Commonwealth Edison
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Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

July 29, 1988

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk,
Washington, D.C. 20555

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
LaSalle County Station Units 1 and 2
Response to Generic Letter 88-01
Docket Nos. 50-237/249, 50-254/265,
50-373/374

Reference: (a) Generic Letter 88-01 dated
January 25, 1988

Dear Sir:

The above referenced NRC Generic Letter (a) requested that all licensees of Boiling Water Reactors and holders of construction permits for BWRs, furnish-current plans relating to piping replacement, inspection, repair and leakage detection in regards to the implementation of the new staff position on IGSCC in BWR Austenitic Stainless Steel Piping.

Commonwealth Edison has completed its review pursuant to the request outlined in the Generic Letter for Dresden, Quad Cities and LaSalle County Stations. This information is attached in Enclosure 1.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

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

Respectfully,

Wayne E Morgan

W.E. Morgan
Nuclear Licensing Administrator

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Attachments

cc: A.B. Davis
Resident Inspectors - D/QC/LSC

Subscribed and Sworn to
before me this 29th day
of July, 1988

Nellie A. Mays
Notary Public

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ENCLOSURE 1

Dresden, LaSalle, Quad Cities Stations,
Response to Generic Letter 88-01

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A. DRESDEN UNIT 2

I. Item 1 Response: Commonwealth Edison Co. (CECo.) is presently pursuing an Integrated Program to mitigate the occurrence of IGSCC and its effects at this unit. The program involves the use of four (04) primary mitigators:

1. Hydrogen Water Chemistry - (HWC) has been implemented since 1983 on a trial basis. Examination of the recirculation welds to date has shown positive benefit of Hydrogen Water Chemistry. HWC will continue to be used at this unit.

2. Stress Improvement is being considered for this unit in order to achieve a very low risk of future IGSCC.

3. Weld Overlay has been and is being used to reinforce welds that have flaw indications in excess of the ASME Section XI, Subsection IWB 3500 limits. Most of the weld overlay reinforcements have been applied and inspected in accordance with NUREG-0313, revision 2.

4. System Removal is being investigated to reduce the population of IGSCC susceptible welds in this unit. A potential candidate is the CRD Return line.

II. Item 2 Response: Beginning with the next refueling outage for Dresden Unit 2 (September 1988), ASME Code class 1, 2, and 3 piping made of stainless steel that is four (04) inches or larger in nominal diameter and contains reactor coolant at a temperature above 200° F during power operation will be subjected to an augmented inspection program. This augmented inspection program will conform to the NRC staff position on methods and personnel and sample expansion delineated in Generic Letter 88-01.

Dresden Station requests that a factor of two (2) in reduction of inspection frequency be given to IGSCC categories B, C, D, E, and G weldments examined in 1983 and 1984 for the following reasons:

1. Hydrogen water chemistry (HWC) has been implemented at Dresden Unit 2 since April 1983.

2. Improvement of the reactor water conductivity performance from an acceptable average of approximately 0.2 micromho to an exceptional level of 0.06 micromho. Table-1 contains sample of conductivity and reactor coolant dissolved oxygen readings for Dresden Unit 2 during the months of March and April 1988.

TABLE-1

DRESDEN U.2 REACTOR WATER CHEMISTRY

MARCH 1988				APRIL 1988			
Day	Power (MWTH)	Conductivity (micromhs)	Dissolved O_2 (ppb)	Day	Power (MWTH)	Conductivity (micromhs)	Dissolved O_2 (ppb)
1	2368	0.058	4.4	1	2378	0.061	
2	2257	0.062	185	2	2398	0.058	
3	2341	0.058	2.8	3	2374	0.058	
4	2243	0.058	2.6	4	2402	0.060	4.9
5	1146	0.060		5	2401	0.060	4.7
6	1810	0.059		6	2380	0.060	2.9
7	2353	0.058	1.2	7	2238	0.058	2.9
8	2424	0.057	4.7	8	1707	0.063	1.3
9	2441	0.059	4.8	9	2375	0.062	
10	2337	0.067	205	10	2196	0.062	
11	2386	0.058	2.9	11	2191	0.057	3.0
12	2354	0.066		12	2196	0.060	
13	2348	0.060		13	2179	0.057	3.1
14	2393	0.059	2.4	14	936	0.062	98.0
15	2343	0.059	3.0	15	1889	0.058	1.3
16	2334	0.059	2.5	16	2428	0.060	
17	2314	0.058	2.5	17	2208	0.060	
18	2345	0.059	3.4	18	2412	0.057	4.3
19	899	0.065	1.0	19	2362	0.057	3.7
20	1704	0.060	3.9	20	2363	0.057	4.5
21	2076	0.058	187	21	2364	0.057	3.8
22	2415	0.058	5.2	22	2365	0.058	5.0
23	2165	0.064	201	23	2359	0.060	
24	2438	0.059	207	24	2334	0.060	
25	2379	0.066	4.2	25	2426	0.057	6.7
26	2401	0.061		26	2433	0.057	6.4
27	2400	0.060		27	1319	0.057	1.6
28	2398	0.064	207	28	1610	0.058	1.3
29	2394	0.057	4.2	29	1635	0.058	1.0
30	2437	0.057	5.4	30	2430	0.060	
31	2322	0.057	3.5				

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3. Excellent ultrasonic testing (UT) results from repeated examinations of flawed and unflawed welds, except in one (01) case on a Reactor Water Clean Up weld, confirm the effectiveness of HWC and of UT performed between 1983 and 1985. Note that the scanning sensitivity level of the CECo's UT procedure used in 1983 and 1984 examinations met the current EPRI recommended level for the detection of IGSCC.

With HWC credit, approximately fifty (50) percent of category G welds that were examined in 1983 and 1984, and all category G welds that have not been examined since 1983 will be examined in the upcoming refueling outage. Additionally, a weld selection/prioritization technique that takes into account factors such as fabrication history, prior UT history and results, system consideration, will be utilized to assure examination of welds with high risk of having IGSCC during the upcoming outage. For these reasons and because all category G welds that were examined in 1983 and 1984 will be examined within the next two (02) refueling cycles, examination of an additional sample of category G welds that were examined in 1983 and 1984 will not be necessary if flaws are found.

