UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In The Matter of	)	
	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454 0L
	)	50-455 0L
	)	
(Byron Nuclear Power Station, Units 1 & 2)	)	

RESPONSE OF COMMONWEALTH EDISON COMPANY
TO INTERVENOR'S PETITIONS FOR
WAIVER OF OR EXCEPTION TO
NRC REGULATIONS

This Response of Commonwealth Edison Company ("Edison or Applicant") answers two Petitions filed by the Rockford League of Women Voters (the "League") which purport to be in compliance with 10 CFR § 2.758.*/ The League styled its pleadings "Petition of Rockford League of Women Voters for Waiver of or Exception to Financial Qualifications Regulations" (the "Financial Qualifications Petition") and "Petition of Rockford League of Women Voters for Waiver of or Exception to Need for Power and Alternative Energy Source Regulations" 10 CFR §§ 51.23(e) and 51.53(c) (the

*/ In passing, we note that the League has suggested that Edison's response to its Petitions may be untimely, apparently based on its belief that the 10 day reply to motions requirement of 10 CFR § 2.730(c) is applicable. 10 CFR § 2.758(b) which authorizes responses to waiver petitions does not establish a time by which responses must be filed; we interpret this to mean that answers are due within a reasonable period of time. Given the rather voluminous filing by the League, Edison submits that its response is being filed within such a time frame.

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"Need for Power Petition") (the two Petitions will be hereinafter collectively referred to as the "Petitions"). Through its Petitions the League seeks admission, and in some cases readmission, of certain of its contentions previously filed in this licensing proceeding which purportedly relate to need for power and financial qualifications issues.

For the following reasons the Petitions must be denied, and the contentions dismissed. The Petitions seek authority to challenge the application of regulations of the U.S. Nuclear Regulatory Commission (the "Commission") in the Byron licensing proceeding, a practice which is generally disallowed by 10 CFR § 2.758(a). The Petitions are not, however, supported by proper affidavits which set forth with particularity the special circumstances alleged to justify the waiver or exception requested as required by 10 CFR § 2.758(b). Moreover, even assuming that the Petitions were supported by competent affidavits, they fail to make a prima facie showing of "special circumstances" as required by 10 CFR § 2.758(b) to justify waiver of or exception to the application of the Commission's regulations. The matters raised by the Petition merely reiterate generalized arguments that were considered and rejected by the Commission in promulgating the regulations in question. Further, the Petitions allege a number of "special circumstances" which are not supported by the exhibits attached thereto.

1. The League's Petitions Are Not Supported By Competent Affidavits.

The rule is clearly stated in 10 CFR § 2.758(a) that Commission regulations are generally not subject to attack in the course of an adjudicatory proceeding:

> (a) Except as provided in paragraphs (b), (c) and (d) of this section, any rule or regulation of the Commission, or any provision thereof, issued in its program for the licensing and regulation of production and utilization facilities, source material, special nuclear material or by-product material, shall not be subject to attack by way of discovery, proof, argument, or other means in any adjudicatory proceeding involving initial licensing subject to this subpart . . . .

The Petitions purport to invoke the exception under 10 CFR § 2.758(b) to the rule prohibiting an attack on Commission regulations in the course of an adjudicatory proceeding. 10 CFR § 2.758(b) provides:

> The sole ground for petition for waiver or exception shall be that special circumstances with respect to the subject matter of the particular proceeding are such that application of the rule or regulation (or provision thereof) would not serve the purposes for which the rule or regulation was adopted. The petition shall be accompanied by an affidavit that identifies the specific aspect or aspects of the subject matter of the proceeding as to which application of the rule or regulation (or provision thereof) would not serve the purposes for which the rule or regulation was adopted, and shall set forth with particularity the special circumstances alleged to justify the waiver or exception requested.....

The affidavits submitted by the League which purport to demonstrate the existence of special circumstances are affidavits of Myron M. Cherry, one of the attorneys for the League, which incorporate by reference a number of documents prepared by other individuals as well as the Petitions

themselves, and which, based on Mr. Cherry's "information and belief" are "true to the best of his knowledge." For the most part, the matters to which Mr. Cherry swears relate to complex, technical matters as to which Mr. Cherry neither demonstrates his competance to testify nor his personal knowledge of the underlying facts. As such, Mr. Cherry's affidavits amount to nothing more than an attempt to set forth special circumstances through evidence which is clearly entitled to no weight.

