Commonwealth Edison
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November 21, 1979

Dr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden Station Units 1, 2 and 3
Quad-Cities Station Units 1 and 2
Zion Station Units 1 and 2
Commitments to Meet Near-Term
Requirements of the Lessons Learned Task
Force
NRC Docket Nos. 50-10/237/~~246~~ 249, 50-254/265,
and 50-295/304

References (1): H. R. Denton letter to all operating plants dated October 30, 1979

(2): C. Reed letter to D. G. Eisenhut dated October 18, 1979

(3): D. G. Eisenhut letter to all operating plants dated September 13, 1979

Dear Dr. Denton:

Commonwealth Edison has reviewed its commitments in light of the clarification and revised requirements contained in your October 30, 1979 letter. The enclosed supplementary and revised responses resulted from that review and should be incorporated into our October 18, 1979 letter on Lessons Learned commitments.

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

Very truly yours,

Cordell Reed
Manager of Nuclear Operations

enclosure

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ADD:
J OLSHINSKI
J KERRIGAN
J BURDOIN
C WILLIS
M FIELDS
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Replace the Last Paragraph of Section 2.1.1 with:

Dresden Units 2/3 and Quad-Cities Units 1/2 each have 4 relief valves and 1 safety/relief valve per unit. The safety/relief valve is provided with DC power and instrument air via an accumulator sized to ensure a minimum of 5 valve operations in the relief mode after any loss of air supply. Neither air nor DC power is required for the safety/relief valve to operate in the safety valve mode. The 4 relief valves per unit are each provided with safety grade DC power and by design do not require air to operate. Since the existing design provides assurance of long-term relief capability regardless of the maintenance of an air supply, no changes are proposed.

Replace the Second Paragraph of Section 2.1.2 with:

Commonwealth Edison will participate in the performance test program for relief and safety valves being developed by the General Electric BWR Owner's Group. It is anticipated that the BWR Owner's Group will make every effort to provide a program description and test schedule by January 1, 1980.

Replace our response to Section 2.1.3b for Subcooling Meter with:

At Zion core subcooling in degrees Farenheit is presently displayed on a control board meter. The signal is derived by the unit's process computer using core exit thermocouples and reactor coolant system pressure as input. Operating personnel have been instructed not to make operational decisions based on this single plant parameter when confirmatory indications are available.

The digital process computer based system is a highly reliable indicator of core subcooling. Last year, the average process computer availability at Zion was 99%. Core temperature is read from an array of 65 thermocouples which measure temperatures just above the reactor core. Reactor coolant system pressure is determined from the average of four protection-grade pressurizer pressure instruments. Subcooling is re-computed every 32 seconds and is continuously displayed on a meter on the main control board. This calculation is performed routinely and is not bypassed, even if the alarm sequence typewriter is overloaded.

By the end of 1979, reactor coolant system pressure will be read from a 0 - 3000 psig indication channel rather than the average of four narrow-range channels. The process computer driven control board meter indication of subcooling will be replaced by indications of core temperature, saturation temperature based on system pressure, and saturation pressure based on core temperature. By the end of 1980, a second 0 - 3000 psig reactor coolant system pressure signal will be input to the process computer. Both 0 - 3000 psig reactor coolant system pressure indicators will be modified to the extent practicable to improve their environmental qualifications. Core temperature and saturation temperature and pressure calculations will be performed at the same frequency as indicated above for subcooling.

Other reliable instrumentation is also available to the operator such that subcooling can be determined manually in the unlikely event that the process computer is unavailable during a transient. All core exit thermocouples can be read manually on a permanently-mounted meter in the control room. Two hard-wired, analogue indications of reactor pressure are displayed and recorded on the control board. Similar hard-wired meters give continuous, recorded analogue indication of temperatures in the hot and cold legs of each of the four reactor coolant loops. These temperatures are derived from RTDs mounted directly in the main loop piping. Other narrow range RTDs are mounted in manifolds in small-diameter bypass lines on each leg. The analogue output of these RTDs is continuously displayed on the control board as ΔT across each loop.

Proximity makes the core thermocouples better indicators of core subcooling than the loop RTDs. The RTDs can be useful, however, in monitoring convective flow through the loops. Pressure varies so little through the reactor coolant system that the location of that instrumentation is not critical to subcooling determinations.