Table-2 gives the known weld population that is within the scope of the augmented inspection program along with the number of susceptible welds to be inspected at the next refueling outage.

Included in Table-2 are four (04) IGSCC susceptible welds that are not accessible for non-destructive examination (NDE).

1. Two (02) inaccessible welds are branch pipe connections. One is on the Isolation Condenser Condensate Return piping, the other is on the Reactor Water Clean up suction piping. Both welds should receive the full benefit of HWC. Each of these pressure retaining welds is completely encased within a reinforcement saddle which precludes any type of NDE. The assurance of continuity of the joints fabricated in this fashion is afforded by the fact that the reinforcement saddles strengthen the joints and reduce the stresses on the internal welds. These joints will be visually examined for evidence of leakage during the pressure tests required by IWB-5000 of ASME Section XI.

2. Two (02) inaccessible welds are located inside the containment penetration assemblies. One is on the Isolation Condenser Condensate Return piping, the other is on the Isolation Condenser Supply piping. Plans to install acoustic monitoring device at these locations are under investigation.

III. Item 3 Response: The Inservice Inspection Program for piping covered by the scope of Generic Letter 88-01 at Dresden Station will be in conformance with the staff positions on schedule, methods, and personnel, and sample expansion except as noted for Dresden Unit 2 in the response to Item 2. Since the Station is currently reviewing and revising the Technical Specifications under the Dresden Station Improvement Program - Technical Specifications Action Plan, this statement will be included in the Dresden ISI program once NRC review of Dresden's request for relief from the staff position on inspection schedule is completed.

IV. Item 4 Response: Dresden station will revise the Technical Specifications related to leakage detection to be in conformance with the staff position on leak detection included in Generic Letter 88-01 with the following exceptions:

1. Individual identified leakage is not flow-metered, but all identified leakage is collected and conducted to a separate collection system from unidentified leakage. Total identified leakage is monitored via the drywell equipment drain sump pump discharge flow totalizer meter.

   Strict compliance with this provision would require unnecessary plant modifications, therefore, the station does not plan to pursue an amendment in this area.

2. Sump operability is defined by the station as the ability to measure reactor coolant leakage rather than strictly depending on the operability of a leakage measurement instrument.

   Since there is only one channel for the unidentified leakage monitoring system, strict compliance with the staff position would result in unnecessary restrictive plant operating conditions.

3. The increase in unidentified leakage shall be 2 gpm over the previous 24 hour average. The 24 hour average will preclude unit shutdown due to variations in measured reactor coolant leakage between 4 hour intervals.

V. Item 5 Response: The Nuclear Regulatory Commission will be notified of the following conditions identified during the course of examination in accordance with Generic Letter 88-01:

1. Flaw indications exceeding the acceptance criteria of applicable Section XI, Subsection IWB-3500.

2. Change found in the condition of the welds previously known to have flaw indications.

3. The evaluation by the CECo Engineering Department for the above conditions for continued operation and/or the necessary corrective action to be taken.

Notification will be made by the CECo Nuclear Licensing Department to the appropriate NRR project manager.

TABLE-2

DRESDEN UNIT - 2 (W/ H_2 Chem. Credit)	SIZE	TOTAL	CATEGORIES A	B	C	D	E	F	G Insp. '83, '84	G Pre 1983	REMARKS
RECIRCULATION											
Outlets	28"	31	2	0	0	9^2	0	2^2	7^2	11^2	
Noz - SE	28"	2	0	0	0	1^2	0	0	1^2	0	2 furnace sensitized SE's
Header	22"	20	8	0	0	7^2	0	0	2^2	3^2	
Risers	12"	40	0	0	0	19^2	7	2^2	12^2	0	
SE's Noz - SE	12"	10	0	0	0	2^2	0	0	8^2	0	10 furnace sensitized
Bypass	4"	28	9	0	0	4^2	0	0	12^2	3^2	
RHR - LPCI	16"	25	0	0	0	11^1	0	0	$1^1 + 6^2$	4^1+3^2	
- SDC	16"	8	2	0	0	5^2	0	0	1^2	0	
ISCO -											
Supply	14"	26	0	0	0	11^1	0	0	5^1	9^1+1	1 inaccessible weld
	12"	13	0	0	0	2^1	0	0	0	11^1	
Return	12	14	0	0	0	7^1+3^2	0	0	1^1+1^2	2	2 inaccessible welds
CORE SPRAY	10"	6	6	0	0	0	0	0	0	0	
Jet Pump Inst.	12",8",4"	10	0	0	0	5^2	0	0	5^2	0	2 furnace sensitized SE's
RWCU	8"	28	17	0	0	7^2	2	0	1^2	1	1 inaccessible weld
N-18 A, B Nozzles	6"	4	4	0	0	0	0	0	0	0	
Head Vent	4"	3	0	0	0	1^1	0	0	2^1	0	
CRD	4"	6	0	0	0	4^2	0	0	2^2	0	
TOTAL		274	48	0	0	98	9	4	67	48	Total welds to be inspected during the next refueling outage:
			4.16%			25%	25%	100%	50%	100%	
Inspection next refueling outage			2			25	3^3	4	34 +	44	112

Note: 1 Welds w/o HWC credit (partial or no HWC benefit)
2 Welds w/ HWC credit (full HWC benefit)
3 Three weld overlays (not inspected in 1986) to be inspected in 1988 per SER.

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B. DRESDEN UNIT 3

I. Item 1 Response: An extensive IGSCC mitigation program has been implemented at Dresden Unit 3. A major piping replacement project was completed in 1986 and included the following methods of addressing IGSCC in susceptible piping:

-- Portions of sensitized piping were replaced with type 316 NG material. The number of field welds was reduced. Shop welds were solution heat treated and energy input was controlled on field welds.

-- Some susceptible lines were permanently eliminated.

-- Welds on non-replaced sections of susceptible piping were Stress Improved. The Mechanical Stress Improvement Process was utilized.