The requirement that an affidavit be submitted in support of a Petition filed under 10 CFR § 2.758(b) is quite obviously intended to accomplish more than to provide a means by which counsel for the petitioner can verify the truth and accuracy of the statements contained therein. Indeed, under 10 CFR § 2.708(c), the simple signing of a pleading by a party or its counsel accomplishes this result. An affidavit is required by § 2.758(b) to provide the Licensing Board, or where the question is certified, the Commission, with competent, admissible evidence regarding the existence of special circumstances which demonstrate that application of the regulation in question would not serve the purposes for which it was adopted. The fact that Mr. Cherry, a non-expert, may believe that the statements and expert opinions contained in the exhibits to be true is of no import. Such lay opinion is neither reliable nor probative evidence, and cannot reasonably be deemed adequate support for the League's Petitions.

Moreover, since it is Mr. Cherry, and not the individuals who purportedly prepared the exhibits incorporated into Mr. Cherry's affidavit, who is swearing to the truth and accuracy of the statements and opinions contained in the exhibits, application of the hearsay rule precludes the admission of the exhibits. In the context of summary disposition practice, it is clear that affidavits containing inadmissible evidence, such as is contained in Mr. Cherry's affidavits, would not be sufficient to defeat summary disposition motions. See Scott v. Dollahite, 54 FRD 430 (N.D. Miss. 1972) and Becker v. Koza, 53 FRD 416 (D. Neb. 1971), two instances in which courts granted summary judgment in favor of plaintiffs where defendants submitted affidavits of counsel containing inadmissible evidence. Given the strong Commission policy against admitting contentions which challenge Commission regulations, we know of no reason the same standards governing the adequacy of affidavits in the summary disposition context should not apply to affidavits filed pursuant to § 2.758.

In short, the League's effort amounts to an impermissible attempt to circumvent the requirement that a petitioner present competent evidence in support of its § 2.758 petition by means of an affidavit. Having failed to comply with this requirement, the affidavits should be stricken and the Petitions denied.

2. The Financial Qualification Petition fails to make a prima facie showing of "special circumstances".

Even assuming, for the sake of argument, that the affidavits submitted in support of the League's Petitions

were acceptable, the League has failed to demonstrate the existence of special circumstances which would entitle it to a waiver of or exception from the regulations in question. The reason for the recently enacted regulations eliminating the need for financial qualification review for electric utilities was clearly stated by the Commission when it proposed the new rule: "The Commission believes that its existing financial qualifications review has done little to identify substantial health and safety concerns at nuclear power plants." 46 Fed. Reg. 41786 (August 3, 1982). The Commission also reasoned that "the Commission's inspection and enforcement activities provide more effective protection of public health and safety." 47 Fed. Reg. 13751. This belief of the Commission is based on the experience at other facilities such as Seabrook which revealed no demonstrable link between the financial difficulties of the utility and the safety of the public. Public Service Company of New Hampshire, et al., (Seabrook Station, Units 1 and 2), CLI - 78 - 1, 7 NRC 1, 8 (1978). The Commission received no comments to the proposed rule to persuade it to significantly change its reasoning. 47 Fed. Reg. 13751 (March 31, 1982).

The Financial Qualification Petition does nothing more than parrot the arguments made in the course of the recent rulemaking. The League fails to raise any issue which was not considered and resolved by the Commission. Hence, the Petition must be denied for failure to show "special circumstances" such that the application of the

rule eliminating financial qualification review would not serve the purpose for which the rule was adopted.

The League goes to great lengths to allege that Edison will not be able to finance the completion of construction, operation, and decommissioning of the Byron facility.*/ However, the Commission stated that it "does not find any reason to consider, in a vacuum, the general ability of utilities to finance the construction of new generation facilities ... [o]nly when joined with the issue of adequate protection does this issue become pertinent." 47 Fed. Reg. 13751. The League fails completely to establish any link between Edison's alleged financial straits and any public health and safety concern. At best, the Financial Qualifications Petition speculates that Edison is "likely" to "cut corners" in construction and speculates that Edison will attempt to commence operation of the Byron facility without the funds necessary to protect the public health and safety. Financial Qualifications Petition at 10-11. These claims are wholly unsupported by the testimony cited in the Petition.

During the course of the rulemaking proceeding the Commission considered and rejected the argument that the

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*/ It should be noted that insofar as the League's financial qualifications contentions were admitted by the Licensing Board, prior to the promulgation of the rule eliminating financial consideration matters in licensing proceedings, the Board excluded issues related to the financial ability of Edison to complete construction of the Byron Station. The Board properly reasoned that at the operating license stage the issue of the utilities' financial ability to complete construction is not appropriate. Commonwealth Edison Company, (Byron Nuclear Power Station, Units 1 and 2), LBP-80-30, 12 NRC 683 at 692, (1980).

inability to recover costs provides an incentive for utilities to skimp on important safety components and quality assurance standards. The Commission cited the experience at other plants that showed that the difficulty in financing has not led to actions which have adversely affected public safety. Rather the response of most other utilities encountering financial difficulties has been to postpone or cancel their plants, "actions clearly not inimical to public health and safety under the Atomic Energy Act." 47 Fed. Reg. 13751. */ Further, the Commission accepted the argument of commenters to proposed rule which reasoned:

> "cutting corners" in construction or operation is not in the self-interest of the utility, as it is imperative that a plant provide long-term operating reliably and safely in accordance with NRC regulations...[and]...financial savings that could be achieved through 'corner-cutting' would be small compared to the sums required to complete the project.

