Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

April 6, 1984
LIC-84-094

Mr. James R. Miller, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

References: (1) Docket No. 50-285
(2) Response to Section 2.2-2.4 of Enclosure 3, NUREG-0612, transmitted to NRC by District letter, Jones to Eisenhut (LIC-82-033), dated January 21, 1982
(3) District Letter, Jones to Miller (LIC-84-039) dated February 14, 1984.

Dear Mr. Miller:

Fort Calhoun Station Unit No. 1
Control of Heavy Loads, Phase 2

In accordance with Reference 2 (page 33, item 1), the Omaha Public Power District made a commitment to develop written procedure to provide an alternate path for shutdown cooling water in the event of a load drop in the area bounded by Columns 10 and 11, and the biological shield wall, in containment. This procedure would then allow use of the polar crane in this area. Reference (3) changed the completion date for this commitment so that additional calculations verifying the validity of the proposal could be performed. These calculations were completed and, as reported to Mr. E. G. Tourigny of your staff in a telephone call on February 29, 1984, revealed that the proposed solution is not acceptable. The District has further investigated the possible alternatives and provides the following discussion and revised commitment.

The commitment was initially made to meet the requirements of Enclosure 3, Section 2.4 of NUREG-0612, which addresses crane operation in the area where equipment for decay heat removal is located. The area of concern contains the Safety Injection and Charging headers which are required for shutdown cooling and boron injection.

On further evaluation it has been determined that because of the following reasons, use of the proposed alternate flow path is not the best alternative for meeting the requirements of NUREG-0612 Enclosure 3, Section 2.4:

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1. The alternate flow path requires that the flow must be routed through a 1" line at a very high rate. This may result in excessive vibrations leading to failure of this line if used for an extended period.

2. The flow rate provided is not adequate to remove the decay heat from the core until 45 days from reactor shutdown, and after 1/3 of the core has been removed. Thus, the current load restriction would need to be maintained until 45 days after reactor shutdown.

3. In the event of a load drop the alternate path would have to be physically realigned to become operational. Many of the valves requiring alignment are locked closed, and a time factor could become important to get the system operational.

4. This procedure could only be used when the reactor head is removed from the core, and the water level is above 15 feet.

Because of the above, we believe that this procedure will be unduly restrictive and will not be practical. Therefore, we would like to revise this commitment and exercise other options provided in NUREG-0612.

Enclosure 3, Subsection 2.4.2.b of NUREG-0612, provides three options to meet the established criteria. These options are:

1. If separation and/or redundancy is provided between safety-related equipment, then the hazard is eliminated in the event of a load drop.

2. Where mechanical stops or electrical interlocks are provided to prevent carrying loads over safe shutdown equipment, the hazard can be eliminated.

3. Where technical specifications or administrative procedures are used to eliminate the load hazard, discussion should be provided to ensure validity of the constraints.

In accordance with the existing plant procedures, the District is in compliance with requirements of Enclosure 3, Section 2.4 by meeting the requirements outlined in subsection 2.4.2.b option 3 above.

An administrative procedure (OI-HE-1) currently exists which restricts load handling in the area defined above. This procedure has been in effect since May 8, 1981. This procedure administratively prohibits loads from being carried in the hazard area. This restriction can only be overridden for a limited period or for handling a specific load per a written procedure with Plant Review Committee approval. This ensures that probability of a load drop in this area is extremely small, which is consistent with the requirements of NUREG-0612 Enclosure 3, Section 2.4.

Attached please find a copy of the revised pages to the District's Section 2.2-2.4 response, which incorporates the results of the above discussion.

This revised corrective action requires use of administrative controls to meet Section 2.4.2.b requirements. Further, these change pages incorporate the current status of Proposed Corrective Actions, Items 2, 3, and 4, as it was detailed in Reference 3.

Sincerely,

W C Jones

W. C. Jones
Division Manager
Production Operations

WCJ/MDH/rh-N

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

Mr. E. G. Tourigny, Project Manager
Mr. L. A. Yandell, Senior Resident Inspector

ATTACHMENT

Revised Pages to the District's
January 21, 1982 Letter

4.0 PROPOSED CORRECTIVE ACTIONS

The following corrective actions are planned based upon the results of the design evaluation and analysis of Control of Heavy Loads at Nuclear Power Plants.

1. The load handling operations by the polar crane shall continued to be governed by the existing procedure (OI-HE-1). This procedure restricts load handling in the area bounded by Column Lines 10 and 11 and the biological shield wall in the contain- 1
ment (Ref. GHDR #11405-A-5, Appendix B). This restriction can only be removed for a limited period or for handling a specific load per a written PRC approved procedure. This procedure has been implemented since May 8, 1981, as an interim corrective action to meet the requirements of NUREG-0612.

2. A procedure will be written to prevent the loss of the raw water pumps due to a load drop accident destroying the power supply cables. The procedure will:

   a) Prohibit loads from being carried over the area above the cable tray supplying power to all four raw water pumps, and/or

   b) Outline emergency repair procedures to connect the fire pump discharge into the raw water header to provide component cooling during the repair of the raw water pump power cables.

   This item is complete; see the District's letter dated February 1
   14, 1984 (LIC-84-039).

3. The design of the access door to the reactor vessel cavity at EL. 976'-0" (Ref. GHDR Drawing No. 11450-A-14, Appendix B) will be reviewed. This design evaluation will ensure that the door can withstand hydrostatic pressure of the flooded cavity after a postulated load drop shears off all the nozzles of the reactor vessel and the vessel falls into the cavity. If the door design is found to be deficient, appropriate steps will be taken to ensure the reactor core remains covered with coolant.