Replacement was performed on the following piping:

- Recirculation System -- all piping 2-1/2" and larger including the reactor vessel nozzle safe ends and the jet pump instrumentation safe ends and flanges.

- Low Pressure Coolant Injection System -- A and B loops from the Recirculation System connection to the outboard isolation valve.

- Core Spray System -- A and B loops from the reactor vessel nozzle to the inboard check valve including both nozzle safe ends.

- Shut Down Cooling System -- A and B loops of supply piping from the Recirculation suction connection to the inboard isolation valve; A and B loops return piping from the LPCI connection outside the drywell to the outboard isolation valve.

- Isolation Condenser System -- Condensate Return piping from the Shut Down Cooling connection to the outboard isolation valves.

The following piping was permanently eliminated:

- Control Rod Drive Return Line -- from the reactor vessel nozzle to the outboard check valve. A replacement safe end and cap were installed on the reactor nozzle.

- The 22-inch Recirculation discharge cross-tie.

- The 4-inch Recirculation discharge valve bypass lines.

The following piping was Stress Improved:

- A and B loops of the Core Spray system from the inboard check valve to the drywell penetration.

- Isolation Condenser steam supply line from the reactor vessel nozzle safe end to the Isolation Condenser inlet nozzles.

Hydrogen Water Chemistry is under consideration for this unit.

II. Item 2 Response: Beginning with the next refueling outage for Dresden Unit-3 (December 1989), ASME Code class 1, 2, and 3 piping made of stainless steel that is four (04) inches or larger in nominal diameter and contains reactor coolant at a temperature above 200° F during power operation will be subjected to an augmented inspection program. This augmented inspection program will conform to the NRC staff positions on inspection schedules, methods and personnel, and sample expansion delineated in Generic Letter 88-01.

Table-3 gives the known weld population that is within the scope of the augmented inspection program along with the number of susceptible welds to be inspected at the next refueling outage.

Included in Table-3 is one (01) IGSCC susceptible weld that is not accessible for non-destructive examination. This inaccessible weld is located inside the containment penetration assembly on the Isolation Condenser Supply piping. A plan to install an acoustic monitoring device at this location is under investigation.

III. Item 3 Response: The Inservice Inspection Program for piping covered by the scope of Generic Letter 88-01 at Dresden Station will be in conformance with the staff positions on schedule, methods, and personnel, and sample expansion except as noted for Dresden Unit 2 in the response to Item 2. Since the Station is currently reviewing and revising the Technical Specifications under the Dresden Station Improvement Program - Technical Specification Action Plan, this statement will be included in the Dresden ISI program once NRC review of Dresden's request for relief from the staff position on inspection schedule is completed.

IV. Item 4 Response: Dresden Station will revise the Technical Specifications related to leakage detection to be in conformance with the staff position on leak detection included in Generic Letter 88-01 with the following exceptions:

1. Individual identified leakage is not flow-metered, but all identified leakage is collected and conducted to a separate collection system from unidentified leakage. Total identified leakage is monitored via the drywell equipment drain sump pump discharge flow totalizer meter.

   Strict compliance with this provision would require unnecessary plant modifications, therefore, the station does not plan to pursue and amendment in this area.

2. Sump operability is defined by the Station as the ability to measure reactor coolant leakage rather than stability depending on the operability of a leakage measurement instrument.

   Since there is only one channel for the unidentified leakage monitoring system, strict compliance with the staff position would result in unnecessarily restrictive plant operating conditions.

3. The increase in unidentified leakage shall be 2 gpm over the previous 24 hour average. The 24 hour average will preclude unit shutdown due to variations in measured reactor coolant leakage between 4 hour intervals.

V. <u>Item 5 Response:</u> The Nuclear Regulatory Commission will be notified of the following conditions identified during the course of examination in accordance with Generic Letter 88-01.

1. Flaw indications exceeding the acceptance criteria of applicable Section XI, Subsection IWB-3500.

2. Change found in the condition of the welds previously known to have flaw indications.

3. The evaluation by the CECo Engineering Department for the above conditions for continued operation and/or the necessary corrective action to be taken.

Notification will be made by the CECo Nuclear Licensing Department to the appropriate NRR project manager.

TABLE-3

DRESDEN UNIT - 3	SIZE	TOTAL	CATEGORIES A	B	C	D	E	F	G	REMARKS
RECIRCULATION										
Outlets	28"	22	22	0	0	0	0	0	0	
Noz - SE	28"	2	2	0	0	0	0	0	0	
Header	22"	4	4	0	0	0	0	0	0	
Risers	12"	24	24	0	0	0	0	0	0	
Noz - SE	12"	10	10	0	0	0	0	0	0	
Bypass	4"	0	0	0	0	0	0	0	0	
RHR - LPCI	16"	20	20	0	0	0	0	0	0	
- SDC	16"	7	7	0	0	0	0	0	0	
	14"	3	3	0	0	0	0	0	0	
ISOLATION CONDENSER										
Supply	14"	24	0	0	23	0	0	0	1	1 inaccessible weld
	12"	15	0	0	15	0	0	0	0	
Return	12	11	11	0	0	0	0	0	0	
CORE SPRAY	10"	28	12	0	16	0	0	0	0	
Jet Pump Inst.	4"	2	2	0	0	0	0	0	0	
RWCU	8"	16	16	0	0	0	0	0	0	
	6"	1	1	0	0	0	0	0	0	
N-18 A, B Nozzles	6"	4	0	0	0	4	0	0	0	
Head Vent	4"	3	0	0	0	3	0	0	0	
TOTAL		196	134	0	54	7	0	0	1	Total welds to be inspected during the next refueling outage: 36
			4.16%		x50%	x50%				
Inspection next refueling outage			6		27	3				

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C. LA SALLE UNIT - 1

1. Item 1 Response: Commonwealth Edison Co. (CECo) is currently pursuing an Integrated Program to mitigate the occurrence of IGSCC and its effects at this unit. During the initial design and construction, changes were made to reduce the amount of IGSCC susceptible stainless steel piping and to revise construction practices in order to make this material less sensitive to IGSCC. The program also involves the use of three (03) primary mitigators:

   1. Hydrogen Water Chemistry is being evaluated as a long term mitigation scheme to provide additional protection for IGSCC susceptible welds.