46 Fed. Reg. 41787. In short, the League's arguments have already been considered and rejected by the Commission. The League has provided no facts from which the Board could reasonably conclude that in the case of Edison there is a direct relationship between its financial condition and "corner-cutting."

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*/ This response of electric utilities faced with financial difficulties contemplated by the Commission is precisely the action Edison plans to take should inadequate financing for the construction of Byron facility be attained. Edison's intent in this regard is evidenced by the testimony of Robert J. Schultz, Edison Vice President: "Without the ability to issue either senior or junior debt it would become necessary for us to suspend all construction and take advantage of every other measure available to use to conserve funds." Exhibit A to Financial Qualifications Petition at 8.

The Petition also alleges that NRC inspection and enforcement is inadequate to detect and correct threats to the public health and safety from alleged "corner-cutting" in construction. Financial Qualifications Petition at 11. This same argument was raised by opponents to rule eliminating financial qualifications review for electric utilities. The Commission responded that "in the absence of facts to the contrary, the Commission cannot accept unsupported statements that as a general matter its inspection and enforcements efforts are inadequate." 47 Fed. Reg. 3751. The League attempts to establish such "contrary facts" by pointing to current allegations relating to construction practices at the Edison's LaSalle Plant which went "undetected or unreported by NRC inspectors, and were brought to light only because a local environmental group took the initiative to interview construction workers." Financial Qualifications Petition at 11. However, contrary to the League's implication, the fact that the LaSalle construction practices are now being reviewed by the NRC indicates that the NRC is fulfilling its responsibility of inspection and enforcement to assure the public health and safety. Moreover, the results of a recent NRC investigation into these matters demonstrates the lack of merit regarding the substance of the League's claim, regarding the impact on safety of Edison's alleged financial difficulties. On July 19, 1982, the NRC completed its review of the allegations made

regarding Edison's LaSalle Station insofar as these allegations related to the question of whether Edison should be authorized to operate beyond zero power, and based upon that review low power operation was authorized. (A copy of the report and NRC's approval of low power operation is attached.)

Finally, the League's characterization of Edison's financial situation is largely erroneous. The League's claim that there is no reasonable assurance that the Illinois Commerce Commission ("ICC") will grant the necessary rate relief is completely unsubstantiated by the evidence cited. The fact that Edison did not receive all that it requested in the interim rate order does not reflect a growing resistance of the ICC to Edison's construction program. The interim rate increase referred to was granted for the purpose of assuring Edison's continuing ability to finance its construction program during the pendency of hearings on the appropriate level of permanent rates. The difference between the amount requested and the amount granted does not indicate a resistance of the ICC to the construction program but rather a different perception as to the appropriate level of financing. Moreover, the ICC has expressly recognized the importance of early completion of Edison's construction program for the benefit of Edison and its ratepayers. In an investigation of Edison's plant construction program, the ICC stated that Edison "has a duty to its ratepayers to complete the generating units under construction in as timely a manner as the construction and licensing constraints

allow." Order, ICC Docket No. 78-0646, October 15, 1980, p. 12.

In sum, the Financial Qualifications Petition fails in all respects to make any showing of "special circumstances" with respect to the Byron facility such that the application of the rule eliminating financial qualifications review for electric utilities would not serve the purpose of that rule. Therefore, the Financial Qualifications Petition must be denied.*/

3. The Need for Power Petition fails to make a prima facie showing of "special circumstances".

The recently enacted amendments to the Commission's regulations relating to the Need for Power Petition provide that for purposes of the National Environmental Policy Act (NEPA), need for power and alternative energy sources will not be considered in operating license proceedings. The purpose of this rule is to "avoid unnecessary consideration of issues that are not likely to tilt the cost-benefit balance by effectively eliminating need for power and alternative energy source issues from consideration at the operating license stage." 47 Fed. Reg. 12940 (March 26, 1982). The Commission reasoned that, based on past experience, the situation at the operating license stage is

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*/ In passing we note that the League, through Mr. Cherry, asserts its belief that additional information will be available in the near future which will demonstrate the existence of special circumstances. Quite obviously, this assertion cannot properly serve as an adequate basis for granting an exception to or waiver of application of the Commission's regulations.

such that "the plant would be needed to either meet increased energy needs or replace older less economical generating activity and that no viable alternatives to the completed nuclear plant are likely to exist which could tip the NEPA cost benefit balance against issuance of the operating license." 47 Fed. Reg. 12940 (March 26, 1982).

