



October 16, 1991

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

Subject: LaSalle County Station Units 1 and 2
Supplement to Application for Amendment to Facility
Operating Licenses NPF-11 and NPF-18, Appendix A,
Technical Specifications
NRC Docket Nos. 50-373 and 50-374.

Reference (a): W.E. Morgan letter to USNRC dated October 10, 1990;
transmitting CECO's original request to remove
Technical Specification requirements for the HPCS CST
Suction Valve and adding containment isolation
requirements for the new RCIC line.

Gentlemen:

Reference (a) transmitted Commonwealth Edison's proposed Technical Specification Amendment to remove the requirements for the High Pressure Core Spray (HPCS) Condensate Storage Tank (CST) Suction Valve. In addition, Reference (a) proposed the addition of containment isolation requirements for the planned, new Reactor Core Isolation Cooling (RCIC) system full flow test line to the suppression pool. The NRC Staff has determined that the proposed valve configuration for RCIC does not satisfy the requirements of the General Design Criteria specified in 10CFR50, Appendix A nor the Standard Review Plan. As a result, footnote 'n' to Table 3.6.3-1 will be modified to show compliance to the General Design Criteria.

The plant modifications are scheduled to be installed during the next refueling outage for each unit (Unit 1, October 1992; Unit 2, January 1992). Approval of this amendment is requested by November 15, 1991 to support the upcoming Unit 2, January 1992 refueling outage. It is requested that the amendments be made effective upon startup of each unit following the L2R04 and L1R05 refueling outages.

Subsequent design modifications to the affected systems necessitates changes to the Bypass Device of Valves 1E22-F015 and 2E22-F015 from Accident Conditions to Continuous Conditions in Technical Specification Table 3.8.3.3-1. In addition, several typographical errors have been corrected for both Unit 1 and for Unit 2. The following attachments provide Commonwealth Edison's revised submittal:

1. Attachment A gives a description and safety analysis of the supplement to the original request.
2. Attachment B includes the marked-up Technical Specifications pages with the requested changes indicated. The inclusion of the marked up pages from the Reference (a) submittal are provided for continuity. Pages 3/4 3-26, 3/4 3-30, 3/4 3-30a, 3/4 3-32, 3/4 3-33, 3/4 5-6, 3/4 6-33, 3/4 6-34, 3/4 6-34a and 3/4 8-30 for Unit 1 and Pages 3/4 3-30, 3/4 3-30a, 3/4 5-6, 3/4 6-36, 3/4 6-37a and 3/4 8-30 for Unit 2 are the only pages affected by this supplemental request.

ADD
~~ADD~~

9111070227 911016
PDR ADUCK 05000373
PDR

60033

CECo's original evaluation performed in accordance with 10CFR50.92(c), that confirmed that no significant hazards considerations were involved remains valid and conservatively bounds this supplemental analysis; therefore, the significant hazards considerations do not require supplemental information. CECo's original Environmental Assessment also remains valid.

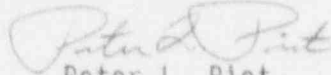
This supplement to Reference (a) has been reviewed and approved by CECo On-Site and Off-Site Review in accordance with Commonwealth Edison procedures.

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 obtained 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.

Commonwealth Edison is notifying the State of Illinois of our revised amendment request by transmitting a copy of this letter and its attachments to the designated state official.

Please direct any questions you may have concerning this supplement to this office.

Very truly yours,



Peter L. Piet

Nuclear Licensing Administrator

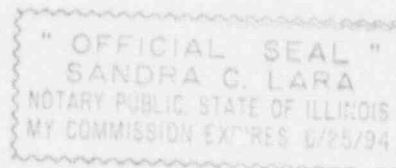
Attachments:

- A. Description of Safety Analysis of the Proposed Changes
- B. Marked-Up Technical Specification Pages

cc: A.B. Davis - Regional Administrator, RIII
 Senior Resident Inspector - LSCS
 B. L. Siegel - NRR, Project Manager
 Office of Nuclear Facility Safety - IDNS

PLP:

