August 3, 1988

Docket No. 50-285

Mr. Kenneth J. Morris
Division Manager - Nuclear Operations
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102-2247

Dear Mr. Morris:

SUBJECT: REQUEST FOR SUPPLEMENTAL INFORMATION ON STEAM GENERATOR TUBE RUPTURE EVENT METHODOLOGY

References: (1) OPPD letter, LIC-87-598, R. Andrews to NRC, dated September 30, 1987
(2) OPPD Nuclear Analysis, Reload Core Analysis Methodology, Transient and Accident Methods and Verification, OPPD-NA-8303-P, Revision 1, dated November 1986

Omaha Public Power District (OPPD) requested, in reference (1), the review and approval of the proposed Steam Generator Tube Rupture Event Methodology for incorporation into reference (2). The NRC technical staff has reviewed this submittal and found that further information is necessary. This was addressed in a telephone conversation with Mr. K. Holthaus and other members of your staff on July 28, 1988. In this conversation it was stated that information addressing the steam generator overfill scenario and the worst-case offsite dose scenario would be required. The information needed is detailed in the enclosure to this letter. Additionally, it was determined that the inclusion of this event methodology into reference (2) was not necessary for the Cycle 12 reload analysis submittal.

Therefore, OPPD is required to submit, within 120 days of the receipt of this letter, a response which addresses the items in the enclosure.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under Public Law 96-511.

Sincerely,
/s/
Patrick D. Milano, Project Manager
Project Directorate IV
Division of Reactor Projects - III, IV, V and Special Projects

Enclosure:
As stated
cc w/enclosure:
See next page
DISTRIBUTION

Docket File	NRC PDR	Local PDR	D. Crutchfield
L. Rubenstein	J. Calvo	A. Thadani	W. Hodges
J. Callan, RIV	P. Milano	A. Gilbert	T. Westerman, RIV
P. Noonan	OGC-Rockville	E. Jordan	B. Grimes
ACRS (10)	Plant File	PD4 Reading	

LA:PD4	PM:PD4	SRXB	D:PD4
PNoonan	PMilano	RJones	JCalvo
08/1 /88	08/1 /88	08/2 /88	08/3 /88

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
August 3, 1988

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Project Directorate IV
Division of Reactor Projects - III, IV, V and Special Projects

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Mr. Kenneth J. Morris
Omaha Public Power District

Fort Calhoun Station
Unit No. 1

cc:
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Washington, D.C. 20036

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Washington County Board
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Blair, Nebraska 68008

Mr. Phillip Harrell, Resident Inspector
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Fort Calhoun, Nebraska 68023

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Fort Calhoun, Nebraska 68023

ISSUES REQUIRING RESOLUTION

STEAM GENERATOR TUBE RUPTURE EVENT METHODOLOGY

FORT CALHOUN STATION (CE)

The accident analysis methodology for the Steam Generator Tube Rupture (SGTR) Event proposed for incorporation into the Updated Safety Analysis Report (USAR), Chapter 14, Section 14, has been determined to be inadequate. Two scenarios exist that could result in worst-case consequences: the steam generator overfill scenario and the worst-case offsite dose scenario. These scenarios should be addressed in a revised submittal. The following criteria should specifically be considered when evaluating both aspects for a design basis SGTR event.

A. Assumptions

1. Assume a most limiting single active failure for the steam generator overfill scenario (i.e. possibly the auxiliary feedwater flow control valve failure full open or the steam generator atmospheric dump valve (ADV) failure closed, associated with the faulted SG).

2. Assume a most limiting single active failure for the worst-case radiological offsite dose (possibly a stuck open ADV associated with the faulted SG).

   a. Determine whether the worst case for dose is overfill with maximized break flow, or an uncovered tube break with minimized break flow as in the North Anna event.

   b. Determine whether the loss of offsite power would be more conservative at the onset of the SGTR event or at the time of reactor trip.

3. Assume a maximum Technical Specification leakage in the unaffected steam generator (1 gpm).

4. Assume fuel failure for any rod with MDNBR below 1.19 as a result of the SGTR event.

B. Information Needed to Present the Results of the SGTR Analysis

1. Sequence of events on a time scale, from the onset of the tube rupture to the pressure equalization between the primary and secondary coolant systems.

2. Operator action times for identifying the faulted SG, isolation of the faulted SG, initiation of cooldown, depressurization, etc. This should also be presented on a time scale.

3. Discussion of the issue of a potential SG overfill, including the integrity of the steam lines under a water-filled condition and, if applicable, the effects of water flow through the safety valves, since these were not designed for this service.

4. Discussion of the basis for the operator action times utilized (i.e. simulator runs, etc).

5. Major transient curves including RCS pressure, secondary system pressure, DNBR, SG water level, leak rates for the faulted and intact SGs, etc.

6. Amount of fuel failure based on DNBR.

7. Calculated radiological consequences as compared to the limits set forth in 10 CFR Part 100 (2 hours and 8 hours), including pre-accident and coincident iodine spiking and noble gas inventory based on tech spec limits (usually $100/\bar{E}$).

8. Radiological parameters and curves including mass flow loss with respect to time, flashed fractions, and partition and decontamination factors in accordance with SRP 15.6.3.

9. Discussion of the consequences of a postulated break at the top of the tube bundle, as it had occurred at North Anna.

Contact: A. Gilbert