Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

September 16, 2005
LIC-05-0109

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

References: 1. Docket No. 50-285
2. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk) dated July 1, 2005, "Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event," (LIC-05-0001)

SUBJECT: Response to Request for Additional Information and Revision of Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event"

The Omaha Public Power District (OPPD) previously submitted in Reference 2 an update to the Updated Safety Analysis Report in accordance with 10 CFR 50.90 to clarify existing operator actions during a Loss of Main Feedwater event. During the NRC staff review, two requests for additional information were transmitted via email to OPPD on July 27 and 28, 2005, and were discussed with the NRC Project Manager.

Attachment 1 to this letter provides the Omaha Public Power District (OPPD) response to the NRC requests for additional information. Attachment 2 to this letter revises the application of ANSI N18.2, to correct the calculation for allowable operator response time previously presented in Reference 2 Attachment 1, Section 4.

No commitments to the NRC are made in this letter. I declare under penalty of perjury that the foregoing is true and correct. (Executed September 16, 2005)

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If you require additional information, please contact Thomas C. Matthews at (402) 533-6938.

Sincerely,

R. T. Ridenoure
Vice President

RTR/rlj

Attachments:

1. OPPD's Response to Request for Additional Information on Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event"
2. Revision of Supporting Information to Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event"

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

OPPD Response to Request for Additional Information on Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event"

NRC Question 1:

Provide the bases of using the loss of main feedwater event and feedwater line break accident as the design basis events to determine the auxiliary feedwater actuation signal.

OPPD Response:

The Fort Calhoun Station (FCS) safety-grade automatic Auxiliary Feedwater Actuation System (AFAS) was designed and installed in response to NUREG-0737, Item II.E.1.2, with NRC acceptance of this design and setpoints in Reference 1 of this attachment. NRC acceptance of this system and design are further documented in References 2 and 3.

The Safety Evaluation attached to Reference 1 summarizes how the system functionally operates with the steam generator (SG) wide range (WR) low level setpoint and the steam generator pressure permissive setpoints. For information, the AFAS setpoints are summarized below:

Setpoint	Analytical Setpoint	Equipment Setpoint
----------Level Setpoint----------		
SG WR Level (%)	<15.0	<28.2
AND		
----------Pressure Permissive----------		
Pressure (psia)	<435.0	<466.7
Delta Pressure (psid)	>135.0	>119.7

Analytically the system functions as follows:

- Both SG Pressures >435 psia: Feed any SG that has its level <15% WR.

- <u>One SG Pressure <435 psia:</u> Feed only the SG that has a pressure ≥435 psia if its level <15% WR.
- <u>Both SG Pressures <435 psia</u>: If one SG has a differential pressure >135 psid than the other and a level <15% WR, feed it. If neither SG has a differential pressure >135 psid than the other and a level <15% WR, neither SG will be fed.

The Auxiliary Feedwater (AFW) system was designed to provide AFW flow to mitigate the condition where the SG inventory is being depleted and RCS heat removal capability has been lost, which occurs for the limiting conditions of the Loss of Main Feedwater (LMFW) event and the Feedwater Line Break (FWLB) accidents. The Main Steam Line Break (MSLB) accident was also analyzed to ensure that the ruptured SG would not be fed with AFW because the consequences of adding AFW during this accident result in adversely increasing the event cooldown. Thus, the system setpoints were established to ensure that the affected SG would not be fed during an MSLB.

Therefore, one of these two events, LMFW or FWLB, is the limiting event for the determination of the AFAS setpoints because either may present the greatest challenge to the loss of SG inventory.

<u>NRC Question 2:</u>

In determination of the 15-minute operator action time for steam generator blowdown system isolation, are other heat-up events (such as loss of load, loss of condenser vacuum, loss of ac power, and small break LOCA) analyzed? Provide information to justify that the loss of main feedwater is the limiting heat-up event for determining the operator action time of 15 minutes.

<u>OPPD Response:</u>

For the determination of the AFAS setpoints and the time that operator action to isolate SG blowdown is required, other heat-up events were considered but not analyzed because the other heat-up events are bounded by the LMFW event as discussed below:

- Loss of Load:
  This is an Anticipated Operational Occurrence (AOO) for FCS and main feedwater remains available following the event (i.e., loss of offsite power is not part of the FCS design basis for AOOs). Thus, there is no need for AFW flow to the SGs and this is a non-limiting event.
- Loss of Condenser Vacuum:
  This is not a design basis event for FCS. If the event occurred, however, main feedwater (MFW) and condensate would remain available (i.e., no loss of offsite power). This event is bounded by the Loss of Load Event, discussed above.
- Loss of AC Power:
  This event is similar to the LMFW event in that electric power to the MFW pumps is lost. Also significant is the loss of power to the reactor coolant pumps

(RCPs). The loss of the RCPs (specifically, forced circulation) has two effects which result in the LMFW event (with AC power available) being more limiting with respect to depletion of SG inventory. These effects are:

- Loss of the RCPs results in a reduction in RCS Heat Removal requirements, since for FCS there is no longer 5.6 MWth of RCP heat that needs to be removed from the RCS in addition to the decay heat.
- With the loss of the RCPs, decay heat removal will occur upon the establishment of natural circulation. Natural circulation is established with the retention of a finite amount of decay heat in the RCS, heat which under forced circulation conditions would have been transferred to the SGs earlier. The net result is a delay in full decay heat removal and delayed SG inventory depletion.

- Small Break Loss of Coolant Accident (SB LOCA):
  Since RCS Heat Removal for a SB LOCA is accomplished by both SG heat removal and heat removal out the break, the net heat removal through the SGs for the event (with or without a loss of offsite power) is bounded by the LMFW.

