Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

July 14, 1994

Mr. William T. Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn.: Document Control Desk

Subject: Quad Cities Station Unit 1
'B' Core Spray Header Crack Indication
NRC Docket No. 50-254

References: (a) Teleconference between the NRC (R. Assa, et al.) and Commonwealth Edison (J. Schrage, et al.) on July 7, 1994.

(b) J. L. Schrage to W.T. Russell letter dated June 17, 1994.

Mr. Russell:

In the Reference (a) teleconference, the Staff requested additional information from Commonwealth Edison (ComEd) regarding the 'B' Core Spray header crack for Quad Cities Station Unit 1. More specifically, the Staff requested information on the impact/consequences of a postulated 360° through-wall crack in the 'B' Core Spray header at the junction (tee) box location. In addition, the Staff requested the approach utilized to determine the amount of leakage which would be expected from the crack.

As indicated in Reference (b), the crack in the 'B' Core Spray header is located where the piping and junction box meet in the heat affected zone of the weld. As detailed on Figure 1, two standard bolted brackets are located +/- 60° from the junction box, and two additional clamps are welded +/- 10° from the junction box. The clamps closest to the junction box (+/- 10°) provide additional vertical support of the Core Spray piping, and would be expected to restrict displacement of the header in the event of a 360° through-wall crack.

ComEd utilized a finite element model (GE-ANSYS) to determine the leakage flow from a 360° through-wall crack in the header. This model was originally constructed for the 'B' Core Spray header configuration to support the flaw evaluation, which was presented in Reference (b). As detailed in the Attachment to this letter, the model was modified to include a 360° pipe break at the crack location. A flow load force was determined for the break location utilizing design flow conditions. This load was applied to the break location, and the model was rerun to establish the leakage area (area after the crack has fully displaced). From this leakage area, which was determined to be 2.2 square inches, the leakage flow rate was calculated to be 656 gpm.

An evaluation was performed to determine the LOCA Analysis impact of a 360° through-wall crack of the 'B' Core Spray header (utilizing the SAFER/GESTR model for Quad Cities Station Unit 1). For this evaluation (which is also detailed in the Attachment), the Recirculation Line Break LOCA was selected as the bounding accident because it results in

A001

the highest peak cladding temperature (PCT). In addition to the 656 gpm coolant leakage from the 'B' Core Spray header, the evaluation took into account conservative coolant leakage estimates from other current Unit 1 reactor internals conditions (postulated through-wall core shroud crack, loop 'A' recirculation system jet pump riser crack, and postulated through-wall cracking of the loop 'B' recirculation system internals as a result of the runout event during the current outage).

The following single failure cases were evaluated because they result in the largest negative contribution to the PCT analysis:

- the 125V battery failure, and
- the Low Pressure Coolant Injection (LPCI) system injection valve failure.

For the battery failure, the remaining contributing Emergency Core Cooling Systems (ECCS) are one Core Spray Pump and two LPCI pumps into a single recirculation loop. For the LPCI injection valve failure, the two Core Spray pumps are the remaining contributing ECCS systems.

The licensing basis peak cladding temperature, assuming no coolant loss due to reactor vessel internals cracking, for the 125V battery failure and LPCI injection failure cases, is approximately 1600° F. Given a 656 gpm leak from the 'B' Core Spray header (from a 360° through-wall crack), coupled with conservative coolant leakage values for the core shroud and recirculation system internals, ComEd determined that the bounding PCT increase would be less than 110° F. This PCT increase is tolerable in view of the 2200° degree F limit in 10 CFR 50.46.

If there are any further questions or comments, please contact John L. Schrage at 708-663-7283.

