Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

January 8, 1982

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dredsen Station Units 2 and 3
Quad Cities Station Units 1 and 2
Zion Station Units 1 and 2
NUREG 0737 Items Requiring a
January 1, 1982, Submittal
NRC Docket Nos. 50-237/249,
50-254/265 and 50-295/304

References (a): D. L. Peoples letter to J. G. Keppler dated January 18, 1980.

(b): D. L. Peoples letter to J. G. Keppler dated January 25, 1980.

(c): D. G. Eisenhut letter to All Licensees dated October 31, 1980.

(d): J. S. Abel letter to D. G. Eisenhut dated December 15, 1980.

(e): L. O. DelGeorge letter to D. G. Eisenhut dated July 1, 1981.

(f): E. D. Swartz letter to D. G. Eisenhut dated December 15, 1981.

(g): T. J. Rausch letter to D. G. Eisenhut dated December 29, 1981.

Dear Mr. Eisenhut:

In response to the requirement set forth in Reference (c) enclosed are the outstanding Commonwealth Edison Company responses to those NUREG 0737 items requiring a January 1, 1982, submittal for our Dresden, Quad Cities and Zion Stations.

The enclosure to this letter includes a response for those items requiring a January 1, 1982, "Licensee Submittal" where we have not previously provided one. This includes documenting various Owners Group submittals that are applicable to our stations. Additionally, a response has been provided for Item II.F.1.3 that was inadvertently omitted from our Reference (f) request for implementation schedule postponement.

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Finally, a response has been provided for Item II.K.3.29. The Commonwealth Edison Company has reviewed the NRC Safety Evaluations that we received for our Dresden Units 2 and 3. In our judgement, we do not believe it is necessary or prudent to include the isolation condenser vents as part of Item II.B.1 and we do not intend to prepare specific venting procedures for operator use at this time.

To the best of my knowledge and belief, the statements contained herein and in the enclosure are true and correct. In some respects, these statements may not be based upon my personal knowledge, but upon information furnished by other Commonwealth Edison employees and consultants. This information has been reviewed in accordance with Company practice and I believe it to be reliable.

Please address any questions that you or your staff may have concerning this matter to this office.

One (1) signed original and seventy-nine (79) copies of this letter and enclosure are provided for your use.

Very truly yours,

E. Douglas Swartz

E. Douglas Swartz
Nuclear Licensing Administrator

Enclosure

cc: J. G. Keppler, Reg. Adm. - Region III
Region III Inspector - Dresden
Region III Inspector - Quad Cities
Region III Inspector - Zion

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ENCLOSURE

COMMONWEALTH EDISON COMPANY

Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Zion Station Units 1 and 2

Outstanding responses to those NUREG 0737 Items that require a January 1, 1982, "Licensee Submittal"

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II.E.1.1 Auxiliary Feedwater System Evaluation

Zion Response:

See Reference (e). All NUREG 0737 requirements for this item have been met.

There is one final AFW modification that originated from the "lessons-learned" long-term recommendations that is currently in progress. This modification, not under the schedule implementation requirements of NUREG -0737, provides for the installation of check valves in the auxiliary feedwater lines to preclude steam generator blowdown in the event of an auxiliary feedwater line rupture. Since this modification is outage related and involves major piping re-routing and installation of long lead time equipment, the current completion schedule is Fall of 1982 for Unit #2 and Spring of 1983 for Unit #1.

II.F.1.3 Containment High-Range Rad Monitor

Zion Response:

References (f) and (g) were submitted to request Commission approval to postpone some items due January 1, 1982, that the Commonwealth Edison Company could not meet. This item was inadvertently omitted. Commission approval is hereby requested to postpone completion of this item to April 1, 1982, which will accommodate our pre-operational testing requirements. This was verbally discussed with D. L. Wigginton on January 7, 1982.

II.F.1.4 Containment Pressure Monitor

Dresden, Quad Cities and Zion Response:

NUREG 0737 requires Licensees to inform the NRC when the required design modifications have been completed. This modification has been completed at our Quad Cities and Zion Stations. Reference (d) provided the associated design information. The final installation and system design documents are available for NRC review upon request. (One instrument at Quad Cities is experiencing a calibration drift problem which is being rectified through normal instrument maintenance procedures).

