Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

September 16, 2004
LIC-04-0095

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555

References:
1. Docket No. 50-285
2. Letter from OPPD (Ralph L. Phelps) to NRC (Document Control Desk) dated May 21, 2004, "Incorporation of Allowance to Secure Containment Spray Pumps During a Loss-of-Coolant-Accident to Minimize the Potential for Containment Sump Clogging" (LIC-04-0050)
3. Letter from NRC (Alan B. Wang) to OPPD (R. T. Ridenoure) dated August 23, 2004, Request for Additional Information (TAC No. MC3217) (NRC-04-0106)

SUBJECT: Response to Request For Additional Information Regarding 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"

In support of the 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" (Reference 2), the Omaha Public Power District (OPPD) provides the attached response to the Nuclear Regulatory Commission's (NRC's) Request for Additional Information of Reference 3.

I declare under penalty of perjury that the forgoing is true and correct. (Executed on September 16, 2004). No commitments to the NRC are made in this letter.

If you have any questions or require additional information, please contact Thomas R. Byrne of the Fort Calhoun Station Unit No. 1 Licensing staff at (402) 533-7368.

Sincerely,

Ross T. Ridenoure
Vice President

RTR/TRB/trb

Attachment 1: Response to Request For Additional Information Regarding 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"

c: Division Administrator - Public Health Assurance, State of Nebraska

ATTACHMENT 1

Response to Request For Additional Information Regarding 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"

Question 1:

Describe the steps necessary to re-start a CS [containment spray] pump that has been stopped after starting on an automatic actuation signal. Assess the likelihood that it will not re-start when required.

OPPD Response:

The selected CS pumps are secured by taking the control switch to the "pull-to-lock" position. The start signal to the CS pumps is not reset; therefore, the only operator action required to restart the selected CS pump is to reposition the control switch out of "pull-to-lock" and verify proper pump operation (motor amps and indicating light) and CS system response (proper CS flow). No other system manipulations are required to restore CS flow.

The likelihood that a CS pump will fail to restart when required is considered to be low based on the following:

- A review of the Maintenance Rule data reveals that the CS pumps at Fort Calhoun Station Unit No. 1 (FCS) have never experienced a failure to start;

- In order to take the compensatory measure of reducing to one CS pump operation, it is required that CS pumps started and ran properly at the onset of the event. Therefore, operability of the pumps has been demonstrated;

- Restart of a secured CS pump will not exceed duty cycle limits for the pump motors or major electrical components;

- Restart of the secured CS pump will not result in emergency bus loading concerns;

- The likelihood of air binding of the pumps is low based on the fact that the pumps are secured prior to experiencing Net Positive Suction Head (NPSH) or air ingestion concerns and sufficient head of water exists above the pump suctions to prevent air introduction into the pumps during the period that they are secured; and,

- The thermal transient due to CS flow being stopped and restarted is not expected to exceed the design thermal transient limits for the CS pumps.

Question 2:

Verify that the large break LOCA peak cladding calculation is done in accordance with Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," of the Standard Review Plan, in particular, Branch Technical Position CSB 6-1, B.2, which states that all engineered safety feature containment heat removal systems operate at maximum heat removal capacity. Does the analysis assume operation of all three CS pumps and both trains of containment fan coolers? Where is this documented?

OPPD Response:

The large break LOCA Peak Cladding Temperature (PCT) calculation is performed in accordance with the Standard Review Plan (SRP) Section 6.2.1.5 and the referenced Branch Technical Position (BTP).

The large break LOCA Analysis is performed using two single failure scenarios: (1) failure of one emergency diesel generator (EDG), and (2) one LPSI pump fails to start/run. The first scenario results in the minimum containment heat removal systems (one CS pump/header and one train of Containment Fan Coolers (CFCs)). The LPSI fail to start/run scenario results in the operation of all engineered safety feature (ESF) containment heat removal systems (i.e., 3 CS pumps/2 headers and all CFCs). The LPSI failure scenario results in the highest calculated PCT of 1956°F and is therefore the limiting case.

The EDG failure scenario, assuming operation of one CS pump/header and one train of CFC, results in a calculated PCT of 1948°F. Although the action to reduce to one CS pump as described in the LAR would result in slightly higher amount of containment cooling than analyzed for in the large break LOCA analysis EDG failure case because both CFC trains are operating, it is still bounded by the limiting case of full ESF containment cooling that produces a higher PCT. Therefore, the action to secure CS pumps as described in the License Amendment Request does not adversely affect the large break LOCA PCT calculation.

The large break LOCA analysis is documented in Framatome ANP, Inc, EMF-2734, Revision 0, "Fort Calhoun Cycle 21 Large Break LOCA/ECCS Analysis with Reduced Reactor Coolant System Flow Rate," dated April 12, 2002 (FCS Updated Safety Analysis Report (USAR) Reference 14.15-34).

Question 3:

Verify that the design basis calculation of radiological dose following a large break LOCA credits one CS pump. Where is this documented?

OPPD Response:

The design basis radiological dose calculation credits the following CS flow rates for fission product removal:

- Injection Phase: 1,885 gpm
- Recirculation Phase: 2,800 gpm

These flow rates correspond to one CS pump/one CS Header operation following a LOCA and take into account pump degradation, instrument uncertainties, and flow through the minimum recirculation lines.

The calculation is summarized in FCS USAR Section 14.15.8, and is contained in FCS USAR Reference 14.15-27, "Implementation of Alternate Source Terms Site Boundary & Control Room Dose Analysis for Fort Calhoun Station", January 2001, Stone & Webster using the credited value of 3,100 gpm for the Recirculation Phase. This was transmitted in the letter from OPPD (W. G. Gates) to the NRC (Document Control Desk) dated February 7, 2001, Application for Amendment of Operating License (LIC-01-0010) (ML010400079). Subsequently in a February 28, 2004 revision to this calculation, the credited value for the Recirculation Phase flow rate was conservatively reduced to 2,800 gpm.

Question 4:

In addition to the successful operation of the safety injection system, describe other indications of possible inadequate core cooling that the operator would use during a large break LOCA.

OPPD Response:

The following indications that may indicate possible inadequate core cooling are checked by the Shift Technical Advisor (STA) every 10 minutes during the Safety Function Status Checks (SFSCs) contained in Emergency Operating Procedures EOP-3, Loss of Coolant Accident, and EOP-20, Functional Recovery Procedure:

- Reactor Vessel Level Measurement System (RVLMS) indicates that the core is no longer covered
- Core Exit Thermocouples (CETs) indicate greater than superheat

- Hot Leg Resistance Temperature Detectors (RTDs) indicate greater than superheat
- Steam Generator Wide Range Levels <20%, indicating the possibility of reduced heat removal via the steam generators
- Cold Leg RTDs indication rising temperature, indicating inadequate heat removal from the Reactor Coolant System