Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

July 25, 2006
LIC-06-0082

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

References:
1. Docket No. 50-285
2. Letter from Omaha Public Power District (R. T. Ridenoure) to Nuclear Regulatory Commission (Document Control Desk), dated October 31, 2005, Fort Calhoun Station Unit No.1 License Amendment Request, "Updated Safety Analysis Report Revision for Radiological Consequences Analysis for Replacement NSSS Components," (LIC-05-0107)
3. Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure), dated July 19, 2006, "Request for Additional Information (RAI) Related to Revising the Fort Calhoun Updated Safety Analysis Report (TAC No. MC8857)" (NRC-06-0090)

SUBJECT: **Response to Request for Addition Information, Fort Calhoun Station Unit No.1 License Amendment Request, "Updated Safety Analysis Report Revision for Radiological Consequences Analysis for Replacement NSSS Components" (TAC No. MC8857)**

Reference 2 provided the Omaha Public Power District's (OPPD) license amendment request to revise the Updated Safety Analysis Report (USAR) radiological consequences for operation following the fall 2006 outage replacement of NSSS components. The request revised Safety Analysis, General, Section 14.1, as well as the radiological consequences analyses for the events of Seized Rotor (SR), Section 14.6.2.8; Main Steam Line Break (MSLB), Section 14.12.6; Control Element Assembly Ejection (CEAE), Section 14.13.4; and Steam Generator Tube Rupture (SGTR), Section 14.14.3.

The Attachment of this letter provides the OPPD Response to the Nuclear Regulatory Commission Request for Additional Information contained in Reference 3.

I declare under penalty of perjury that the forgoing is true and correct. (Executed July 25, 2006) No commitments are made in this letter.

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

Sincerely,

Sudesh Gambhir
Division Manager – Nuclear Projects
Fort Calhoun Station

SKG/rlj

Attachment:
Response to Request for Addition Information of the Updated Safety Analysis Report Revision for Radiological Consequences Analysis for Replacement NSSS Components

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

Omaha Public Power District
Fort Calhoun Station
Response to Request for Addition Information of the Updated Safety Analysis Report Revision for Radiological Consequences Analysis for Replacement NSSS Components

NRC Question 1:

The U. S. Nuclear Regulatory Commission (NRC) staff notes that the atmospheric dispersion factors (χ/Q values) provided in the October 31, 2005, license amendment request (LAR) are the same as those approved in the safety evaluation associated with FCS Amendment No. 201. Amendment 201 implements use of the alternative source term in the FCS design-basis accident dose assessments. Therefore, for each design basis accident or event addressed in the October 31, 2005, LAR, please discuss any differences between the two sets of release scenarios. If there are differences, please justify how the differences do not impact the applicability of the χ/Q values associated with Amendment No. 201 to the dose assessment associated with the October 31, 2005, LAR.

Question 2 Response:

The release paths of the LAR are the same as the release paths of Amendment No. 201 as summarized below.

Release Paths

Event	Amendment No. 201	October 31, 2005 LAR
SGTR	Before Trip- Condenser/Air Ejector	Before Trip- Condenser/Air Ejector
	After Trip – MSSVs/ADV	After Trip – MSSVs/ADV*
MSLB	Affected Steam Generator (SG) – Room 81 blowout panel via Steam Line (SL) break	Affected SG – Room 81 blowout panel via SL break
	Intact SG – Via Automatic Dump Valve (ADV)	Intact SG -Via MSSVs/ADV*
CEA Ejection	MSSVs/ADV, Containment Leak	MSSVs/ADV*, Containment Leak
SR	MSSVs/ADV	MSSVs/ADV*

*A small leak from the exhaust of the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) is conservatively assumed to be released from the Main Steam Safety Valves / Automatic Dump Valve (MSSVs/ADV). This is because the χ/Q values from the TDAFWP exhaust are lower than the χ/Q values from the MSSVs/ADV as shown below:

Time	MSSV/ADV (χ/Q) (s/m^3)	TDAFWP (χ/Q) (s/m^3)
0-2 hr	5.06E-03	4.73E-03
2-8 hr	4.46E-03	3.75E-03
8-24 hr	2.08E-03	1.88E-03
1-4 day	1.59E-03	1.36E-03
4-30 day	1.34E-03	1.17E-03

NRC Question 2:

Which water mass inventories (both primary and secondary) were used in calculating coolant activities? Please, indicate if the used values are the minimum, nominal or maximum.