At Dresden Station, the equipment is installed but not functioning. Due to inclusion of this equipment in a modification package with equipment having a delayed installation schedule (containment water level Item II.F.1.5), the containment pressure equipment was inadvertently not calibrated and not turned over for operation. The Dresden 2 equipment will be functioning by 1/15/82 and Dresden 3 equipment will be operable prior to startup from the current refueling outage, which began on 1/02/82.

II.F.1.5 Containment Water Level Monitor

Zion Response:

This modification is installed and operable in accordance with the NUREG 0737 implementation schedule. Installation and system design documents are available for NRC review upon request.

II.F.1.6 Containment Hydrogen Monitor

Zion Response:

This modification is installed and operable in accordance with the NUREG 0737 implementation schedule. Installation and system design documents are available for NRC review upon request.

II.K.2.13 Thermal Mechanical Report-Effect of High-Pressure Injection on Vessel Integrity for SB LOCA with No Auxiliary Feedwater

Zion Response:

Reference (10) - Letter OG-66, dated December 30, 1981, O.D. Kingsley, Jr. (Chairman, Westinghouse Owners Group) to H. R. Denton (NRC).

This item requires a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater. Westinghouse has performed an analysis for generic Westinghouse plant groupings to address this issue. Reference (10) transmitted WCAP-10019 entitled "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants" to the NRC which provides a conservative assessment of reactor vessel integrity for all operating Westinghouse reactors.

II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Zion Response:

Reference (1) - Letter OG-57, dated April 20, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to P. S. Check (NRC).

Reference (2) - Letter OG-64, dated November 30, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. G. Eisenhut (NRC).

Westinghouse has performed a study which addresses the potential for void formation in Westinghouse designed Nuclear Steam Supply Systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC (Reference (1)) and is applicable to Zion Station. In addition, a natural circulation cooldown guideline has been developed that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/ depressurization transients, and specifies those conditions under which upper head voiding may occur. These generic guidelines have been submitted to the NRC (Reference (2)). The generic guidance will be utilized in the implementation of Zion Station plant specific operating procedures.

II.K.3.5 Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident

Zion Response:

Reference (4) - "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply Systems," WCAP-9584 (Proprietary) and WCAP-9585 (Non-Proprietary), August 1979.

Reference (5) - Letter OG-49, dated March 3, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. F. Ross, Jr. (NRC).

Reference (6) - Letter OG-50, Dated March 23, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. F. Ross, Jr. (NRC).

Reference (7) - Letter OG-60, dated June 15, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to P. S. Check (NRC).

Westinghouse has performed an analysis of delayed reactor coolant pump trip during small-break LOCAs. This analysis is documented in Reference (4). In addition, Westinghouse has performed test predictions of LOFT Experiments L3-1 and L3-6. The results of these predictions are documented in References (5), (6) and (7).

Based on: 1) The Westinghouse analysis, (2) The excellent prediction of the LOFT Experiment L3-6 results using the Westinghouse analytical model, and (3) Westinghouse simulator data related to operator response time, the Westinghouse and Commonwealth Edison Company position is that automatic reactor coolant pump trip is not necessary since sufficient time is available for manual tripping of the pumps.

Our understanding of the schedule for final resolution of the issue is:

a) Once the NRC formally approves the Westinghouse model, a 3-month study period will ensue during which the Westinghouse Owners Group will attempt to demonstrate compliance with some NRC acceptance criteria for manual RCP trip. The NRC acceptance criteria will accompany their formal approval of the Westinghouse models.

b) If, at the end of the 3-month period the Westinghouse Owners Group cannot show compliance with the acceptance criteria, the NRC will formally notify Commonwealth Edison that we must submit an automatic RCP trip design.

II.K.3.25 Effect of Loss of AC Power on Pump Seals

Zion Response:

See Reference (d). This item requires that the consequences of a loss of RCP Seal Cooling due to a loss of AC power (defined as loss of off-site power) for at least 2 hours be demonstrated.

During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power, the RCP motor is deenergized and both of these cooling supplies are terminated; however, the diesel generators are automatically started and both seal injection flow and component cooling water to the thermal barrier heat exchanger is automatically restored within approximately 30 seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during a loss of offsite power for at least 2 hours.

II.K.3.28 Verify Qualification of Accumulators on ADS Valves

Dresden and Quad Cities Response:

Reference (d) indicated that this item would be addressed by the BWR Owners Group. The following is provided in lieu of an Owners Group response.

The NRC position for the subject topic states:

> "Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable."