   2. Stress Improvement has been performed on all unit 1's susceptible piping welds beginning with the first refueling outage and completed during the second refueling outage. This includes the Safe-End-to-Nozzle and Inconel 182 Buttered Safe End welds. Both Induction Heating Stress Improvement and Mechanical Stress Improvement processes have been used in treating susceptible welds in this unit.

   3. Weld Overlay has been reviewed for application to welds that have flaw indications in excess of the limits permitted by NUREG-0313, revision 2. To date, no flaw indications of a magnitude requiring an overlay reinforcement have been found at LaSalle Unit 1.

II. Item 2 Response: Beginning with the next refueling outage for LaSalle Unit 1 (December 1989), ASME Code class 1, 2 and 3 piping made of stainless steel that is four (04) inches or larger in nominal diameter and contains reactor coolant at a temperature above 200° F during power operation will be subjected to an augmented inspection program. This augmented inspection program will conform to the NRC staff positions on inspection schedules, methods and personnel, and sample expansion delineated in Generic Letter 88-01.

Table-4 gives the known welds population that is within the scope of the augmented inspection program along with the number of susceptible welds to be inspected at the next refueling outage. Note that the number of welds to be inspected is not evenly distributed to each outage.

III. Item 3 Responses: It is the intention of LaSalle Station to implement an augmented inspection program which conforms to the NRC staff positions on inspection schedules, methods, personnel qualifications, and sample expansion as delineated in NRC Generic Letter 88-01. However, it is the position of the station that a technical specification amendment to indicate compliance to a generic letter is neither necessary nor appropriate for the following reasons:

- The amendment would unnecessarily clutter the technical specifications with information not appropriate for immediate operator reference.

- The amendment does not meet the screening criteria for determining which regulatory requirements and operating restrictions should be retained in the standard technical specifications and ultimately in the plant technical specifications, as given in the Interim Policy statement on Technical Specification Improvements, 52FR3788, February 6, 1987.

- As the industry and the NRC gain more insight into the causes of, and methods for prevention IGSCC, the requirements in this area will be changing. Therefore, addition of a reference to the generic letter in the technical specifications would require that the technical specifications be updated as the requirements in this area evolve.

It is the opinion of the station that it would be more appropriate to pursue or change to the station's Inservice Inspection program to include a statement indicating conformance to Generic Letter 88-01.

IV. Item 4 Response:

a. Staff Position - "Leakage detection systems should be in conformance with Position C of Regulatory Guide 1.45 . . ."

Response - Section 5.25 of the LaSalle county SER outlined the leak detection systems. The conclusion reached was that the leakage detection system provides reasonable assurance for detecting small leaks across the reactor coolant pressure boundary as required by Criterion 30 of GDC and Reg. Guide 1.45 and are acceptable.

b. Staff Position - "Plant shutdown should be initiated for inspection and corrective action when, within any period of 24 hours or less, any leakage detection system indicates an increase in rate of unidentified leakage in excess of 2 gpm or its equivalent, . . ."

Response - The sump fill-up rate is monitored continuously by chart recorder, and an alarm sounds each time the sump pump starts. The sump pump discharge flowmeter totalizer is recorded shiftly, and the leakage rate is determined. LaSalle Unit 1 has only 2 Category E welds which have been stable over the 2nd cycle. It is the station's opinion that addition of the "2 gpm" requirement to the technical specifications would not provide a significant increase in the level of plant safety.

c. Staff Position - ". . . or when the total unidentified leakage attains a rate of 5 gpm or equivalent, . . ."

Response - The LaSalle Station technical specifications are currently in conformance with this recommendation.

d. Staff Position - ". . . For sump level monitoring systems with fixed-measurement-interval methods, the level should be monitored at approximately 4 hour intervals or less."

Response - LaSalle Station does not use the fixed-measurement-interval method.

e. The technical specification definition for identified/unidentified leakage is in compliance with the definition suggested in the generic letter, except where in paragraph 2a of the staff position, it states that identified leakage which is captured is to be flow metered and conducted to a collection tank. The LaSalle technical specifications do not require this leakage to be flow metered. Strict compliance with this

provision might require plant modifications to allow metering of all individual identified leakage. Total identified leakage is monitored via the drywell equipment drain sump pump discharge flow totalizer meter. Readings from this totalizer are taken shiftly and the total identified leakage is determined from these readings. The sump fill-up rate is monitored continuously on a control room chart recorder. Since the LaSalle technical specifications are currently in compliance with Regulatory Guide 1.45, May 1973, and the standard technical specifications, the station does not intend to pursue an amendment in this area.

f. Staff Position - "For plants operating with any IGSCC D, E, F or G welds, at least one of the leakage measurement instruments associated with each sump shall be operable, and the outage time for inoperable instruments shall be limited to 24 hours, or immediately initiate an orderly shutdown."

Response - Three separate systems for reactor coolant leakage detection are required by the technical specifications. Continued operation for thirty days is allowed with one of the systems inoperable. Since there is only one channel for sump flow monitoring, the staff position would require an immediate shutdown of the unit if the channel were to become inoperable. This situation could be unnecessarily restrictive to plant operations. Since, the station is currently in compliance with the GE Standard Technical Specifications and Regulatory Guide 1.45, an amendment to the LaSalle Technical Specifications in this area is not deemed to be necessary. As indicated above, LaSalle Unit 1 has only 2 Category E welds which have been stable over the 2nd cycle.

V. Item 5 Response: The Nuclear Regulatory Commission will be notified of the following conditions identified during the course of examination in accordance with Generic Letter 88-01:

1. Flaw indications exceeding the acceptance criteria of applicable Section XI, Subsection IWB-3500.

2. Change found in the condition of the welds previously known to have flaw indicators.

3. The evaluation by the CECo Engineering Department for the above conditions for continued operation and/or the necessary corrective action to be taken.

Notification will be made by the CECo Nuclear Licensing Department to the appropriate NRR project manager.