As noted by the League, the comments to the new rule specifically address special circumstances under 10 CFR § 2.758(b) which would justify waiver of or exception to the regulation. The League claims that its Petition demonstrates that a waiver or exception should be granted. However, the League fails to appreciate the burden imposed by § 2.758:

> Section 2.758(c) requires the petitioning party to make a prima facie showing that application of the regulation to a particular aspect of the proceeding would not serve the purposes for which the rule was adopted. This is a much stricter standard than the current requirements for raising need for power and alternative energy sources in OL proceedings.

47 Fed. Reg. 12941 (March 26, 1982). The League's contentions on need for power and alternative energy sources were denied under the prior rule. Byron, supra, 12 NRC 683 at 691-692. The League should not be allowed to revive those contentions at this time in the face of the stricter standard of a prima facie showing of "special circumstances."

The League attempts to demonstrate the existence of special circumstances based on new evidence. Need for Power Petition at 2. However, the allegations merely repeat the arguments considered by the Commission in the recent

rulemaking and the evidence cited by the League is inconsequential if not totally irrelevant.

The Petition alleges that there is no need for the power to be generated at the Byron facility. In support of its Petition the League alleges that demand projections are inflated and costs are understated. Need for Power Petition at 3-4. Regardless of the merits of the League's claim, it is inconsequential for purposes of showing special circumstances since during the course of the rulemaking the Commission conservatively assumed that "the plant is not needed to satisfy increased energy needs but rather is justified, if at all, as a substitute for other generating capacity." 46 Fed. Reg. 39441 (August 3, 1981). In this latter regard, the League's own evidence points out that the Byron plant "should produce relatively cheap electricity during the 1990's and the first decades of the 21st century." Testimony of Irvin C. Bupp, Exhibit E to Financial Qualifications Petition at 8.

The League also claims that the Commission incorrectly assumed that alternative energy sources would be more expensive and environmentally inferior. However, the League fails to identify any relevant evidence to support this contention. In support of its claim the League cites two documents relating to other nuclear facilities owned by other utilities and located in different geographic regions, as well as "further expert testimony relating specifically to C.E. and the Byron facility expected to be available shortly."

Quite obviously, the materials on which the League relies in its attempt to demonstrate the existence of special circumstances cannot possibly be deemed to establish the existence of special circumstances with respect to Byron which support the requested waiver or exception. Moreover, the League's assertion that nonoperation of the Byron facility is an environmentally superior alternative is refuted by the very exhibits on which it relies. It is clear from Dr. Bupp's testimony that the operation of Byron will result in decreasing the amount of coal-produced electricity (Testimony of Irwin C. Bupp, Exhibit E to Financial Qualifications Petition at 9-12). Without question, the environmental consequences of operating a coal generating plant are greater than for a nuclear plant.*/ See Byron FES-CP at 9.1.2.2. Thus, non-operation of Byron is an environmentally inferior alternative.

Finally, the League argues that the alleged decline in power since the FES-CP in 1974 and "other evidence" alters the cost-benefit analysis performed at the construction stage. In both the proposed and final rules, however, the Commission expressly considered the possibility that certain factors could change between the construction permit proceeding and and the operating license proceeding. 47 Fed. Reg. 12,942 (1982); 46 Fed. Reg. 39,441 (1981). Thus, the

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*/ For the purposes of the Need for Power Petition, only the environmental consequences of the operation of the plant are at issue because the plant must be completed for either 10 CFR § 51.23(e) or 10 CFR § 51.53(c) to apply.

Commission has specifically considered this argument and rejected the notion that changes from the CP stage to the OL stage, without more, constitute special circumstances sufficient to justify waiving the requirements of the regulations.

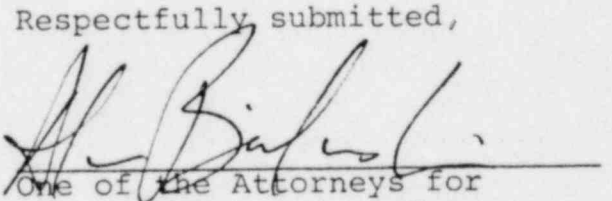
As shown above, the League has failed to show that any special circumstances exist that would justify the consideration of the need for power or alternative energy sources. Because they have not satisfied even the most basis requirements of 10 CFR § 2.758, the League's request must be denied.

CONCLUSION

For all of the foregoing reasons, the League's Petitions seeking a waiver of or exception to the Commission's regulations which preclude litigation of financial qualifications, need for power, and alternative energy sources in this proceeding should be denied, and the contentions bearing on these issues should be dismissed.