References:

1. Letter from R.A. Clark (NRC) to W.C. Jones (OPPD), Technical Specification Amendment No. 65, dated June 18, 1982, (NRC-82-0104)
2. Letter from R.A. Clark (NRC) to W.C. Jones (OPPD), Franklin Research Center Technical Evaluation Report, dated July 22, 1982, (NRC-82-0132)
3. Letter from R.A. Clark (NRC) to W.C. Jones (OPPD), "NUREG-0737, Item II.E.1.2, Auxiliary Feedwater System Initiation and Flow," dated September 1, 1982 (NRC-82-0166)

Revision of Supporting Information to Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event"

Purpose of This Attachment:

Following the submittal of the License Amendment Request (LAR) by Reference 1 below, two inaccuracies were found in the information supporting the LAR. The purpose of this attachment is to correct and revise the documentation of the calculation for allowable operator response time, based upon ANSI/ANS-58.8-1984, presented in Attachment 1, page 4 of Reference 1.

Basis of the Corrections:

Two separate corrections to the calculation of minimum operator response time are being made:

1. In accordance with the definitions of "Plant Conditions" as defined in ANSI N18.2-1973 (Reference 2 below), the Loss of Feedwater event is a level 2 Plant Condition, not a level 3 Plant Condition as used in the original LAR supporting information of Reference 1. This reclassification changes the minimum time margin to be applied using Table 1 of ANS/ANSI 58.8-1984 from 10 minutes to 5 minutes. Also, the reclassification changes the minimum operator action time delay equation to be applied using Table 2 of ANSI 58.8-1984 from 3 + n x 1 minutes to 1 + n x 1 minutes where n signifies the number of discrete manipulations.
2. In Reference 1, one manipulation was credited to isolate steam generator blowdown where in fact two discrete manipulations, one for each steam generator are required.

Therefore, OPPD has revised the LAR to correct these two inaccuracies and add a reference for the ANSI N18.2 as follows:

Resulting Text Corrections:

1. The marked up revision of LIC-05-0001, Attachment 1, Section 4 Technical Analysis is:

   Manual isolation of SG blowdown is acceptable because the action is performed from within the control room and occurs soon after a reactor trip associated with LMFW. This action satisfies the criteria of References 7.6 and 7.7 assuming a Plant Condition ~~3~~ 2, which allows a Time Margin of ~~10~~ 5 minutes and an Operator Action Time Delay of ~~3~~ 1 + n x 1 minute, where n signifies the number of discrete manipulations. In this case, n equals ~~one~~ 2 (one discrete action per

SG), and the Operator Action Time Delay is 4 3 minutes, for a total of 8 minutes for operator action, which is less than the 15 minutes assumed by the revised NSSSRP (Nuclear Steam Supply System Replacement Project) LMFW event. Based on simulator training observations, FCS operators complete this action within 8 minutes following reactor trip which indicates that crediting 15 minutes for this operator action in the LMFW event is conservative. Therefore, OPPD is specifically requesting the NRC to allow the requested clarification to USAR Section 14.10.1 to credit the use of operator action to isolate the SG blowdown valves within 15 minutes following a reactor trip during a LMFW.

Also, Reference 7.7 is added to page 7:

7.7 Nuclear Safety Criteria for the Design of Station Pressurized Water Reactor Plants, ANSI N18.2-1973

2. The clean revision of LIC-05-0001, Attachment 1, Section 4 Technical Analysis becomes:

   Manual isolation of SG blowdown is acceptable because the action is performed from within the control room and occurs soon after a reactor trip associated with a LMFW. This action satisfies the criteria of References 7.6 and 7.7 assuming a Plant Condition 2, which allows a Time Margin of 5 minutes and an Operator Action Time Delay of 1 + n x 1 minute, where n signifies the number of discrete manipulations. In this case, n equals 2 (one discrete action per SG) and the Operator Action Time Delay is 3 minutes, for a total of 8 minutes for operator action, which is less than the 15 minutes assumed by the revised NSSSRP (Nuclear Steam Supply System Replacement Project) LMFW event. Based on simulator training observations, FCS operators complete this action within 8 minutes following reactor trip which indicates that crediting 15 minutes for this operator action in the LMFW event is conservative. Therefore, OPPD is specifically requesting the NRC to allow the requested clarification to USAR Section 14.10.1 to credit the use of operator action to isolate the SG blowdown valves within 15 minutes following a reactor trip during a LMFW.

   With the addition of Reference 7.7 to page 7:

   7.7 Nuclear Safety Criteria for the Design of Station Pressurized Water Reactor Plants, ANSI N18.2-1973

Conclusion:

The requirement for completion of operator actions to isolate SG blowdown within 15 minutes of the reactor trip for a Loss of Main Feedwater event is justified based on the following time limitations:

1. The transient analysis which now models SG blowdown demonstrates that heat sink capability is not lost unless operator action exceeds 15 minutes. 15 minutes is therefore the maximum allowable time for operator action.
2. The ANSI standards prescribe that operator action can be credited 8 minutes following the plant trip. This minimum allowable time for operator action is less than the maximum allowable time set by the transient analysis, therefore the crediting of operator action for the period between 8 and 15 minutes following the trip is acceptable.
3. Actual simulator observations demonstrate isolation of SG blowdown is completed within 8 minutes of the reactor trip. This actual performance demonstrates that the operators are effectively trained and will complete the action before the maximum allowable action time of 15 minutes.

Attachment 2 References:

1. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk) dated July 1, 2005, "Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event," (LIC-05-0001)
2. ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Station Pressurized Water Reactor Plants