The proposed arrangement of continuous display of computer-calculated saturation conditions, backed by manual determination of subcooling from a variety of instruments hard-wired to the control room appears to satisfy NUREG-0578 requirements.

No meters are needed for the BWRs at Dresden and Quad Cities Stations because these units operate at saturated conditions.

Replace Section 2.1.6.a in its entirety with:

Commonwealth Edison has one difference with the recommendations of the October 30 NRC letter. This concerns the use of an integrated leakage test on systems to determine leakage. It is Commonwealth Edison's position that an integrated leak rate test is not effective for the specific concern of out leakage and is not practical, given the plant design considerations. In addition, integrated leak rate testing does not lend itself to an on-going maintenance program which would have the greatest benefit in a leak reduction program.

As an alternative, Commonwealth Edison proposes the following program:

1. Liquid systems will be visually inspected for leakage while systems are at approximate operating pressures. Gas systems will be evaluated using helium leak tests, pressure decay tests for specific tanks and metered make-up pressure tests.

2. The estimated leakage will be determined from these tests and will be reported to the NRC by January 1, 1980.

3. An aggressive maintenance program will be in place, using elements of the existing Total Job Management (TJM) program to assign high priorities to leakage related work requests. Essentially all leakage on concerned systems will be covered.

4. Systems lists will be available for review detailing specific methods used to test systems, the systems involved, frequency of testing and individuals responsible for testing.

5. The stations' Technical Staffs will review leakage-related work requests to evaluate possible modifications to keep leakage "as low as practical."

6. An annual report will be prepared at each station and submitted to the Nuclear Stations Division Manager. This report will include past years performance, current leakage rates, and status of leakage work requests and modifications.

The advantage of our program is that specific components requiring maintenance are identified during the visual surveillance. This facilitates maintenance action. Visual surveillances are supplemented by walkdowns during valve alignment procedures or other inspections. Leaks identified during these frequent inspections can be repaired rapidly thus enhancing the leak reduction effort.

Replace Section 2.1.6.a in its entirety with: (Continued)

In addition to the above, water inventory programs at the stations will be developed to allow trend analysis of leakage by monitoring sump levels, pump run times and tank inventories. In this manner, approximations of leakage may be made and performance of the overall program may be evaluated.

Insert after the First Paragraph of Section 2.1.7.b

The NRC's "clarification" letter dated October 30, 1979 contains a requirement for redundant indication of auxiliary feedwater flow rate.

Zion has a single auxiliary feed flow indicator for each steam generator, backed by multiple indications of steam generator level. We have no plans to install additional instrumentation.

The steam generator level is actually a more direct indication of decay heat removal capability in the Zion steam generators. The auxiliary feed rate is relatively small when compared to the normal inventory. The level indicators are, therefore, considered to be most useful in assessing reactor coolant status. Feed rate is of interest in verifying that additional water is being pumped to the steam generator, but this can also be inferred from a variety of other indicated parameters (steam generator level, pump discharge pressure, secondary storage tank level, etc.).

Replace Section 2.1.9.c, "Containment Water Level Monitor" in its entirety with:

Commonwealth Edison will install containment water level monitors, as described in Enclosure 1 of your October 30, 1979 letter at Zion, Quad-Cities and Dresden Units 2 and 3. It is anticipated that installation will be complete by January 1, 1981.

A supplement to our response to Section 2.2.1.b will be transmitted the week of November 26, 1979.

This supplement will explain in detail Commonwealth Edison's provision for a Shift Technical Advisor.

Replace 2.2.2.b in its entirety with:

Commonwealth Edison will establish an on-site technical support center at each of its operating nuclear stations by January 1, 1980.

Communications with the control room and the NRC will be completed in this time frame. Communications with the near-site emergency operations center will be established on a time frame consistent with the requirements of Enclosure 8 to the September 13, 1979 letter (mid-1980). Procedures will be written to cover the accident assessment function in the Technical Support Center (TSC) and the control room (should the TSC become uninhabitable). Procedures for prevention or reduction of radiation exposure to personnel will be revised or written as required. The direct display of plant parameters in the TSC may not be possible, given the short time frame between now and the end of the year. However, we will provide a procedure and direct communications between knowledgable individuals in both control room and TSC to ensure the reliable and timely transmittal of plant information to the TSC. By January 1, 1981, within the limits of equipment availability and scope of construction, the TSC will be upgraded to meet the recommendations of the Atomic Industrial Forum Subcommittee on Emergency Response Planning.