Dated: July 22, 1982

Respectfully submitted,



One of the Attorneys for
Commonwealth Edison Company

Michael I. Miller
Alan P. Bielawski
ISHAM, LINCOLN & BEALE
Three First National Plaza
Chicago, Illinois 60602
(312) 558-7500

Joseph Gallo
ISHAM, LINCOLN & BEALE
1120 Connecticut Ave., N.W.
Suite 840
Washington, D.C. 20036
(202) 833-9730

CERTIFICATE OF SERVICE

The undersigned, one of the attorneys for Commonwealth Edison Company, certifies that on this date he filed two copies (plus the original) of the attached pleading with the Secretary of the Nuclear Regulatory Commission and served a copy of the same on each of the persons at the addresses shown on the attached service list in the manner indicated.

Date: July 22, 1982

Alan P. Bielawski

SERVICE LIST

COMMONWEALTH EDISON COMPANY -- Byron Station
Docket Nos. 50-454 and 50-455

**Morton B. Margulies, Esq.
Administrative Judge and Chairman
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

**Dr. Richard F. Cole
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

***Myron M. Cherry, Esq.
Cherry & Flynn
Three First National Plaza
Suite 3700
Chicago, Illinois 60602

*Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Chief Hearing Counsel
Office of the Executive
Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Dr. A Dixon Callihan
Union Carbide Corporation
P.O. Box Y
Oak Ridge, Tennessee 37830

**Mr. Steven C. Goldberg
Ms. Mitzi A. Young
Office of the Executive Legal
Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Atomic Safety and Licensing
Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Secretary
Attn: Chief, Docketing and
Service Section
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Ms. Betty Johnson
1907 Stratford Lane
Rockford, Illinois 61107

**Ms. Diane Chavez
SAFE
608 Rome Ave.
Rockford, Illinois 61107

*Dr. Bruce von Zellen
Department of Biological Sciences
Northern Illinois University
DeKalb, Illinois 60115

*Joseph Gallo, Esq.
Isham, Lincoln & Beale
Suite 840
1120 Connecticut Ave., N.W.
Washington, D.C. 20036

***Douglass W. Cassel, Jr.
Jane Whicher
BPI
Suite 1300
109 N. Dearborn
Chicago, IL 60602

* Via U.S. Mail

** Via Express Mail

*** Via Messenger

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 19 1982

Docket No. 50-373

Mr. Cordell Reed
Vice President - Nuclear Operations
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dear Mr. Reed:

Subject: La Salle County Station, Unit 1 - Approval to Proceed above Zero Power Physics Testing

We have completed our review of your reports dated May 7, 1982 and May 18, 1982, which related to allegations submitted by the Attorney General of Illinois and Illinois Friends of the Earth. We have concluded that the matters of License Condition 2.C(4) have been resolved to our satisfaction. Specifically, that license condition stated:

> "The licensee shall complete its assessment of the rebar damaged due to drilling and coring in concrete and the structural adequacy of the off-gas building roof. The results shall be reported to the NRC staff for review and approval, prior to power operation following initial criticality and zero power physics testing."

Accordingly, you are authorized to proceed beyond the zero power physics testing and to operate La Salle County Station, Unit 1 at reactor core power levels not in excess of 5 percent of full power (156 megawatts thermal) in accordance with the conditions specified by License No. NPF-11.

The NRC staff's efforts in dealing with these allegations extended considerably beyond the originally defined areas. A copy of the NRC staff findings with respect to these investigations is being provided to you by NRC's Region III.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

cc: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JUL 19 1982

Docket No. 50-373
Docket No. 50-374

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the special safety inspection conducted by Mr. I. N. Jackiw, and others of the Region III and NRR staffs during May through July 1982, of activities at LaSalle County Nuclear Power Station, Unit 1, authorized by NRC Operating License No. NPF-11.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, interviews with personnel, and special tests.

This special inspection was conducted in response to several allegations regarding the adequacy of construction at the LaSalle Station. The enclosed report is limited to the identification and resolution of technical issues. Further inspection and/or investigation activities will be conducted to determine compliance with regulatory requirements and the extent of possible records falsification. Appropriate enforcement action will be initiated separate from this report.

Subsequent to preparation of the attached report, plans were made to hold a meeting with organizations who had provided the Region III staff with specific allegations that had been referred to them. During a telephone conversation with a representative of the Government Accountability Project on July 15, 1982, the Regional Administrator was advised that the staff had been provided with information regarding the Heating, Ventilation and Air Conditioning (HVAC) Company's work at LaSalle. Such information had not been pursued during the special inspection.

It was subsequently learned that on May 3, 1982, an individual came to the Region III office with allegations regarding operations of the HVAC Contractor. His allegations were considered by the staff to be general in nature and primarily directed at the company's activities at the Midland site. Subsequently, he provided specific records to the Region III staff. These were reviewed, and identified for followup at a later date.