TABLE-4

LASALLE UNIT - 1	SIZE	TOTAL	CATEGORIES A	B[1]	C	D	E	F	G	REMARKS
RECIRCULATION										
Outlets	24"	44	6	36	2	0	0	0	0	
Noz - SE	24"	2	0	2	0	0	0	0	0	
Header	16"	16	8	8	0	0	0	0	0	
Risers	12"	50	20	28	0	0	2	0	0	
Noz - SE	12"	10	0	10	0	0	0	0	0	
Decon & Attach.	4"	12	6	6	0	0	0	0	0	
Shutdown Cooling	20"	11	0	11	0	0	0	0	0	
SDC, LPCI, LPCS, HPCS	12"	27	2	14	11	0	0	0	0	
RCIC, Spare Noz.	6"	3	0	0	3	0	0	0	0	
Rx Clean Up, JPI, Head Vent, CRD Cap.	4"	20	0	9	11	0	0	0	0	
TOTAL		195	42 4.16%	124 8.33%	27 50%	0	2 50%	0	0	Total welds to be inspected during the next refueling outage: 22
Inspection next refueling outage			0	8	13		1			

Note: 1. Stress Improvement applied within two years of commercial operating date.

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D. LASALLE UNIT-2

I. Item 1 Response: Commonwealth Edison Co. (CECo) is currently pursuing an Integrated Program to mitigate the occurrence of IGSCC and its effects at this unit. During the initial design and construction, changes were made to reduce the amount of IGSCC susceptible stainless steel piping and to revise construction practices in order to make this material less sensitive to IGSCC. The program also involves the use of three (03) primary mitigators:

1. Hydrogen Water Chemistry is being evaluated as a long term mitigation scheme to provide additional protection for IGSCC susceptible welds.

2. Stress Improvement has been applied at LaSalle Unit 2. The majority of the susceptible welds received stress improvement prior to commercial operating date. Some welds received stress improvement within two (02) years of commercial operating date. Eight remaining welds will receive stress improvement treatment during the second refueling outage starting in October, 1988. At that time all susceptible piping, Safe End-to-Nozzle, and Inconel 182 Buttered welds will have been treated. Both Induction Heating Stress Improvement and Mechanical stress Improvement processes have been used in treating susceptible welds in this unit.

3. Weld Overlay has been reviewed for application to welds that have flaw indications in excess of the limits permitted by NUREG-0313, revision 2. To date, no weld with flaw indications has been found at LaSalle Unit 2.

II. Item 2 Response: Beginning with the next refueling outage for LaSalle Unit 2 (October 1988), all ASME Code class 1, 2, and 3 piping made of stainless steel that is four (04) inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation will be subjected to an augmented inspection program. This augmented inspection program will conform to the NRC staff positions on inspection schedules, methods and personnel, and sample expansion delineated in Generic Letter 88-01.

Table-5 gives the known weld population that is within the scope of the augmented inspection program along with the number of susceptible welds to be inspected at the next refueling outage. Note that the number of welds inspected is not evenly distributed to each outage.

III. Item 3 Response: It is the intention of LaSalle Station to implement an inservice inspection program which conforms to the NRC staff positions on inspection schedules, methods, personnel qualifications and sample expansion as delineated in NRC Generic Letter 88-01. However, it is the position of the station that a technical specification amendment to indicate compliance to a generic letter is neither necessary nor appropriate for the following reasons:

- The amendment would unnecessarily clutter the technical specifications with information not appropriate for immediate operator reference.

- The amendment does not meet the screening criteria for determining which regulatory requirements and operating restrictions should be retained in the standard technical specifications and ultimately in the plant technical specifications, as given in the Interim Policy Statement on Technical Specification Improvements, 52FR3788, February 6, 1987.

- As the industry and the NRC gain more insight into the causes of , and methods for prevention of IGSCC, the requirements in this area will be changing. Therefore, addition of a reference to the generic letter in the technical specifications would require that the technical specifications be updated as the requirements in this area evolve.

It is the opinion of the station that it would be more appropriate to pursue a change to the station's Inservice Inspection program to include a statement indicating conformance to Generic Letter 88-01.

IV. Item 4 Response:

a. Staff Position - "Leakage detection systems should be in conformance with Position C of Regulatory Guide 1.45 . . . ."

   Response - Section 5.25 of the LaSalle County SER outlined the leak detection systems. The conclusion reached was that the leakage detection system provides reasonable assurance for detecting small leaks across the reactor coolant pressure boundary as required by Criterion 30 of GDC and Reg Guide 1.45 and are acceptable.

b. Staff Position - "Plant shutdown should be initiated for inspection and corrective action when, within any period of 24 hours or less, any leakage detection system indicates an increase in rate of unidentified leakage in excess of 2 gpm or its equivalent, . . ."

Response - The sump fill-up rate is monitored continuously by chart recorder, and an alarm sounds each time the sump pump starts. The sump pump discharge flowmeter totalizer is recorded shiftly, and the leakage rate is determined. LaSalle Unit 2 is expected to have no Category D, E, F or G welds after the 2nd refueling outage scheduled for the Fall of 1988. It is the station's opinion that addition of the "2 gpm" requirement to the technical specifications would not provide a significant increase in the level of plant safety.

c. Staff Position - ". . . or when the total unidentified leakage attains a rate of 5 gpm or equivalent, . . ."

Response - The LaSalle Station technical specifications are currently in conformance with this recommendation.

d. Staff Position - ". . . For sump level monitoring systems with fixed-measurement-interval methods, the level should be monitored at approximately 4 hour intervals or less."

Response - LaSalle Station does not use the fixed-measurement-interval method.

e. The technical specification definition for identified/unidentified leakage is in compliance with the definition suggested in the generic letter, except where in paragraph 2a of the staff position, it states that identified leakage which is captured is to be flow metered and conducted to a collection tank. The LaSalle technical specifications do not require this leakage to be flow metered. Strict compliance with this provision might require plant modifications to allow metering of all individual identified leakage paths. All identified leakage is collected (captured) and conducted to a separate collection system from unidentified leakage. Total identified leakage is monitored via the drywell equipment drain sump pump discharge flow totalizer meter. Readings from his totalizer are taken shiftly and the total identified leakage is determined from these readings. The sump fill-up rate is monitored continuously on a control room chart recorder. Since the LaSalle technical specifications are currently in compliance with Regulatory Guide 1.45, May 1973, and the standard technical specifications, the station does not intend to pursue an amendment in this area.

f. Staff Position - "For plants operating with any IGSCC D, E, F, or G welds, at least one of the leakage measurement instruments associated with each sump shall be operable, and the outage time for inoperable instruments shall be limited to 24 hours, or immediately initiate an orderly shutdown."

