Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

January 17, 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: Dresden Nuclear Power Station Unit 3
Submittal of Relief Request for
Inservice Inspection Program
NRC Docket No. 50-249

References: (a) Conference Call on January 15, 1991 between CECo (M. Richter, G. Whitman) and NRR (R. Hermann).

(b) Conference Call on January 16, 1991 between CECo (M. Richter) and NRR (L. Olshan, R. Hermann).

(c) D. Crutchfield (NRC) letter to D. Farrar (CECo), dated April 26, 1984.

Dr. Murley:

As discussed with your staff in the Reference (a) and (b) teleconferences, the pilot valve assembly was replaced on the Target Rock Safety-Relief Valve during the current forced, mid-cycle outage for Unit 3. 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, CECo is submitting the attached Unit 3 relief request (CR-11) for the Inservice Inspection Program, which is currently in its second ten-year interval (March 1, 1982 to March 1, 1992). The Safety Evaluation Report (SER) for this second ten-year interval was transmitted by Reference (c).

CECo requests verbal approval of this relief request to support Unit 3 startup on January 17, 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-11

cc: A.B. Davis - Regional Administrator, Region III
B.L. Siegel - NRR Project Manager
D.E. Hills - Senior Resident Inspector, Dresden
R.A. Hermann - NRR Technical Staff

MR:lmw
ZNLD703/5

ATTACHMENT

RELIEF REQUEST NUMBER: CR-11

COMPONENT IDENTIFICATION

Code Class:	1
References:	Article IWA-5211(a) Article IWB-5221(a)
Examination Category:	B-P
Item Number:	B15.70
Description:	System leakage test pressure for Target Rock Safety-Relief Valve pilot valve 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

The nominal operating pressure associated with 100% rated reactor power is 1,000 psig. At the end of each refueling outage a system leakage test of all Class 1 pressure retaining components is conducted at this pressure.

During the current forced, mid-cycle outage on Unit 3, the Target Rock Safety-Relief Valve pilot valve assembly was replaced. The replacement requires the disassembly and reassembly of a bolted mechanical connection to the safety-relief valve. For this situation, performance of a Class 1 system leakage test at 1,000 psig would have a significant impact on the unit's critical path outage time and personnel exposure.

In order to perform the normal Class 1 system leakage test, 383 valves must be taken out of service, the safety valves must be gagged and the vessel flooded up. 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 1,000 psig the reactor would have to be at approximately 100% rated power. The dose rates experienced in the drywell at this power level are prohibitive and would pose a great risk to the health and safety of the workers.

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 connection. It would also significantly reduce the personnel exposure and critical path outage time required for the test.

PROPOSED ALTERNATE EXAMINATION

For each Unit 3 mid-cycle outage in which the Target Rock Safety-Relief Valve pilot valve assembly is replaced, the bolted mechanical connection will be pressurized to 920 psig and inspected for leakage during unit startup.

APPLICABLE TIME PERIOD

Relief is requested for Unit 3 for the second ten-year interval (March 1, 1982 to March 1, 1992).