Separately, another individual had been contacted by the staff who also claimed to have allegations regarding the HVAC Contractor's activities at LaSalle. (This individual's name had been provided to the staff during a June 2, 1982 meeting in the Region III office.) However, the only written information received by Region III from this individual pertained to problems at the Midland site, and no information was received by telephone or in writing specific to LaSalle. Consequently, the special inspection, as documented in the attached report, did not include any review of the HVAC Contractor's activities.

Following a review on July 16, 1982, of the information received earlier, it is our conclusion that no reason exists to preclude the LaSalle Unit 1 from going beyond zero power. We plan to pursue the details further, and will report our findings in a subsequent report. If our further inspection identifies problems of safety significance, appropriate regulatory action will be taken.

In conclusion, it is the staff's view that LaSalle Unit 1 can be operated above zero power. This recommendation is being made to the Office of Nuclear Reactor Regulation. It should be clear, however, that no action on your part is authorized until NRR has appropriately amended your operating license.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

James G. Keppler
Regional Administrator

Enclosure: Inspection Reports
No. 50-373/82-35(DETP) and
No. 50-374/82-06(DETP)

cc w/encl:
Louis O. DelGeorge, Director
of Nuclear Licensing
D. L. Shamblin, Site
Construction Superintendent
T. E. Quaka, Quality
Assurance Supervisor
R. H. Holyoak, Station
Superintendent
B. B. Stephenson, Project Manager
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII
Karen Borgstadt, Office of
Assistant Attorney General
Judith S. Goodie, Assistant
Attorney General
Bridget Little Rorem, Illinois
Friends of the Earth
T. Devine, Government Accountability
Project

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-373/82-35(DETP); 50-374/82-06(DETP)

Docket No. 50-373; 50-374 License No. NPF-11; CPPR-100

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: LaSalle Nuclear Power Station, Units 1 and 2

Special Inspection At: LaSalle Site, Marseilles, Illinois
Sargent and Lundy, Chicago, Illinois

Inspection Conducted: May 25 - July 11, 1982

Inspectors who conducted the inspection activities are identified at the beginning of each appropriate Section of the report details.

Approved By: I. N. Jackiw, Chief
Test Program Section

W. E. Norelius for JNJ

Inspection Summary

Inspection on May 25 - July 11, 1982 (Report No. 50-373/82-35(DETP); 50-373/82-06(DETP))

Areas Inspected: Special inspection to follow up on allegations/concerns regarding construction deficiencies at the LaSalle County Site. The inspection involved a total of 1,580 inspector-hours on site by several NRC inspectors.

Results: Thirty-six separate allegations of varying significance were identified. Of these, twenty required prompt consideration regarding the operation of Unit 1; others will be considered later; some require no further NRC action. This report describes allegations, findings and the resolution of the technical issues. Any enforcement action which may result from further inspection/investigation activity will be handled separately.

SUMMARY AND CONCLUSION

As a result of allegations received in Region III by telephone, a request to institute a Show Cause Order from the Attorney General, State of Illinois, on March 24, 1982, a further 10 CFR 2.206 request received from Illinois Friends of the Earth on April 28, 1982, and an amended petition from the Attorney General, State of Illinois, on May 3, 1982, a special inspection was conducted into alleged inadequacies in construction of the LaSalle nuclear facility. As part of this inspection, individuals who filed affidavits with the above referenced petitions and others whom they identified as having pertinent information were contacted. Additional concerns identified during those interviews were also addressed during this inspection.

From the affidavits and statements of those interviewed, thirty-six separate alleged problems were identified. Site tours were conducted with several individuals to identify specific areas of concern. The allegations were categorized by the NRC staff as matters requiring prompt resolution prior to the operation of the Unit 1 facility (Category 1), those matters which require followup on a longer time frame, but have no direct impact on the operation of Unit 1 (Category 2), and those matters which do not require further NRC involvement (Category 3).

Twenty of these allegations were considered to be Category 1 items. Some of the allegations were not substantiated. For several others, the facts stated by concerned individuals were correct; however, these were found to be acceptable when the entire system of controls was examined. One allegation relating to improper site security resulted in finding violations of the licensee's security requirements. These matters were brought to the licensee's attention and promptly corrected. One allegation of falsification of certain calibration records by a site contractor was substantiated. Technical problems resulting from this finding were identified, engineering assessments were made, and a corrective action program was defined and undertaken by the licensee.

Of those matters which were considered important to the operation of Unit 1, all have been pursued to the extent that they were either found not to present a problem to public health and safety, or further licensee action has been defined and completed to provide similar assurance. The staff concludes that with regard to the matters addressed in this report, there are no remaining technical issues to preclude the licensee from operating Unit 1 above zero percent power.