4. The Geared Rotary Limit Switches will be wired for the upper limit on the main hook and the auxiliary hook of the containment polar crane. This will provide redundant limit switches and prevent a two-blocking accident.

   This item is complete; see the District's letter dated February 1
   14, 1984 (LIC-84-039).

5.0 SCHEDULE

Proposed Corrective Action, Item 3 will be completed by the end of
the 1985 Refueling Outage as was detailed in the District's letter 1
dated February 14, 1984 (LIC-84-039).

REV. 1

LOAD/DROP TABLE - A-1

CRANE CONTAINMENT POLAR CRANE LOCATION CONTAINMENT

HEAVY LOAD RV HEAD & LIFT RIG 120 T, LOWER LOAD BLOCK 5T (MOVED DURING REFUELING OUTAGE ONLY) IMPACT AREA SEE FIGURES IN APPENDIX B AND FIGURE B-1

ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION STATEMENT
(1) 994'-0"	*(1) SHUTDOWN COOLING PIPING	(1) CASE (I) EL. 994' IF A LOAD DROP SHEARS OFF PIPING BETWEEN THE CONTAINMENT PENETRATIONS AND VALVES HCV-327, HCV-329, HCV-331, HCV-333, AND HCV-348, THESE VALVES AND THE VALVES OUTSIDE OF CONTAINMENT CAN BE CLOSED TO STOP DRAINING OF THE SYSTEM OR THE REACTOR CAVITY. THE TECH. SPECS. ALLOW 8 HOURS FOR THE SYSTEM TO BE INOPERABLE IF NOT IN REFUELING. THEREFORE, THE PLANT COULD WELD A BLIND FLANGE ON THE NECESSARY PIPING ENDS AND PLACE A PORTION OF THE COOLDOWN SYSTEM BACK IN OPERATION IN THE 8 HOURS TIME LIMIT...NO HAZARD FOR CASE (I). CASE (II) EL. 1013' IF LOAD DROP SHEARS VALVE HCV-348, SYSTEM LOW POINT AT E. 1003', THE REACTOR CAVITY COULD DRAIN, WITH NO OPERATOR ACTION, TO THE BOTTOM OF THE HOT LEG PIPE AT AN APPROX. EL. OF 1005'. THIS WOULD LEAVE APPROX. 4' OF WATER COVERING THE CORE. IT SHOULD BE NOTED, HOWEVER, THAT THE LOWER CAVITY AREA WILL NOT DRAIN BELOW 1013' DUE TO THE CONCRETE WALL SEPARATING THE VESSEL FROM THE LOWER CAVITY. THE TECH SPECS ALLOW THE SYSTEM TO BE INOPERABLE FOR A MAXIMUM OF 8 HOURS AND THE PLANT STAFF COULD REPAIR THE BREAK WITHIN THAT TIME PERIOD...NO HAZARD FOR CASE (II).
(2) 994'-0"	*(2) SAFETY-INJECTION AND CHARGING PIPING	(2) The probability of a load drop in this area is extremely small since a restriction has been placed on the Polar Crane prohibiting carrying loads in this area. The restricted area falls between column lines 10 and 11 on the drawing attached, the containment wall, and steam generator RC-2B wall. This procedure administratively prohibits loads from being carried in the hazard area. This restriction can only be overridden for a limited period or for handling a specific load per a written Plant Review Committee approved procedure. This ensures that probability of a load drop in this area is extremely small, which is consistent with the requirements of NUREG-0612 * EQUIPMENT REFERENCED TO THIS NOTE IS THE ONLY SAFETY RELATED EQUIPMENT REQUIRED TO MEET NUREG-0612 CRITERIA WHICH IS LOCATED WITHIN THE LOAD DROP ENVELOPE.

LOAD/DROP TABLE - A-1

CRANE CONTAINMENT POLAR CRANE (CONT.) LOCATION CONTAINMENT

HEAVY LOAD RV HEAD & LIFT RIG 120T, LOWER LOAD BLOCK 5T IMPACT AREA SEE FIGURES IN APPENDIX B AND FIGURE B-1

ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION STATEMENT
(2) 994'-0" CONTINUED	*(2) SAFETY-INJECTION CHARGING PIPING	Enclosure 3, Section 2.4. This administrative procedure meets the requirements of guideline 2.4.2.b(3). The administrative procedure has been in effect since May 8, 1981. No hazard, therefore, exists.
(3) 1013'-0"	*(3) REACTOR VESSEL	(3) IN THE UNLIKELY EVENT OF THE REACTOR VESSEL HEAD AND LIFT RIG FALLING ON THE REACTOR VESSEL, THE WORST CASE CONSEQUENCES WOULD BE THE SHEARING OF ALL THE NOZZLES WHICH SUPPORT THE REACTOR VESSEL. THE VESSEL WOULD THEN DROP INTO THE REACTOR VESSEL CAVITY. THE CAVITY WOULD FILL WITH WATER FROM THE BROKEN LOOP NOZZLES. ADDITIONAL WATER WOULD BE SUPPLIED FROM THE ACCUMULATORS AND SIRW TANK IF REQUIRED. FLOW WOULD BE ESTABLISHED THROUGH THE SHUTDOWN COOLING SYSTEM AND INTO THE

* EQUIPMENT REFERENCED TO THIS NOTE IS THE ONLY SAFETY REALTED EQUIPMENT REQUIRED