Signed before me on this 16th day
 of October, 1991
 by [Signature]
 Notary Public



ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES TO APPENDIX A, TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSES NPF-11 AND NPF-18

BACKGROUND

This supplemental amendment request is being submitted to amend CECO's original proposed Technical Specification amendment (Reference (a)). Due to design changes to the affected systems, the thermal overload bypass device for valve 1(2)E22-F015 of the HPCS system is being changed from Accident Conditions to Continuous Conditions. The design change affects page 3/4 8-30 for both Unit 1 and Unit 2. In addition, several typographical errors are being corrected (pages 3/4 3-26, 3/4 3-30, 3/4 3-30a, 3/4 3-32, 3/4 3-33, 3/4 5-6, 3/4 6-34, and 3/4 6-34a for Unit 1 and pages 3/4 3-30, 3/4 3-30a, 3/4 5-6, 3/4 6-36 and 3/4 6-37a for Unit 2).

In the time interval since the Reference (a) submittal, several telephone conversations have been held between representatives of CECO and the NRC Staff that discussed LaSalle's proposed modification to the RCIC system. The NRC Staff expressed concerns that the proposed valve configuration did not satisfy the requirements of the General Design Criteria specified in 10CFR50, Appendix A nor the Standard Review Plan. As a result of these discussions, LaSalle is modifying Item d.6 and footnote 'n' to Table 3.6.3-1.

BASES FOR THE REQUESTED REVISIONS

Subsequent to the submittal of the original amendment request, an independent review of the modifications was performed. As a result of this review, it was determined that several additional design changes should be made to the modification package affecting the control logic for the 1(2)E22-F015 valves. These changes required this supplement to the original (Reference (a)) amendment submittal. The type of thermal overload protection bypass device for the 1(2)E22-F015 valves given in Technical Specification Table 3.8.3.3-1, "Motor Operated Valves Thermal Overload Protection," will be changed from "accident" to "continuous" conditions.

Section 6.2.4.II.6.e of the Standard Review Plan provides requirements for engineered safety systems associated with reactor containment isolation. This criteria is provided in lieu of the General Design Criteria of 10CFR50, Appendix A. Section 6.2.4.II.6.e of the Standard Review Plan specifies the following: "Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, ...". The NRC Staff has expressed concerns that the proposed valve configuration for the RCIC system following modification will not satisfy the requirements of Standard Review Plan Section 6.2.4.II.6.e. As a result of these discussions, LaSalle is modifying Item d.6 and footnote 'n' to Table 3.6.3-1. The change to footnote 'n' to Table 3.6.3-1 will require that in the event the line supplying water from the CST to the RCIC system is deactivated, then either valve 1(2)E51-F362 or 1(2)E51-F363 will be considered a primary containment isolation valve and closed when the RCIC system is not in use.

HPCS DISCUSSION

Section 6.2.4.2.3 of the UFSAR provides the design criterion for the Emergency Core Cooling System (ECCS) and Reactor Core Isolation Cooling (RCIC) system suction lines. Each of these systems are equipped with a remote manually operated suction line gate valve that provides assurance of being able to isolate these lines in the event of a break and provide for long-term leakage control. The suction piping from the suppression pool for each of the ECCS systems including the HPCS system is considered to be an extension of the containment since it must be available for long term usage under accident conditions. Isolation of the ECCS suction lines is required only in the event of a suction line break or for maintenance and surveillance activities.

Since the ECCS pumps are required to take suction from the suppression pool to mitigate the consequences of a loss-of-coolant accident, the normal position for the ECCS pump suction valves is open. HPCS is a unique ECCS system at LaSalle Station in that it was originally designed to first, take suction from the CSI and second, take suction under prescribed conditions from the suppression pool. The new design for the HPCS system leaves the suppression pool suction valve open and makes the system configuration very close to that of the Low Pressure Core Spray (LPCS) system. The modifications will change the thermal overload bypass circuitry for the 1(2)E22-F015 valve to be bypassed continuously as opposed to being bypassed under accident conditions only. This change will make the thermal overload bypass circuitry configuration for the 1(2)E22-F015 valve consistent with the circuitry for the LPCS system suppression pool suction valve (1(2)E21-F001).