DETAILS

Persons Contacted

Commonwealth Edison Company

C. Reed, Vice President
L. O. DelGeorge, Director of Nuclear Licensing
J. J. Maley, Manager of Projects
T. E. Quaka, Site QA Superintendent
R. T. Rose, Lead Structural Engineer
C. Schroder, Nuclear Licensing
D. L. Shamblin, Staff Assistant Project Manager
W. J. Shewski, Manager of QA
D. J. Skoza, QA Engineer
B. B. Stephenson, Project Manager
G. Marcus, Director of QA

Sargent and Lundy (S&L)

R. J. Mazza, Project Director
H. S. Taylor, Head, Quality Assurance Division
E. R. Kurtz, Supervisor, Project Section, QC Division
M. E. Schuster, Head, Quality Control Division

Walsh Construction Company

M. R. Dougherty, QA Manager

Morrison Construction Company

K. J. Hamilton, Project Manager
T. G. O'Conner, Superintendent
M. Wherry, QC Supervisor

Numerous other licensee and contractor staff members were interviewed during the course of this inspection.

Background Information

In early 1982, the Region III office received allegations regarding construction activities at the LaSalle Nuclear facility. Two of these allegations related to (1) inadequate roof slab thickness for the offgas building, and (2) improper coring and drilling of holes in the walls of safety related structures. The Region III staff concluded that the offgas building was not a safety related structure, and therefore no action was taken. The staff concluded, however, that the allegation regarding concrete coring and drilling activities merited further investigation. An onsite inspection was initiated on March 24, 1982, into the coring and drilling program requirements of the licensee and its principal electrical contractor, the H. P. Foley Company.

Also, on March 24, 1982, the Attorney General for the State of Illinois initiated a request to institute a Show Cause Proceeding under the provisions of 10 CFR 2.206. This petition covered the same two concerns identified above regarding the offgas building and coring and drilling activities. Attached to that request were two affidavits. One was from a laborer who related his activities in drilling holes at the site. The second affidavit was from a consultant who stated that the coring and drilling activities which damaged reinforcing steel could weaken the structure if not properly controlled.

Because the Region III office had made the initial determination not to look into concerns regarding the thickness of the concrete roof slab on the offgas building, it was determined that the NRC's Office of Inspection and Enforcement would independently review this matter. The Office of Inspection and Enforcement completed its review on April 14, 1982. They agreed with the initial Region III assessment that the offgas building was a non-safety related structure. However, they also pursued the question of the roof slab thickness, and found that the roof was built as designed with the proper thickness.

The Director of NRC's Office of Nuclear Reactor Regulation requested a meeting with the licensee and its architect engineer, Sargent and Lundy, on March 31, 1982 to discuss their programs for determining that their coring and drilling activities did not unduly weaken facility structures.

NRR and the Region III staff began a technical assessment of the report provided at the meeting by the licensee and Sargent and Lundy. An inspection at Sargent & Lundy to review engineering judgment and calculations which had been performed with regard to damaged reinforcing steel, was performed on April 8, 1982. As a result of the initial look at the programs of Commonwealth Edison and Sargent & Lundy, a report was issued by Region III (Inspection Report No. 50-373/82-21) on April 27, 1982.

On April 28, 1982, the Illinois Friends of the Earth issued an additional request to institute a proceeding to Show Cause and provided four additional affidavits which contained additional allegations of improper construction practices at the LaSalle facility. The same issues were addressed in an amended petition from the Attorney General, State of Illinois, by letter dated May 3, 1982.

A special inspection was initiated to review the allegations made in the affidavits and by additional individuals whose names were obtained from the affiants, the Illinois Attorney General's office, the Illinois Friends of the Earth and the Government Accountability Project. A meeting was held with these groups on June 2, 1982 to assure that the Region III staff's understanding of their concerns was accurate and complete. Additional names of individuals who had expressed concern about activities at LaSalle were provided during this meeting.

Interviews were conducted with concerned individuals. Based on these interviews and the statements provided, 36 separate issues were defined. The allegations and concerns ranged considerably in detail and safety

significance. Several items were alleged to have occurred as early as 1975.

The Region III staff decided to categorize these issues such that resolution of the safety significant issues affecting Unit 1 operation would receive top priority.

Category 1 matters are those needing prompt resolution prior to power operation; Category 2 matters were judged to require followup, but on a longer time frame; and Category 3 matters require no further action by the NRC staff. Category 2 matters included allegations relating to Unit 2, personnel matters, and activities that did not have an immediate safety impact. Category 3 matters included allegations which were too general to pursue, involved non-safety related systems or are subject to other regulatory jurisdictions. The summary of the items and category of each are set forth in Attachment A to this report.

Allegations/Concerns

The remainder of this report shows the Category 1 allegations and the NRC findings for each. (The details are separated by Sections based on the individual inspectors who reviewed the concerns. A single Section may address more than one allegation.