Response - Three separate systems for reactor coolant leakage detection are required by the technical specifications. Continued operation for thirty days is allowed with one of the systems inoperable. Since there is only one channel for sump flow monitoring, the staff position would require an immediate shutdown of the unit is the channel were to become inoperable. This situation could be unnecessarily restrictive to plant operations. Since, the station is currently in compliance with the GE Standard technical Specifications and Regulatory Guide 1.45, an amendment to the LaSalle Technical Specifications in this area is not deemed to be necessary. As indicated above, LaSalle Unit 2 is expected to have no Category D, E, F, or G welds after the 2nd refueling outage scheduled for the Fall of 1988.

V. Item 5 Response: The Nuclear Regulatory Commission will be notified of the following conditions identified during the course of examination in accordance with Generic Letter 88-01:

1. Flaw indications exceeding the acceptance criteria of applicable Section XI, Subsection IWB-3500.

2. Change found in the condition of the welds previously known to have flaw indications.

3. The evaluation by the CECo Engineering Department for the above conditions for continued operation and/or the necessary corrective action to be taken.

Notification will be made by the CECo Nuclear Licensing Department to the appropriate NRR project manager.

TABLE-5

LASALLE UNIT - 2	SIZE	TOTAL	CATEGORIES A	B[1]	C	D	E	F	G	REMARKS
RECIRCULATION										
Outlets	24"	45	4	40	0	1	0	0	0	
Noz - SE	24"	2	0	2	0	0	0	0	0	
Header	16"	16	8	8	0	0	0	0	0	
Risers	12"	50	30	20	0	0	0	0	0	
Noz - SE	12"	10	0	10	0	0	0	0	0	
Decon	4"	12	6	4	0	0	0	0	2	
Shutdown Cooling	20"	11	0	10	0	0	0	0	1	
SDC, LPCI, LPCS, HPCS, FW	12"	26	2	23	0	0	0	0	1	
RCIC, Spare Noz	6"	3	0	2	0	1	0	0	0	
Rx Clean Up, JPI, Head Vent, CRD Cap.	4"	20	0	18	0	0	0	0	2	
Total		195	50 4.16%	137 8.33%	0	2 50%	0	0	6 100%	Total welds to be inspected during the next refueling outage: 26
Inspection next refueling outage			0	18		2			6	

Note: 1. Stress Improvement applied prior to and within two years of commercial operating date.

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E. QUAD CITIES UNIT-1

I. Item 1 Response: Commonwealth Edison Co. (CECo) is presently pursuing an Integrated Program to mitigate the occurrence of IGSCC and its effects at this unit. The program involves the use of four (04) primary mitigators:

1. Hydrogen Water Chemistry system is being installed at this unit. The system is scheduled to be operational at the end of 1988.

2. Stress Improvement has been applied to most of the weldments with high IGSCC risk. Two stress improvements processes, Induction Heating Stress Improvement and Mechanical Stress Improvement Process, have been utilized.

3. Weld Overlay has been and is being used to reinforce welds that have flaw indications in excess of the ASME Section XI, Subsection IWB-3500 limits. Most of the weld overlay reinforcements have been applied and inspected in accordance with NUREG-0313, revision 2.

4. System Removal is being investigated to reduce the population of IGSCC susceptible welds in this unit. Potential candidates are the Residual Heat Removal/Head Spray and the CRD return lines.

II. Item 2 Response: Beginning with the next refueling outage for Quad Cities unit 1 (June 1989), ASME Code class 1, 2, and 3 piping made of stainless steel that is four (04) inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation will be subjected to an augmented inspection program. This augmented inspection program will conform to the NRC staff positions on inspection schedules, methods and personnel, and sample expansion delineated in Generic Letter 88-01.

Table-6 gives the known weld population that is within the scope of the augmented inspection program along with the number of susceptible welds to be inspected at the next refueling outage.

Included in Table-6 are four (04) IGSCC susceptible welds that are not accessible for non-destructive examination (NDE):

1. One (01) inaccessible weld is located inside the containment penetration assembly on the Reactor Water Clean-Up suction line. A plan to replace this section of piping with IGSCC resistant material is under consideration.

2. One (01) inaccessible weld is a branch pipe connection on the Reactor Water Clean Up suction line. This pressure retaining weld is completely encased within a reinforcement saddle which precludes any type of non-destructive examination. The assurance of the continued integrity of the joint fabricated in this fashion is afforded by the fact that the reinforcement saddle strengthens the joint and reduces the stresses on the internal weld. Additionally, the reinforcement saddle has a "weep hole" which facilitates visual examination for evidence of leakage conducted during the pressure tests required by IWB-5000 of ASME Section XI.

3. One (01) inaccessible weld is on the Residual Heat Removal/Head Spray line. The weld is located at the floor of the refueling cavity within a water barrier and sleeve arrangement which precludes any type of non-destructive examination. A plan to completely remove this Residual Heat Removal/Head Spray section of piping is under investigation.

4. One (01) inaccessible weld is located inside the containment penetration assembly on the Residual Heat Removal/Shutdown Cooling line. A plan to install acoustic monitoring device at this location is under investigation.

III. Item 3 Response: It is the intention of Quad Cities Station to implement an augmented inspection program which follows the NRC staff positions on inspection schedules, methods, personnel qualifications and sample expansion with exceptions as delineated in the preceding response to item 2. However, it is the position of the station that a technical specification amendment to indicate compliance to a generic letter is neither necessary nor appropriate for the following reasons:

1. The amendment would unnecessarily clutter the technical specifications with information not appropriate for immediate operator reference.