Requirements of Regulatory Guide 1.106

Thermal overload protection devices on motors are designed primarily to protect continuous-duty motors while they are running. Valve motor operators are considered to be intermittent-duty devices and are routinely subjected to frequent start and stops. The cumulative effect of heating caused by this type of operation can lead to undesired actuations of the thermal overload protection devices. In a safety related application, the trip of a thermal overload protection device could unnecessarily hinder the successful completion of a required safety function. It is therefore desirable to have these protective devices bypassed.

Regulatory Guide 1.106 provides standards for control of thermal overload protection devices used on safety related valve motor operators. The Regulatory Guide offers various options for bypassing these devices to ensure that they do not hinder operation of a safety related valve motor operator under accident conditions. Under one option, the thermal overload protection devices are continuously bypassed and placed into force only when the valve motors are undergoing testing. Another option leaves the thermal overload protection devices in effect during normal operation and allows for their bypass automatically under accident conditions. Either of these options are equally acceptable under the guidance of Regulatory Guide 1.106.

ATTACHMENT A (continued)

Valve 1(2)E22-F015 Configuration

The normal operating position for valve 1(2)E22-F015 is open. This valve is only required to be isolated in the event of a break in the HPCS suction line that would require isolation. A second assumed failure in conjunction with the DBA is not considered as a credible event.

The configuration of valve 1(2)E22-F015 is consistent with all LaSalle Station ECCS systems. It is unnecessary to impose thermal overload protection on the 1(2) E22-F015 valve because its normal position (open) does not require manipulation in the event of a HPCS initiation. It is more appropriate for MOVs which are normally manipulated during other plant evolutions to utilize thermal overload protection devices to prevent the inadvertent degradation of the motor function of the valve. Such valves would require that its thermal overload protective devices be bypassed only during accident conditions to allow the valve to perform its intended safety function regardless of the consequence to the motor function of the valve. Thus, LaSalle Station's proposed modification to continuously bypass the thermal overload protective device for valve 1(2)E22-F015 does not adversely impact plant safety.

RCIC SYSTEM APPLICABILITY TO THE GENERAL DESIGN CRITERIA

The original system design for RCIC and the HPCS test line did not meet the requirements of the General Design Criteria specified in 10CFR50, Appendix A. In lieu of this, Standard Review Plan 6.2.4.II.6.e provides alternate criteria for engineered safety systems associated with reactor containment isolation. The original Safety Evaluation Report (SER, NUREG-0519) for LaSalle Station accepted CECO's alternate system designs when compared against the requirements specified in the Standard Review Plan. However, in subsequent discussions held with the NRC Staff in the time interval since the Reference (a) submittal, it has been determined that the proposed modification to the configuration of the RCIC system does not satisfy the requirements of the Standard Review Plan.

Section 6.2.4.II.6.e of the Standard Review Plan specifies the following: "Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, ...". LaSalle's proposed modification to the RCIC system specifies that in the event that the suction line from the CST to the RCIC system is deactivated, the alternate line (including manually operated valves 1(2)E51-F363 and 1(2)E51-F362) from the suppression pool can be utilized.

The NRC Staff has determined that the alternate line-up to the suppression pool does not provide greater system reliability because the 1(2)E51-F022 MOV is the only remotely operated isolation device in the system. As a result, in the event that the suction line from the CST to the RCIC system is deactivated, LaSalle will be required to keep either valve 1(2)E51-F362 or 1(2)E51-F363 (not both) closed during conditions when RCIC is not in use. A more complete analysis of this supplemental change is provided in the following section.

