Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

April 12, 1991

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Quad Cities Nuclear Power Station Unit 2
Submittal of Relief Request for
Inservice Inspection Program
NRC Docket No. 50-265

References: (a) Conference Call on April 12, 1991 between CECo (M. Richter) and NRR (B. Siegel).

(b) M. H. Richter (CECo) to T. E. Murley (NRC) letter dated April 5, 1991.

(c) J. Zwolinski (NRC) to D. Farrar (CECo) letter dated March 31, 1986.

Dr. Murley:

As discussed with your staff in the Reference (a) teleconference, the bonnet assembly on the Reactor Core Isolation Coolant (RCIC) System Steam Isolation Valve (2-1301-16) was replaced during the current forced maintenance outage for Unit 2. As a result of this work, an ASME Code Class 1 system leakage test is required by Section XI. Based on the personnel exposure and critical path outage impact for the performance of the Class 1 system leakage test, Commonwealth Edison Company (CECo) is submitting the attached Unit 2 relief request (CR-15) to the Inservice Inspection (ISI) Program, to support the startup of Unit 2. Recently, CECo has submitted an ISI Program relief request (CR-14) which would address the current situation and avoid case-by-case relief requests in the future (Reference (b)). The Safety Evaluation Report for the current ten-year ISI Program interval was transmitted by Reference (c).

CECo requests verbal approval of this relief request to support Unit 2 startup on April 16, 1991. CECo appreciates the prompt attention that has been given by your staff to this matter.

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Please direct any questions or comments on this letter to this office.

Respectfully,

Milton H. Richter

M. H. Richter
Nuclear Licensing Administrator

Attachment: Relief Request Number CR-15

cc: A.B. Davis - Regional Administrator, Region III
L.N. Olshan - NRR Project Manager
T.E. Taylor - Senior Resident Inspector, Quad Cities
R.A. Hermann - NRR Technical Staff

MR:lmw
ZNLD703/5

ATTACHMENT

RELIEF REQUEST NUMBER: CR-15

COMPONENT IDENTIFICATION

Code Class:	ONE
References:	Article IWA-5211(a) Article IWB-5221(a)
Examination Category:	B-P
Item Number:	B15.70
Description:	System leakage test pressure for RCIC System Steam Isolation Valve (2-1301-16) bonnet assembly replacement.

CODE REQUIREMENT

IWA-5211(a) requires a system leakage test to be conducted following opening and reclosing of a component in the system after pressurization to nominal operating pressure.

IWB-5221(a) states that the system leakage test shall be conducted at a test pressure not less than the nominal operating pressure associated with 100% rated reactor power.

BASIS FOR RELIEF

During the current Unit 2 forced maintenance outage, Quad Cities Station replaced the bonnet assembly on the Reactor Core Isolation Coolant (RCIC) System Steam Isolation Valve (2-1301-16). This replacement involved the disassembly and reassembly of a Class 1 mechanical connection which is located in the drywell and cannot be isolated from the reactor vessel. For this situation, the performance of a Class 1 system leakage test at 1,000 psig (which is the nominal operating pressure at 100% rated reactor power) would have a significant impact on the unit's critical path outage time and personnel exposure.

The normal Class 1 system leakage test, which is performed with the vessel flooded up, requires numerous equipment outages (e.g., in excess of 300 valves must be taken out-of-service, Main Steam safety valves must be gagged). Performance of the equipment outages, coupled with the performance of the system leakage test, takes approximately 5 days (3 shifts per day) with a total personnel exposure of approximately 2.5 Man-Rem.

Performance of the system leakage test using reactor pressure during unit startup is possible, however, the test can not be performed at 1,000 psig. During unit startup, the Electro-Hydraulic Control System precludes a reactor pressure above 950 psig without significant increases in reactor power. In order to achieve a pressure of 1,000 psig, the reactor would have to be at approximately 100% rated power. The radiation levels in the drywell at this power level are prohibitive, and prevent drywell entry by plant personnel.

A drywell entry to inspect for leakage can be performed at 920 psig, which is associated with approximately 15% reactor power. Performance of the leakage test in this manner would have an insignificant impact on the ability to detect leakage from the reassembled mechanical connection. It would also significantly reduce the personnel exposure and critical path outage time required for the test.

PROPOSED ALTERNATE EXAMINATION

For the RCIC System Steam Isolation Valve (2-1301-16) bonnet assembly, the Class 1 mechanical connection will be pressurized to 920 psig and inspected for leakage during unit startup.

APPLICABLE TIME PERIOD

Relief is requested for the RCIC System Steam Isolation Valve (2-1301-16) bonnet assembly replacement which occurred during the short Unit 2 forced maintenance outage. Unit 2 startup is currently scheduled for April 16, 1991.