Very truly yours,

John L. Schrage
Nuclear Licensing Administrator

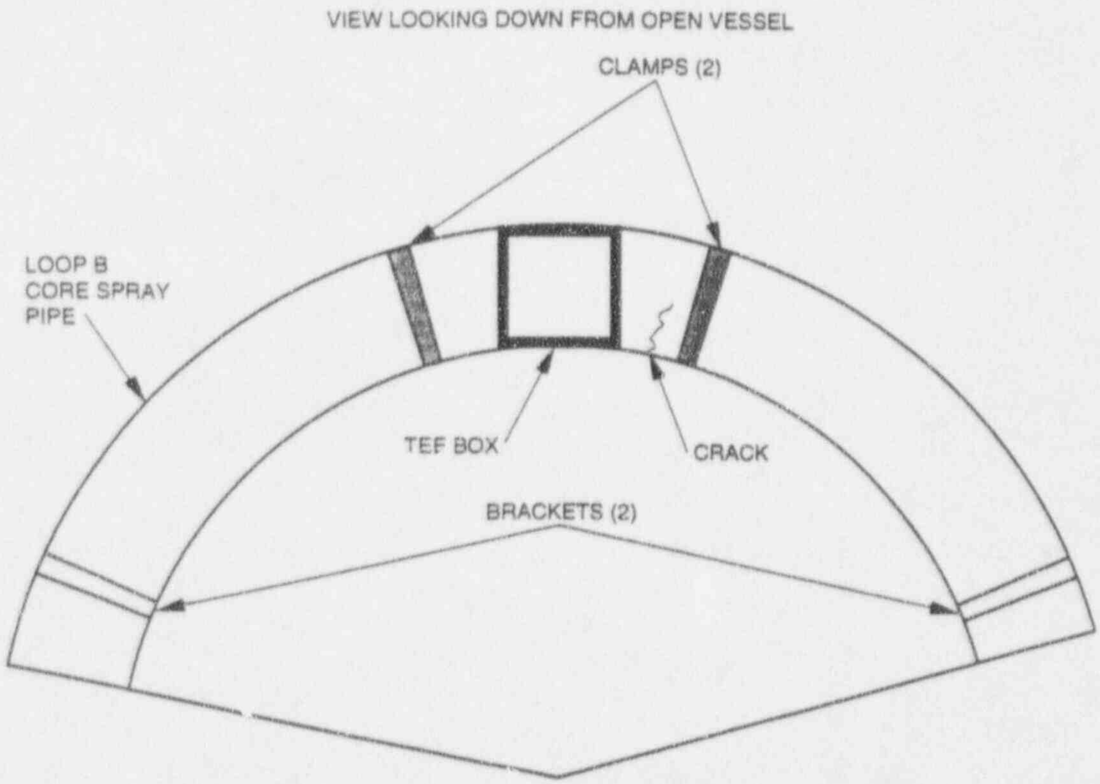
Attachments: Figure 1 - Top View of 'B' Core Spray Header

Analysis of Postulated Pipe Break in Quad Cities Unit 1 'B' Core Spray Header (V. McCarthy - GE Nuclear Energy to M. Richter letter dated July 13, 1994)

cc: J. Martin, Regional Administrator - Region III
C. Miller, Senior Resident Inspector - Quad Cities Station
C. Patel, Project Manager - NRR
W. Koo, Technical Staff - NRR
Office of Nuclear Facility Safety - IDNS

FIGURE 1

Top View of 'B' Core Spray Header



SCHEMATIC OF QUAD CITIES CORE SPRAY LINE SHOWS CRACK (APPROXIMATELY 120° OF PIPE CIRCUMFERENCE) NEAR TEE BOX, TWO BOLTED BRACKETS ±60° FROM TEE BOX, AND TWO ADDITIONAL WELDED CLAMPS ±10° FROM TEE BOX

Quad Cities Unit 1

ATTACHMENT

Analysis of Postulated Pipe Break in Quad Cities Unit 1 'B' Core Spray Header

GE Nuclear Energy

Date: July 13, 1994

To: M. Richter, Commonwealth Edison

From: V. McCarthy - Engineer, Structural Mechanics Projects
D. Shen - Principal Engineer, Regulatory Services

Subj: Analysis of Postulated Pipe Break in Quad Cities Unit 1 'B' Core Spray Header

GENE-523-A80-0594, "Core Spray Crack Analysis for Quad Cities 1 and 2", provides a structural, lost parts, and ECCS analysis for the crack detected at the tee box location in the Quad Cities Unit 1 'B' core spray header. Based on the structural analysis, it was determined that the Quad Cities core spray line crack would not propagate to more than 180° of the pipe circumference.

