 percent, 80 percent, 90 percent and 100 percent of the largest feasible break area of 0.9 ft^2. The analysis assumed that the SG blowdown flow was isolated on a SG low pressure signal, rather than on a CIAS based on either CPHS or a PPLS. During the review, the NRC staff questioned the adequacy of the assumption. In response (Reference 9), the licensee provided the results of an analysis to confirm the conservative nature of the assumption on using the SG low pressure signal versus CPHS or PPLS to isolate SG blowdown. In the analysis, the time of CPHS to occur was calculated with an S-RELAP5 containment model for FLB sizes of 100 percent and 60 percent. The S-RELAP5 containment model is the same as that developed for the main steam line break analysis of Reference 4, which was previously approved by the NRC in Reference 5. The results demonstrated that the CPHS would occur before actuation of the SG low pressure signal. Therefore, the NRC staff agreed with the licensee that use of the SG low pressure signal, rather than the CPHS, to isolate SG blowdown flow would result in less SG water inventory for decay heat removal, resulting in a higher post-trip hot-leg temperature, and is adequate and acceptable. The licensee's break size analysis identified that the limiting break size regarding post-trip hot-leg temperature was the 80 percent (0.72 ft^2) break size.

The licensee presented the response of the limiting FLB break case in Figures 9 through 18 and the associated sequence of events in Table 5 of Reference 2. With an instantaneous loss of all MFW at an FLB initiation, the SG water level rapidly decreased to reach a low SG water level trip setpoint that resulted in a reactor trip at about 6 seconds. The reactor trip was followed by a turbine trip. During the FLB event, the pressure in the affected SG decreased rapidly and reached the SG low pressure isolation actuation setpoint that isolated the main steam isolation valves (MSIVs) at about 48 seconds. SG blowdown was isolated on a low SG pressure signal at 49 seconds. Once the MSIVs closed, the affected SG quickly depressurized to the atmospheric pressure, while the unaffected SG pressure continued to increase and reached the MSSV lifting setpoints that opened the first MSSV in the unaffected SG at 308 seconds. As the fluid in the unaffected SG continued to boil off through the MSSVs, the unaffected SG water level reached the AFAS setpoint at 634 seconds. Assuming a delay time of 51 seconds for the AFW to reach the SGs, the AFW started to deliver flow to the unaffected SG at 685 seconds. The AFW flow was initially not sufficient to remove the primary side energy and the unaffected SG went dry at about 1,516 seconds. As a result, the RCS coolant temperature began to increase. The peak post-trip hot-leg temperature reached 557 °F at 2,832 seconds when the decay heat level had decreased sufficiently. Subsequently, the AFW was sufficient to remove the decay heat, and the RCS temperature and pressure began to decrease.

The licensee showed that the RCS primary and secondary pressures were within the 110 percent of the design pressure limits for the FLB analysis. The licensee calculated the maximum post-trip hot-leg temperature to be less than 600 °F and the subcooling margin to be significantly greater than 20 °F, thus, assuring that boiling and DNB would not occur. Therefore, the DNB and fuel centerline melting limits would not be exceeded. Furthermore, the licensee showed that the maximum pressurizer level was less than the initial value, assuring that the pressurizer would not fill and liquid would not discharge through the PORVs or PSVs.

The licensee also performed (Reference 9) an FLB analysis for a 10 percent break size with the S-RELAP5 containment model and showed that a CPHS would occur for break sizes down to a size of 10 percent to actuate automatic isolation of SG blowdown flow, and the maximum post-trip hot-leg temperature would be less than 600 °F criterion. For FLB sizes less than 10 percent, the licensee assumed that the CPHS would not occur. For such cases, the operator would isolate SG blowdown flow within 15 minutes of a reactor trip, similar to the LMFW event, as directed by EOP-00 step 13. In support of the adequacy of the 15-minute operator action time after a reactor trip to isolate SG blowdown flow for small FLB events, the licensee performed an analysis for a representative very small break (5 percent). The results showed that the transient of an FLB with a very small break size was similar to the LMFW and the credit of operator action to isolate the SG blowdown within 15 minutes following a reactor trip would limit the maximum post-trip hot-leg temperature to less than 600 °F criterion.