Supplemental RCIC Changes

Penetrations into the primary containment are required to have two barriers between the primary containment environment and the outside environment, unless it can be demonstrated that the containment isolation provisions for a specific class of lines are acceptable on some other defined basis (GDC 56). In the case of the RCIC system, which is normally in operation during post-accident conditions, the first barrier is a motor operated valve. The second barrier is the RCIC system itself which is considered to be a closed loop system (UFSAR Table 6.2-21 note 2B). The new full flow test line will connect the area between valves 1(2)E51-F022 and 1(2)E51-F059 (refer to Figure 2) to the primary containment. If the system is lined up to the CST (Mode 1 operation), the isolation valves on the new line to be installed (1(2)E51-F362 and 1(2)E51-F363), must be considered to be primary containment isolation valves, and must be locked closed. If the system is lined up to the suppression pool (Mode 2 operation), both of the manual valves may be open during the period of time that the full flow test line is in use. However, one of the valves must be closed at all other times. Valve 1(2)E51-F022 will become a primary containment isolation valve during Mode 2 operation. This valve automatically closes upon initiation of the RCIC system on a low reactor water level signal (-50 inches). The leak rate testing requirements for these valves are presented in Table 1.

CORRECTIONS TO TYPOGRAPHICAL ERRORS

Included with this package are corrections to typographical errors discovered since the original submittal to the NRC Staff (Reference (a)). The errors involved several valve numbers in two footnotes being added to Technical Specification Table 3.6.3-1 "Primary Containment Isolation Valves." Other errors involved the inadvertent omission of the deletion of footnote (d) from page 3/4 3-26 for Unit 1; changing the word 'Basis' to 'Buses' on pages 3/4 3-30 and 3/4 3-30a for Units 1 and 2; the inadvertent omission of the deletion of '***' from Table 4.3.3.1-1, Item A.1.h and B.1.f on pages 3/4 3-32 and 3/4 3-33, respectively, for Unit 1; the inadvertent omission of changing ':' to '.' on page 3/4 5-6 for Units 1 and 2. These changes are administrative in nature and have no impact on plant safety.

At the request of the NRC Staff, a simplified drawing of the modified HPCS system is attached with this evaluation (Figure 1).

SCHEDULE REQUIREMENTS

This modification is scheduled to be installed during the next refueling outage for each unit (Unit 1, October 1992; Unit 2, January 1992). Approval of this amendment is requested by November 15, 1991 to support the upcoming Unit 2, January 1992 refueling outage. It is requested that the amendments be made effective upon startup of each unit following the L2R04 and L1R05 refueling outages.

ATTACHMENT A (continued)

Table 1
RCIC FULL FLOW TEST VALVE
LEAK RATE TEST REQUIREMENTS

VALVE NUMBER	MODE ¹	VALVE POSITION ⁴		PRIMARY CONTAINMENT ISOLATION REQUIREMENTS
		STANDBY	TEST	
1(2)E51-F022	1	closed	open	Type "C" water leak rate test is required. An air leak rate test will normally be performed in order to allow a transition to Mode 2 operation any time during the fuel cycle.
	2	closed	open	Type "C" air leak rate test is required.
1(2)E51-F059	1	closed	open	No leak rate testing is required.
	2	closed	closed	Type "C" air leak rate test is required.
1(2)E51-F362	1	closed	closed	Type "C" air leak rate test is required.
	2	open/ closed	open	If closed ² , Type "C" air leak rate test is required.
1(2)E51-F363	1	closed	closed	Type "C" air leak rate test is required.
	2	open/ closed	open	If closed ² , Type "C" air leak rate test is required.
Blind Flange ³	1	open	open	No leak rate testing is required.
	2	closed	closed	Type "C" air leak rate test is required.

- Mode 1 - System is lined up for full flow test operation to the CST with the full flow valves to the suppression pool locked closed.
Mode 2 - System is lined up for full flow test operation to the suppression pool with the 1(2)E51-F509 valve locked closed and the downstream spectacle flange closed.
- Either the 1(2)E51-362 or the 1(2)E51-363 must be closed and leak rate tested in Mode 2.
- Spectacle flange installed downstream of 1(2)E51-F509 is open in Mode 1 and closed in Mode 2.
- Standby position - Position required for normal system standby operation in plant Operational Conditions 1, 2 or 3.
Test position - Position required for system operation in the full flow test mode.