Although the core spray crack should not result in a pipe break, an analysis was performed to determine the resultant displacement of the pipe ends, and the leakage rate from the pipe, if a 360° pipe break was postulated. The results from the leakage analysis were then applied to a reactor internals evaluation for Unit 1, which considered the LOCA analysis impact based on the current condition of reactor vessel internals identified during the March 1994 outage.

Leakage Analysis

The finite element ANSYS model which was constructed previously for the 'B' core spray header configuration to support the flaw evaluation (presented in GENE-523-A80-0594), was utilized in determining the leakage flow from a 360° through wall crack in the header. The model considered the following loads:

Impingement Load - load due to downcomer flow acting on core spray line which is distributed to nodes comprising horizontal arms of core spray header

$\mathbf{F_z = -1642\ lb}$

Dead weight and Seismic Accelerations - accelerations due to dead weight load and safe shutdown earthquake (SSE) (including a 1.5 factor of safety)

$\mathbf{a_x = 1.08\ g}$

$\mathbf{a_y = 1.08\ g}$

$\mathbf{a_z = -1.24\ g}$

Pressure Stress - stress on the core spray pipe due to a conservative differential pressure of 150 psi

$\boldsymbol{\sigma_p = 776}$ **psi**

Flow Stress - stress on core spray pipe due to a core spray flow of 4500 gpm (including a 2.0 factor of safety)

$\boldsymbol{\sigma_f = 348}$ **psi**

Figure 1 shows the finite element model which includes the following boundary conditions:

Nodes 1, 49, 54:	completely fixed
Nodes 13, 37:	fixed in vessel radial direction for bolted vessel brackets
Nodes 23,27:	fixed in vessel radial and vertical directions for welded vessel clamps

This model was next modified to include a 360° pipe break at the crack location. A flow load force, F, was determined for the break location utilizing the design flow conditions:

$$F = \rho A v^2 / g + (P - P_{inf}) A$$

$$Q = Av$$

where: $\rho = 62.4\ lb/ft^3$

A = cross sectional area of 6" SCH 40 pipe = 28.9 in^2

$P-Pinf$ = 64 psid (GE Drawing 161F312, Rev. 1 - limiting ΔP for a maximum core spray flow of 5350 gpm)

Q = rated core spray flow = 4500 gpm

Based on the above inputs, F was calculated to be 2883 lb. This load was applied to the node at the break location, and the model was rerun, yielding the following displacements of the pipe ends:

x displacement (in)	0.010
y displacement (in)	0.343
z displacement (in)	0.017
x rotation ($^\circ$)	0.002
y rotation ($^\circ$)	1.15

From these displacements, a total area through which leakage could occur, the "leakage area", A_L, was calculated to be 2.2 in^2. Since the displacement was small, the velocity of the leakage flow was calculated using the A and ($P-P_{inf}$) from above, along with the following equations:

$$v_L = [(P-P_{inf})\, 2g / \rho]^{1/2}$$
$$Q_L = A_L v$$

As a result, the leakage flow rate for a 360° through wall crack was calculated to be 656 gpm.

Reactor Vessel Internals Evaluation

During the Quad Cities Unit 1 March 1994 refueling outage, crack indications were identified in the following reactor vessel internal components:

1. circumferential cracking indications in the core shroud, including the H5 circumferential weld;

2. 'B' core spray header crack indication in the heat affected zone of the weld which joins the header pipe and junction (tee) box; and

3. radial and circumferential cracking indications in the heat affected zone of the jet pump riser and riser brace weld for jet pumps 5/6 on loop 'A' of the recirculation system.

In addition, during the refueling outage, a reactor recirculation runout event for the 'B' loop pump could have resulted in potential jet pump cavitation.

The reactor internals evaluation addresses the LOCA analysis impact from the crack indications, as well as the recirculation runout event. For this evaluation, the Recirculation Line Break LOCA was selected as the bounding accident because it results in the highest peak cladding temperature (PCT). A SAFER/GESTR peak cladding temperature (ECCS) analysis was performed with the following conditions for the reactor vessel internals.