2. The amendment does not meet the screening criteria for determining which regulatory requirements and operating restrictions should be retained in the standard technical specifications and ultimately in the plant technical specifications, as given in the Interim Policy Statement on Technical Specification Improvements, 52FR3788, February 6, 1987.

3. As the industry and the NRC gain more insight into the courses of, and methods for prevention of IGSCC, the requirements in this area will be changing. Therefore, addition of a reference to the generic letter in the technical specifications would require that the technical specifications be updated as the requirements in this area evolved.

It is the opinion of the station that it would be more appropriate to pursue a change to the station's Inservice Inspection program to include a statement indicating conformance to Generic Letter 88-01 with exceptions as delineated in this response.

IV. Item 4 Response: Quad Cities Station will revise the Technical Specifications related to leakage detection to be in conformance with the staff position on leak detection included in Generic Letter 88-01 with the following exceptions:

1. Individual identified leakage is not flow-metered, but all identified leakage is collected and conducted to a separate collection system from unidentified leakage. Total identified leakage is monitored via the drywell equipment drain sump pump discharge flow totalizer meter.

   Strict compliance with this provision would require unnecessary plant modifications, therefore, the station does not plan to pursue an amendment in this area.

2. Sump operability is defined by the station as the ability to measure reactor coolant leakage rather than strictly depending on the operability of a leakage measurement instrument.

   Since there is only one channel for the unidentified leakage monitoring system, strict compliance with the staff position would result in unnecessarily restrictive plant operating conditions.

3. The increase in unidentified leakage shall be 2 gpm over the previous 24 hour average. The 24 hour average will preclude unit shutdown due to variations in measured reactor coolant leakage between 4 hour intervals.

V. Item 5 Response: The Nuclear Regulatory Commission will be notified of the following conditions identified during the course of examination in accordance with Generic Letter 88-01:

1. Flaw indications exceeding the acceptance criteria of applicable ASME Section XI, Subsection IWB-3500.

2. Change found in the condition of the welds previously known to have flaw indications.

3. The evaluation by the CECo Engineering Department for the above conditions for continued operation and/or the necessary corrective action to be taken.

Notification will be made by the CECo Nuclear Licensing Department to the appropriate NRR project manager.

TABLE-6

QUAD CITIES UNIT - 1	SIZE	TOTAL	CATEGORIES A	B	C	D	E	F	G	REMARKS
RECIRCULATION										
Outlets	28"	32	2	0	28	0	2	0	0	
Noz - SE	28"	2	2	0	0	0	0	0	0	
Header	22"	20	8	0	7	2	3	0	0	
Risers	12"	41	0	0	22	0	19	0	0	
Noz - SE	12"	10	10	0	0	0	0	0	0	
Bypass	4"	12	4	0	0	2	0	0	6	
RHR - SDC	20"	17	0	0	4	9	0	0	4	1 inaccessible weld
- LPCI	16"	26	0	0	25	0	1	0	0	
CORE SPRAY	10"	27	1	0	20	0	6	0	0	
Jet Pump Inst.	12",8",4"	10	0	0	0	3	0	0	7	
RWCU, Hd.Sp, Spare Noz.	6"	37	27	0	0	8	0	0	2	2 inaccessible welds on RWCU
Hd.Sp, Hd. Vt, CRD	4"	19	3	0	0	13	0	0	3	1 inaccessible weld on Hd. Sp.
TOTAL		253	57	0	106	37	31	0	22	Total welds to be inspected during the next refueling outage: 108
			4.16%		50%	50%	50%		100%	
Inspection next refueling outage			2		53	19	16		18	

3016C

F. QUAD CITIES UNIT - 2

I. Item 1 Response: Commonwealth Edison Co. (CECo.) is presently pursuing an Integrated Program to mitigate the occurrence of IGSCC and its effects at this unit. The program involves the use of four (04) primary mitigators:

1. Hydrogen Water Chemistry system is being installed at this unit. The system is scheduled to be operational at the end of 1988.

2. Stress Improvement has been applied to most of the weldments with high IGSCC risk. Two (02) stress improvements process, Induction Heating Stress Improvement and Mechanical Stress Improvement Process, have been utilized.

3. Weld Overlay has been and is being used to reinforce welds that have flaw indications in excess of the ASME Section XI, Subsection IWB-3500 limits. Most of the weld overlay reinforcements have been applied and inspected in accordance with NUREG-0313, revision 2.

4. System Removal is being investigated to reduce the population of IGSCC susceptible welds in this unit. Potential candidates are the Residual Heat Removal/Head Spray and the CRD Return lines.

II. Item 2 Response: Beginning with the next refueling outage for Quad Cities Unit 2 (October 1989), ASME Code class 1, 2, and 3 piping made of stainless steel that is four (04) inches or larger in nominal diameter and contains reactor coolant at a temperature above 200° F during power operation will be subjected to an augmented inspection program. This augmented inspection program will conform to the NRC staff positions on inspection schedules, methods and personnel, and sample expansion delineated in Generic Letter 88-01.

Table-7 gives the known weld population that is within the scope of the augmented inspection program along with the number of susceptible welds to be inspected at the next refueling outage.

Included in Table-7 are five (05) IGSCC susceptible welds that are not accessible for non-destructive examination (NDE):

1. Two inaccessible welds are located inside the containment penetration assembly on the Reactor Water Clean Up suction line. A plan to replace this section of piping with IGSCC resistant material is under consideration.

2. One inaccessible weld is a branch pipe connection on the Reactor Water Clean Up suction line. This pressure retaining weld is completely encased within a reinforcement saddle which precludes any type of non-destructive examination. The assurance of the continued integrity of the joint fabricated in this fashion is afforded by the fact that the reinforcement saddle strengthens the joint and reduces the stresses on the internal weld. Additionally, the reinforcement saddle has a "weep hole" which facilitates visual examination for evidence of leakage conducted during the pressure tests required by IWB-5000 of ASME Section XI.

3. One inaccessible weld is on the Residual Heat Removal/Head Spray line. The weld is located at the floor of the refueling cavity within a water barrier and sleeve arrangement which precludes any type of non-destructive examination. A plan to completely remove this Residual Heat Removal/Head to Spray section of piping is under investigation.