At our Dresden 2/3 and Quad Cities 1/2 units, only one of five ADS valves require air or nitrogen to function in the ADS mode. The other four valves are powered by redundant 125 Vdc power supplies and do not require air or nitrogen.

Our response to IE Bulletin 80-01 (References (a) and (b)) demonstrated that the accumulators for the one pneumatically operated valve on each unit have sufficient capacity, including leakage, to allow the valves to perform their ADS function.

With respect to long-term (100 days) function, the four electrically operated valves provide more capacity than is required for this mode of operation, so that long-term availability of the accumulators is not required.

The environmental qualification of the pneumatic and electric actuated valves is being addressed as part of the SEP and IE Bulletin 79-01B activities.

Based upon the above, Commonwealth Edison believes the existing ADS valves and accumulators at Dresden 2/3 and Quad Cities 1/2 satisfy the requirements of Item II.K.3.28, and no modifications are necessary (unless requirements develop as a result of environmental qualification program activities).

II.K.3.29 Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles

Dresden Response:

The Commonwealth Edison Company has reviewed the NRC safety evaluations that we received for this item, D. M. Crutchfield letter to L. DelGeorge dated September 15, 1981, (Dresden Unit 2) and T. A. Ippolito letter to L. DelGeorge dated September 16, 1981, (Dresden Unit 3) and has the following comments:

We concur with the safety evaluation conclusion that, based on the existing design incorporating tube side vents and past operating experience, it is not necessary to demonstrate the adequacy of the isolation condenser to operate with noncondensible gases present.

However, we do not agree with the statement made in each safety evaluation transmittal letter that these valves be considered part of Item II.B.1, Reactor Coolant System Vents. Our previous responses to this item (References (d) and (e)) indicated that venting which normally occurs during HPCI and/or ADS operation was sufficient and other vent paths are only considered as backup for these already adequate systems.

Additionally, we do not believe that use of the isolation condenser tube-side vents as reactor coolant systems vents is prudent since the vents exhaust outside of the primary containment and containment isolation signals would have to be defeated to open the valves.

Finally, as stated in our previous response (Reference (e)), procedures specifically written to instruct reactor operators on using HPCI or ADS for venting purposes are not deemed necessary due to the inherent design of those systems. The existence of tube side vents on the isolation condenser has not altered this position, and our conclusion remains that no specific venting procedures are needed at this time.

Based on the foregoing discussion, we do not believe it is necessary or prudent to include the isolation condenser vents in Item II.B.1 and we do not intend to prepare specific venting procedures for operator use at this time.

II.K.3.30 Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K

and

II.K.3.31 Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46

Zion Response:

Reference (8) - Letter NS-TMA-2818, Dated September 26, 1980, T. M. Anderson (Westinghouse) to D. G. Eisenhut (NRC).

Reference (9) - Letter NS-EPR-2524, Dated November 25, 1981, E. P. Rahe (Westinghouse) to D. G. Eisenhut (NRC).

This item requires that the analysis method used by NSSS vendors and/or fuel suppliers for small-break LOCA analysis for compliance with Appendix K to 10 CFR Part 50 be revised, documented and submitted for NRC approval.

Westinghouse feels very strongly and Commonwealth Edison Company agrees that the small-break LOCA analysis model currently approved by the NRC for use on Zion Station is conservative and in conformance with Appendix K to 10 CFR Part 50. However, (as documented in Reference (8)) Westinghouse believes that improvement in the realism of small-break calculations is a worthwhile effort and has committed to revise its small-break LOCA analysis model to address NRC concerns (e.g., NUREG-0611, NUREG-0623, etc.). This revised Westinghouse model is currently scheduled for submittal to the NRC by April 1, 1982, as documented in Reference (9).

Dresden and Quad Cities Response:

As stated in Reference (d), response to this item was handled directly by the General Electric Co. Based on our discussions with G.E. and the BWR Owner's Group, we believe that all NRC concerns have been addressed by G.E. and no additional response is required.

The following is provided to formally document our current position. The General Electric Co. has determined and documented to the NRC (R.H. Buckholz letter to D. G. Eisenhut dated June 26, 1981) that existing small break LOCA models and methods comply with 10 CFR 50, Appendix K, and do not require revision. Since Dresden 2 and 3 and Quad Cities 1 and 2 LOCA analyses use these existing models and methods, no additional plant specific calculations to show compliance with 10 CFR 50.46 are required either.