1. The shroud leakage flow was estimated to be 35 gpm through a postulated through-wall crack at the H5 weld during a design basis LOCA.[1]

2. The loop 'B' core spray header leakage from the existing crack was determined to be less than 50 gpm under design conditions; however, if the pipe is conservatively assumed to be broken, displaced and shifted, the leakage would be 656 gpm. The 50 gpm leakage rate is used below for ***Condition 'a'***, nominal coolant leakage, while the 656 gpm leakage rate is used for ***Condition 'b'***, very conservative coolant leakage.

3. The LPCI leakage through existing cracks in the recirculation loop 'A' jet pump riser is not expected to exceed 180 gpm during a design basis LOCA (used below for ***Condition 'a'***); for conservatism, an arbitrary leakage case of 1400 gpm was also considered (used for ***Condition 'b'***).[2]

4. As a result of the loop 'B' recirculation runout event, leakage from the loop 'B' internals was conservatively assumed to be equal to the loop 'A' leakage presented in the previous step. This estimate is bounding since visual inspections on loop 'B' following the runout event revealed no trace of any cracking along the jet pump riser pipes or in the riser brace welds, indicating that the loop 'B' recirculation line is intact.

This analysis assumed 1989 GE fuel characteristics (NEDC-31345P) for Quad Cities Unit 1, and other input parameters given in UFSAR Table 6.3-3*; these fuel characteristics are bounding for all GE-8 through GE-10 fuel for peak clad temperature (PCT) analyses. Based on the documented shroud crack indications, no shroud movement (lateral, vertical) is anticipated as a result of a recirculation line break LOCA.

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* Both LPCI and CS injection valve stroke times have been increased.

Two cases were evaluated because they apply to the worst single failure condition: 1) a 125v dc battery failure and 2) an LPCI injection valve failure case. Both of these cases were considered during the design basis accident of a complete severance of the recirculation loop suction piping. For the battery failure, the remaining contributing ECC subsystems are one core spray and two LPCI injection pumps into a single recirculation loop (see UFSAR Table 6.3-7). For the injection valve failure case, the remaining contributing ECC system are two core spray pumps.

Base Case 1 (battery failure), no cracks, App. K PCT = X°F
Base Case 2 (LPCI injection valve failure), no cracks, App. K PCT = Y°F
Condition 'a', nominal coolant leakage = Shroud leakage 35 gpm, core spray crack leakage 50 gpm, and riser leakage 180 gpm from 0.6 sq. in. crack indication
Condition 'b', very conservative coolant leakage = Shroud leakage 35 gpm, core spray header pipe complete separation 656 gpm, and jet pump riser leakage 1400 gpm from a 4.7 sq. in. crack

Base Case 1 + coolant leakage condition 'a' = $X + \Delta PCT_{1xa}$
Base Case 1 + coolant leakage condition 'b' = $X + \Delta PCT_{1xb}$

Base Case 2 + coolant leakage condition 'a' = $Y + \Delta PCT_{2ya}$
Base Case 2 + coolant leakage condition 'b' = $Y + \Delta PCT_{2yb}$

ΔPCT_{1xa} and ΔPCT_{2ya} are both less than 20°F and therefore are considered to be negligible. The ΔPCT_{1xb} is less than 110°F, while the ΔPCT_{2yb} result is less than 80°F.

The above calculated results represent approximate peak cladding temperature increases as the result of the crack indications and also if large crack indications are assumed. Given that the no-crack licensing basis peak cladding temperatures X and Y are approximately 1600°F, it can be seen that the PCT increase as the result of assuming conservative cracks is tolerable in view of the 2200°F limit in 10CFR50.46.

<u>References</u>

1. 'Safety Assessment of Core Shroud Indication for Cycle 14 Operation of Quad Cities Unit 1', GE-NE-523-A92-0694, June 1994.
2. 'Stress Analysis and Fracture Mechanics Evaluation for the Jet Pump Riser Brace Repair', GE-NE-523-A76-0594, May 1994.