4. One inaccessible weld is located inside the containment penetration assembly on the Residual Heat Removal/Shutdown Cooling line. A plan to install acoustic monitoring device at this location is under investigation.

III. Item 3 Response: It is the intention of Quad Cities Station to implement an augmented inspection program which follows the NRC staff positions on inspection schedules, methods, personnel qualifications, and sample expansion with exceptions as delineated in the preceding response to item 2. However, it is the position of the station that a technical specification amendment to indicate compliance to a generic letter is neither necessary nor appropriate for the following reasons:

1. The amendment would unnecessarily clutter the technical specifications with information not appropriate for immediate operator reference.

2. The amendment does not meet the screening criteria for determining which regulatory requirements and operating restrictions should be retained in the standard technical specifications and ultimately in the plant technical specifications, as given in the Interim Policy Statement on Technical Specification Improvements, 52 FR 3788, February 6, 1987.

3. As the industry and the NRC gain more insight into the causes of, and methods for prevention of IGSCC, the requirements in this area will be changing. Therefore, addition of a reference to the generic letter in the technical specifications would require that the technical specifications be updated as the requirements in this area evolved.

It is the opinion of the station that it would be more appropriate to pursue a change to the station's Inservice Inspection program to include a statement indicating conformance to Generic Letter 88-01 with exceptions as delineated in this response.

IV. Item 4 Response: Quad Cities Station will revise the Technical Specifications related to leakage detection to be in conformance with the staff position on leak detection included in Generic Letter 88-01 with the following exceptions:

1. Individual identified leakage is not flow-metered, but all identified leakage is collected and conducted to a separate collection system from unidentified leakage. Total identified leakage is monitored via the drywell equipment drain sump pump discharge flow totalizer meter.

   Strict compliance with this provision would require unnecessary plant modifications, therefore, the station does not plan to pursue and amendment in this area.

2. Sump operability is defined by the Station as the ability to measure reactor coolant leakage rather than strictly depending on the operability of a leakage measurement instrument.

   Since there is only one channel for the unidentified leakage monitoring system, strict compliance with the staff position would result in unnecessarily restrictive plant operating conditions.

3. The increase in unidentified leakage shall be 2 gpm over the previous 24 hour average. The 24 hour average will preclude unit shutdown due to variations in measured reactor coolant leakage between 4 hour intervals.

V. Item 5 Response: The Nuclear Regulatory Commission will be notified of the following conditions identified during the course of examination in accordance with Generic Letter 88-01.

1. Flaw indications exceeding the acceptance criteria of applicable ASME Section XI, Subsection IWB-3500.

2. Change found in the condition of the welds previously known to have flaw indications.

3. The evaluation by the CECo Engineering Department for the above conditions for continued operation and/or the necessary corrective action to be taken.

Notification will be made by the CECo Nuclear Licensing Department to the appropriate NRR project manager.

TABLE-7

QUAD CITIES UNIT - 2	SIZE	TOTAL	CATEGORIES A	B	C	D	E	F	G	REMARKS
RECIRCULATION										
Outlets	28"	32	2	0	19	0	9	2	0	
Noz - SE	28"	2	2	0	0	0	0	0	0	
Header	22"	22	8	0	10	2	2	0	0	
Risers	12"	44	0	0	23	0	21	0	0	
Noz - SE	12"	10	10	0	0	0	0	0	0	
Bypass	4"	12	4	0	0	3	0	0	5	
RHR - SDC	20"	18	0	0	2	5	2	0	9	1 inaccessible weld
- LPCI	16"	26	0	0	26	0	0	0	0	
CORE SPRAY	10"	25	4	0	21	0	0	0	0	
Jet Pump Inst.	12",8",4"	10	0	0	0	4	0	0	6	
RWCU, Hd.Sp, Spare Noz.	6"	45	33	0	0	3	2	0	7	3 inaccessible welds on RWCU
Hd.Sp, Hd. Vt, CRD	4"	27	3	0	0	10	0	0	14	1 inaccessible weld on Hd. Sp.
TOTAL		273	66 4.16%	0	101 50%	27 50%	36 50%	2 100%	41 100%	Total welds to be inspected during the next refueling outage: 123
Inspection next refueling outage			3		50	14	18	2	36	

3016C

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

July 28, 1988

Mr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 and 3
Withdrawal of Relief Request No. VR-16
of the IST Program - Revision 1
NRC Docket Nos. 50-237 and 50-249

References (a): Letter from D.R. Muller to L.D. Butterfield dated September 16, 1987 transmitting Interim SER for the Dresden IST Program (Revision 1).

(b): Letter from J.A. Silady to T.E. Murley dated November 17, 1987 responding to open items of the Interim SER.

(c): Letter from J.A. Silady to T.E. Murley dated May 6, 1987 submitting Revision 2 of the Dresden IST Program.

(d): Conference Call between CECo (J.A. Silady et al.), NRR (B. Siegel, et al.) and EG&G (H. Rockhold).

Dear Mr. Murley:

Based on the Reference (d) discussions with your staff, Commonwealth Edison (CECo) withdraws an earlier Relief Request (No. VR-16) concerning inservice testing of valve stroke times at Dresden. Relief Request VR-16 had been revised as part of the CECo response in Reference (b) to IST open items identified by NRR and EG&G in Reference (a).

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It was also agreed during the Reference (d) discussions that CECo would provide two sets of Dresden Piping and Instrument Diagrams (P&ID) for use by NRR and EG&G in reviewing the new Dresden IST Program which was submitted with Reference (c). By copy of this letter, one set of P&ID's are provided to B.L. Siegel of your staff and one set to H.C. Rockhold of EG&G.

Please contact this office should further information be required.

Very truly yours,

J A Silady

J. A. Silady
Nuclear Licensing Administrator

lm

cc: A.B. Davis - Regional Admin., RIII
B.L. Siegel - Project Manager, NRR
S.G. DuPont - Senior Resident Inspector, Dresden
H.C. Rockhold - EG&G

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