Question 2 Response:

The water mass inventories used in calculating coolant activities in the October 31, 2005 LAR are minimum values (to be conservative) as follows:

For the Steam Generator Tube Rupture (SGTR):
- Minimum primary side inventory of 250,000 lbm
- Minimum secondary side inventory:
    - Unaffected SG 45,708 lbm
    - Affected SG hot side break 70,261 lbm
    - Affected SG cold side break 71,692 lbm

For the Main Steam Line Break (MSLB):
- Minimum primary side inventory of 250,000 lbm
- Minimum secondary side inventory: unaffected SG 45,708 lbm

For the Seized Rotor (SR) and Control Element Assembly (CEA) Ejection:
- Minimum primary side inventory of 250,000 lbm
- Minimum secondary side inventory for each SG 45,708 lbm

NRC Question 3:

The submitted S-RELAP5 analysis indicates 0 percent/0 percent fraction of failed/melted fuel. Is this a change from the analysis of record?

Question 3 Response:

The thermal-hydraulic analyses of record for the SGTR, MSLB, SR, and CEA Ejection events indicate 0% fuel failed and 0% melted fuel. Thus there is no change from those analyses associated with the LAR.

The radiological consequences analyses of record assumed 0% failed/0% melted fuel for the SGTR and MSLB. The radiological consequences analyses of record for the SR and CEA Ejection events conservatively assumed 1% failed/0% melted fuel for the SR event and 10% failed/1% melted fuel for the CEA Ejection event.

The radiological consequences analyses of the LAR assumed 0% failed/0% melted fuel for the SGTR and MSLB. The radiological consequences analyses of the LAR for the SR and CEA Ejection events conservatively assumed 0.5% failed/0% melted fuel for the SR event and 1% failed/1% melted fuel for the CEA Ejection event.

NRC Question 4:

Attachment 4 (AREVA document 86-5064251-00) describes the thermal-hydraulic input in support of the consequence analysis. Please confirm that the main steam safety valves and atmospheric dump valves actuation setpoints used are those specified in the technical specifications.

Question 4 Response:

The setpoints of the main steam safety valves (MSSVs) used in the LAR analyses are based on the Technical Specification nominal values plus 3% setpoint uncertainty. The ADV does not have an automatic initiation setpoint. This valve is activated by the operator.

NRC Question 5:

Presumably, the cumulative steam releases used in the Attachment 6 (Stone & Webster report) came from the Attachment 4 (AREVA document). However, it is not immediately obvious which tables in Attachment 4 were used to generate Tables 7.6-7.9 in Attachment 6. Please clarify.

Question 5 Response:

SR and CEA Ejection events:

The cumulative steam releases of the SR and CEA Ejection events shown in Tables 7.6-3 and 7.7-2 of Attachment 6 are identical. These tables were generated as follows:

- The MSSV release of Tables 7.6-3 and 7.7-2 of Attachment 6 was generated by adding Tables 3.6 and 3.7 of Attachment 4.
- The ADV release shown in Tables 7.6-3 and 7.7-2 of Attachment 6 was generated from Table 3.8 of Attachment 4.

MSLB event:

The cumulative steam release of the MSLB event shown in Table 7.8-2 of Attachment 6 was generated by adding the corresponding values in Tables 4.6 and 4.7 of Attachment 4.

SGTR event:

For the SGTR event, two thermal-hydraulic cases were analyzed to provide input to the radiological consequences analyses, a cold side break and a hot side break. The releases of both cold and hot side breaks were evaluated individually. From the thermal-hydraulic view point, the releases were similar and only the cold releases were included in the AREVA summary report (Attachment 4 of the LAR). The radiological consequences analyses evaluated both the cold and hot side breaks and concluded that the hot side break is more limiting. The cumulative steam releases of the SGTR event of Table 7.9-5 of Attachment 6 are those of the hot side break. Table 7.9-5 of Attachment 6 was generated from the hot side break steam releases of Tables B.6 and B.9. Tables B.6 and B.9 were generated in the detailed SGTR long term evaluation AREVA document 32-5051831-01 "FCS RSG – Steam Generator Tube Rupture with Cooldown to Shutdown Cooling Entry Conditions." This detailed evaluation was not included in the LAR since it contains AREVA proprietary information.