Figure 1

HPCS System Current Alignment

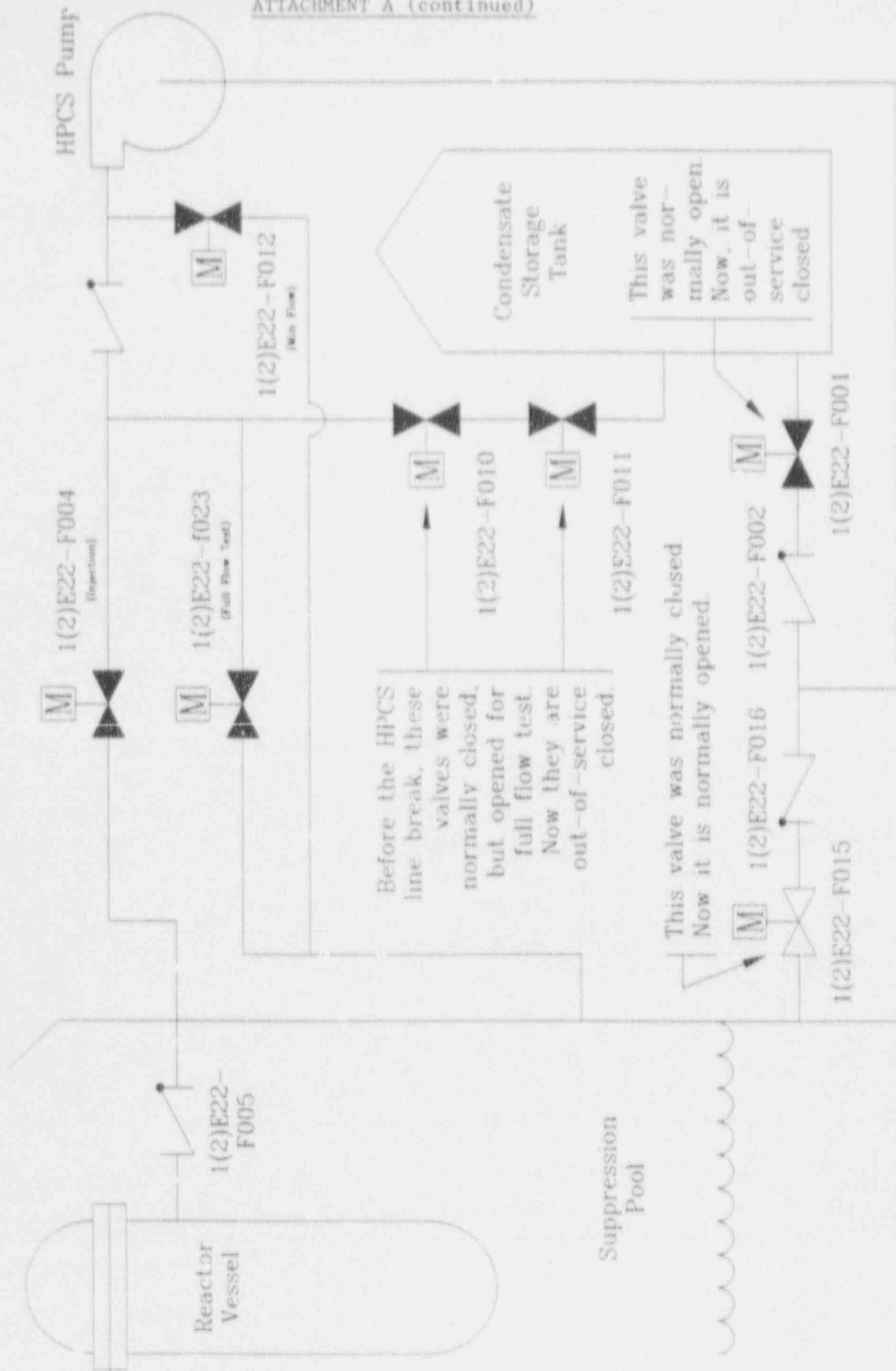


Figure 2

RCIC Full Flow Test Return To Suppression Pool

