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February 22, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: LaSalle County Station Units 1 and 2
Regulatory Guide 1.97 Revision 2
Response to Iterim Report by EG & G
NRC Docket Nos. 50-373 & 50-374

Reference (a) C. W. Schroeder to A. Schwencer letter dated June 29, 1982
(b) T. R. Tramm to H. R. Denton letter dated December 6, 1983
(c) A. Schwencer to D. L. Farrar letter dated December 13, 1984

Dear Mr. Denton:

Reference (c) contained the interim report by EG & G Idaho on Commonwealth Edison's response to Regulatory Guide 1.97 Revision 2. Included in the report was a request for additional information which was discussed with Mr. Bournia and Mr. Joyce of the NRC on February 20, 1985.

Attached are CECo's responses to the conclusions on page 11 of the interim report. One signed original and forty copies are provided for your use.

Sincerely,

G L Alexander

G. L. Alexander
Nuclear Licensing Administrator

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cc: A. Bournia
NRC Resident Inspector - LSCS

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ATTACHMENT

Conclusion 1

The licensee should provide the information identified in Section 2 of this report to document their commitment on conformance to Generic Letter 82-33 (Section 3.1).

Response

This question requests 8 items of information for each of the 69 variables listed in our Reference (a) submittal. Pages 2 and 3 of that submittal addressed two of the items in summary form. The two relevant statements were "Seismic qualification of LaSalle equipment was completed to the IEEE 344-1975 Standards under the SQRT program" and "Edison will comply with the quality assurance requirements using its approved quality assurance program, as described in Topical Report CE-1 as revised.

The other six items requested can not be answered at this time. Reference (b) contains the approved LaSalle schedule for Detailed Control Room Design Review (DCRDR) and for Regulatory Guide 1.97 Revision 2. The DCRDR report is scheduled for submittal to the NRC by 11-01-85. The review will encompass the adequacy of instrumentation display, information, arrangement, and task analysis. If deficiencies are disclosed, instrumentation additions, deletions, relocations, or replacements will be required. It follows that without a final determination of instrumentation acceptability and arrangement, that submission of details such as instrument range, environmental qualification, redundance and sensor location, power supply, location of display, and schedule of installation or upgrade would be premature. I believe this fact was controlling when the schedule for submitting a Reg. Guide 1.97 final report (including schedule for installation) was established and approved as 08-01-86.

After the NRC reviews our DCRDR report we will be able to reevaluate the schedule for submitting the requested information.

Conclusion 2 (Reviewer does not agree)

Neutron flux--the licensee's present instrumentation is acceptable on an interim basis until Category 1 instrumentation is developed and installed (Section 3.3.1).

Response

CECo is pursuing two courses of action. Currently we are reviewing whether recently developed equipment meets Reg. Guide 1.97 Rev. 2. Also in conjunction with the DCRDR we will evaluate whether this parameter is required or can be classified as a Category 3 variable.

Conclusion 3

Radiation exposure rate--the licensee should show that the ranges supplied for this variable encompass the radiation level at the instrument location (Section 3.3.4).

Response

Revised 02-11-85

ISSUE 11. VARIABLE E2

E2: Reactor Building or Secondary Containment Radiation

Issue Definition

Regulatory Guide 1.97 specifies that "Reactor building or secondary containment area radiation" (variable E2) should be monitored over the range of 10^{-1} to 10^{4} R/h for Mark I and II containments, and over the range of 1 to 10^{7} R/hr for Mark III containments. The classification for Mark I and II is Category 2; for Mark III, the classification is Category I.

Discussion

As discussed in the variable C14 position statement (Issue 6), Secondary Containment Area Radiation is an inappropriate parameter to use to detect or assess primary containment leakage.

Conclusion

It is Edison's position that the instrumentation for this variable is not needed as the plant noble gas effluent monitors are more useful and practical in detecting the primary containment leakage. The range of the noble gas effluent monitors at LaSalle is 10^{-7} uci/cc to 10^{5} uci/cc which meets the RG 1.97 requirements.

Conclusion 4

Status of standby power and other energy sources--the licensee should show that the status is monitored for all recommended power sources (Section 3.3.7).

Response

Both on-site and off-site power sources are monitored for status per the requirements of Reg. Guide 1.97 Rev.2.

Conclusion 5

Reactor building or secondary containment area radiation--the licensee should supply additional justification for this deviation (Section 3.3.8).

Response

Revised 02-11-85

E3: Radiation Exposure Rate

Issue Definition

Regulatory Guide 1.97 specifies in Table 1, variable E3, that radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety) be monitored over the range of 10^{-1} to 10^{4} R/hr for detection of significant releases, for release assessment, and for long term surveillance.

Discussion

In general, access is not required to any areas of the secondary containment to service equipment important to safety in a post-accident situation. If and when accessibility is reestablished in the long term, it will be done by a combination of portable radiation survey instrument and post-accident sampling of the secondary containment atmosphere. The radiation exposure rate monitors inside secondary containment at LaSalle County Station are for normal operation and are not intended to continually monitor gross intrusions of the reactor's fission products into secondary containment. These monitors are installed in low or moderate radiation areas to identify abnormal occurrences which would produce radiation environments ranging from a few millirads per hour to several rads per hour as shown in FSAR Table 12.3-13. Abnormal occurrences are any incidents that do not propagate into any of the accidents discussed in Chapter 15 of the FSAR.

The following list of FSAR information contains the data used to establish the ranges of the monitors in FSAR Table 12.3-13, "Area Radiation Monitors."

1. FSAR Table 12.3-3, "Reactor Building Design Data"

2. FSAR Figures 12.3-1, Sht. 2 through Sht. 8, "Radiation Zones During Full Power Operation"

3. FSAR Figures 12.3-2, Sht. 2 through Sht. 8, "Radiation Zones During Shutdown"

4. FSAR Figures 12.3-3, Sht. 2 through Sht. 8. "Shielding Drawings" (It shows monitor locations.) (Note: These figures are not in the USFSAR.)

Conclusion

It is Edison's position that the specified radiation exposure rate monitors inside the Reactor Building are used only to measure the radiation dose rates during normal operation and to detect abnormal occurrences which do not constitute an accident while the reactor is critical. One area monitor is designed to detect and monitor fuel handling incidents (including fuel handling accidents) and has an adequate range up to 10^3 rads/hr. Seven Reactor Building monitors have a range up to 10 rad/hr. The remaining twenty monitors have an upper range of 1 rad/hr or 0.1 rad/hr.

Because the LaSalle design does not require access to a harsh environment area to service safety-related equipment during an accident, this dose rate variable is only used to determine abnormal occurrences and is provided from existing area radiation monitors. This parameter is reclassified as Category 3 and the monitors furnished for this variable have ranges that encompass the expected radiation levels at their locations.

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