Based on its review discussed above, the NRC found that the FLB analysis was performed using the approved method with adequate values for key input parameters, and the results of the analysis showed that the applicable acceptance criteria were met. Therefore, the NRC staff concluded that the FLB analyses were acceptable.

3.3.3 Analysis of Record

The NRC staff has evaluated the licensee's analyses of the LMFW and FLB events with consideration of the SG blowdown flow effect for FCS. The licensee has demonstrated compliance with the three criteria in the AOR. For criterion (1) the licensee has demonstrated that the maximum RCS and SG pressures during an FLB event would be less than 110 percent of the design values, and the LMFW event is bounded by the loss of load event for which the maximum RCS and SG pressures shown in the AOR would be less than 110 percent of the design values. In meeting criterion (2), the licensee showed that the SGs would provide a sufficient heat sink for decay heat and reactor coolant pump heat, as evidenced by the maintenance of the peak post-trip hot-leg temperature of less than 600 °F, and the RCS subcooling of greater than or equal to 20 °F. The licensee demonstrated compliance with criterion (3) by showing that the pressurizer would not overfill and liquid would not discharge through the PORVs or PSVs.

Based on the evaluation discussed above, the NRC staff found that (1) the licensee performed the analyses with the NRC-approved method, (2) the assumptions used in the analyses maximized the challenge to the decay heat removal capability of the AFW system and thus, were adequate, (3) the operator action would prevent the SGs from reaching the minimum water levels needed for decay heat removal; (4) the licensee's operator training and revised simulator model provides reasonable assurance that the operators can perform the SG blowdown isolation within 15 minutes; and (5) the results of the analysis demonstrated that the

acceptance criteria used in the AOR were satisfied in meeting the RCS pressure boundary limits, fuel integrity safety limits, and the maximum pressurizer water level. Therefore, the NRC staff concluded that the LMFW and FLB analyses are acceptable. The NRC staff found that the proposed changes in the USAR, Section 14.10.1, adequately reflects the assumption of operator action time used in the LMFW analysis for the SG blowdown isolation and, therefore, concluded that the changes are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (70 FR 44403; published August 2, 2005). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from S. K. Gambhir (OPPD) to NRC, "Fort Calhoun Station Union No. 1 License Amendment Request, 'Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event,' " dated July 1, 2005.

2. Attachment 4 to Reference 1, "AREVA Document 86-5056804-00, Fort Calhoun Station Document FC-06967, FCS RSG-Auxiliary Feedwater Actuation Signal (AFAS) Setpoint Verification."

3. Letter from R. T. Ridenoure (OPPD) to NRC, "Response to Request for Additional Information and Revision of Fort Calhoun Station Union No. 1 License Amendment Request, 'Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event,' " dated September 16, 2005.

4. EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," dated November 1999.

5. Letter from S. A. Richards (NRC) to J. F. Mallay (FRA-ANP), "Acceptance for Referencing of Licensing Topical Report EMF-2310(P), Revision 0, 'SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors' (TAC No. MA7192)," dated May 11, 2001.

6. Letter from H. N. Berkow (NRC) to J. F. Mallay (FRA-ANF), "Final Safety Evaluation for Topical Report EMF-2310(P), Revision 1, 'SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors' (TAC No. MC0321)," dated May 19, 2004.

7. Letter from A. B. Wang (NRC) to R. T. Ridenoure (OPPD), "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment Re: Addition of Topical Report References to TS 5.9.5, 'Core Operating Limits Report' (TAC No. MB3449)," dated March 4, 2002.

8. Letter from R. T. Ridenoure (OPPD) to NRC, "Response to Request for Additional Information and Revision of Fort Calhoun Station Union No. 1 License Amendment Request, 'Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event,' " dated December 14, 2005.

9. Letter from J. A. Reinhart (OPPD) to NRC, "Response to Fort Calhoun, Unit 1 - Request for Additional Information to License Amendment Request for Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event (TAC No. MC7524)," dated July 6, 2006.

10. Letter from R. T. Ridenoure (OPPD) to NRC, "Response to Request for Additional Information and Revision of Fort Calhoun Station Union No. 1 License Amendment Request, 'Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event,' " dated February 16, 2006.

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