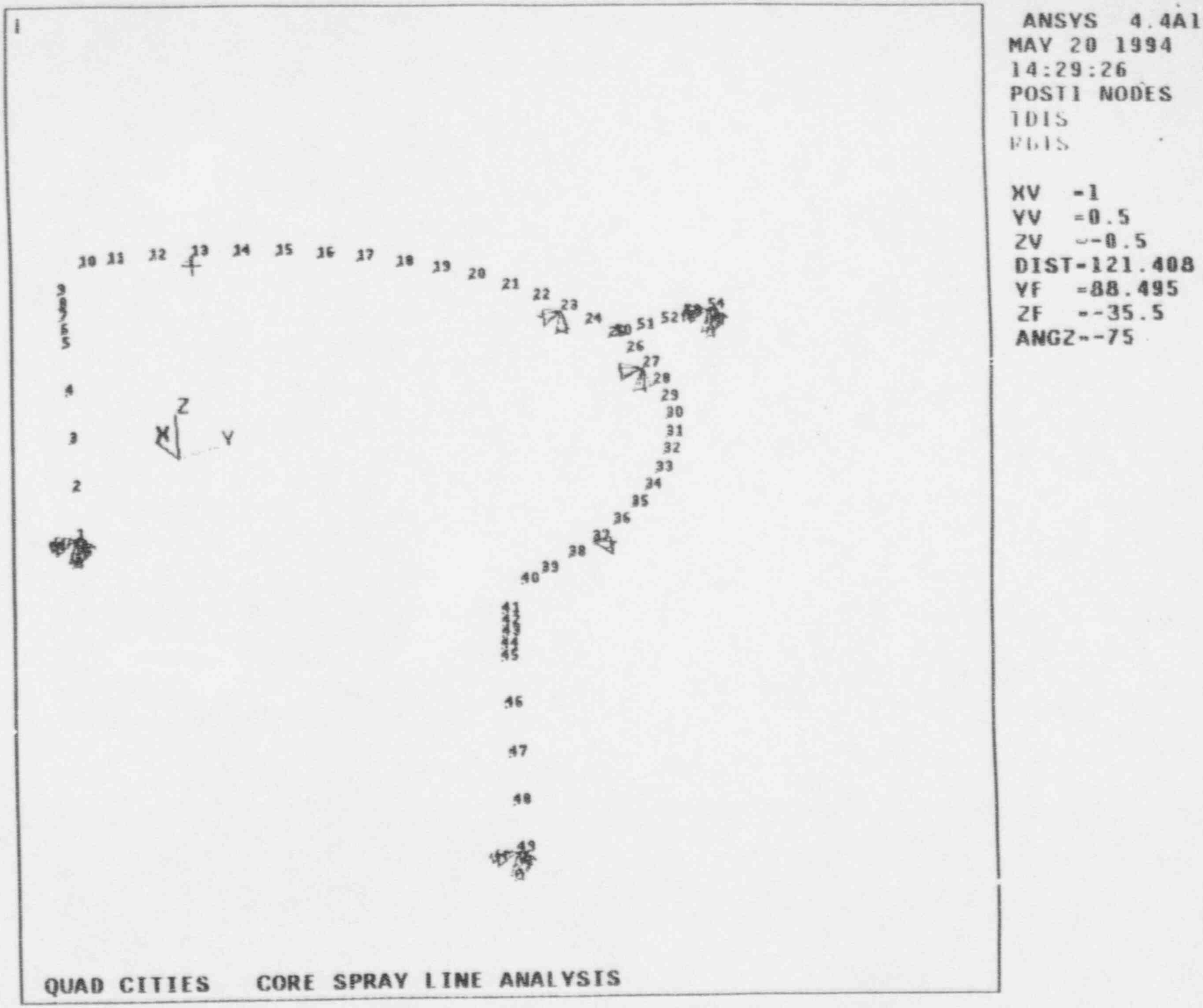

Figure 1. Finite element model for Quad Cities Units 1 and 2 core spray lines

Prepared by: ______________________
V.L. McCarthy
Engineer, Structural Mechanics Projects

Prepared by: ______________________
D.T. Shen
Principal Engineer, Regulatory Services

Concurred by: Gary L. Stevens 7/13/94
G.L. Stevens
Senior Engineer, Structural Mechanics Projects