SECTION I

Prepared by: F. C. Hawkins C E Norelius for FCH

J. H. Neisler C E Norelius for JHN

S. P. Chan C E Norelius for SPC

J. R. Kniceley J R Kniceley

1. Drilling/Coring in Concrete

Generally, concerns were expressed regarding indiscriminate concrete coring and drilling, and inadequate implementation of the site contractors' drilling and coring quality control programs at the LaSalle site.

The specifics regarding these subjects were received by the NRC over the period March 24-May 26, 1982. The results of our initial investigation into the concerns are documented in Inspection Report No. 82-21. Subsequent to the issuance of Inspection Report No. 82-21, the Illinois Attorney General's Office provided the NRC additional affidavits regarding alleged improper concrete drilling and coring activities. In order to fully address the new concerns, the scope of the previous inspection effort was expanded to include all site contractors who performed drilling or coring work at LaSalle.

The purpose of this inspection was to determine the adequacy of the site contractor's programs to control and document concrete drilling and coring activities. This inspection included detailed review of drilling/coring procedures, personnel interviews, observation of in-process work, and review of quality records pertaining to work performed by Walsh Construction Company, Reactor Controls, Inc., Commonwealth Electric Company, Mid-City Architectural Iron Company, H.P. Foley Company, Commercial Concrete Drilling and Sawing Company, Morrison Construction Company, and the Zack Company.

Previously, the acceptability of the Sargent and Lundy program to assimilate and properly assess the field supplied drilling data and to properly control and assess the structural effects of concrete drilling and coring was determined by NRR and Region III on April 8, 1982. The summary evaluation prepared by the Structural Engineering Branch, NRR, is included as Attachment B to this report. The results of the field inspection are documented in Inspection Report No. 82-21 shown as Attachment C to this report.

All available correspondence and documentation regarding the issue was reviewed. This included the following:

- Petition by the People of the State of Illinois by Tyrone C. Fahner, Attorney General, State of Illinois, pursuant to 10 CFR 2.206, dated March 24, 1982.

- Transcript, U.S. Nuclear Regulatory Commission, Docket Nos. 50-373 and 50-374, Room P-422, 7920 Norfolk Avenue, Bethesda, Maryland, March 31, 1982.

- Commonwealth Edison Company submittal of March 31, 1982, entitled "Response to Petition Made by the Office of the Attorney General, State of Illinois, In the Matter of Reinforcing Steel Damaged During the Installation of Cored Holes and Concrete Expansion Anchors."

- Filing by the Illinois Attorney General Office, entitled "Comments of the People of Illinois on Commonwealth Edison Company's Presentation of March 31, 1982", dated April 13, 1982.

- Filing by the Illinois Attorney General's Office, entitled "Amendment to Request for Show Cause Proceding", dated May 3, 1982.

- Commonwealth Edison Company submittal of May 7, 1982, entitled "Final Report in Response to Petition Made by the Office of the Attorney General, State of Illinois, in the Matter of Reinforcing Steel Damaged During the Installation of Cored and Drilled Holes and the Matter of the Off-Gas Building Roof for LaSalle County, Units 1 & 2."

- Commonwealth Edison Company submittal of May 18, 1982, entitled "Report in Response to Amended Petition dated May 3, 1982 Made by the Office of the Attorney General, State of Illinois, in the Matter of Reinforcing Steel Damaged During the Installation of Cored and Drilled Holes for LaSalle County, Units 1 & 2."

- Letter from the Illinois Attorney General's Office to Mr. H. Denton, Director of Nuclear Reactor Regulation, dated May 26, 1982.

The March 24, April 13, May 3, and May 26 982 transmittals from the Illinois Attorney General's Office were reviewed to identify specific items of concern dealing with concrete drilling and coring activities. Paragraphs 1. through 4. of this section identify those items, as understood by the NRC, and the results of our investigation into each. Paragraph 5. reports the results of interviews conducted with personnel knowledgeable of drilling and coring activities at LaSalle. Paragraph 6. documents the results of a review of procedures for LaSalle contractors engaged in drilling or coring activities. Paragraph 7. reports on field inspection of in-process coring activities and the inspection of cores which were retained from previous coring work.

1. <u>March 24, 1982 Filing by the Attorney General of the State of Illinois</u>

This transmittal contained an affidavit from Mr. E. Garrison, which the Attorney General believes alleged certain shortcomings with Commercial Concrete Sawing and Drilling Company's (an

H.P. Foley Company subcontractor) concrete drilling and coring program. During their tenure at LaSalle, Commercial Concrete Drilling and Sawing Company used the Foley procedures and Sargent & Lundy Specification LS-CEA to accomplish all drilling and coring work.

The concerns which the Attorney General raised from the affidavit were of both a specific and general nature. These concerns were evaluated and the results documented in Inspection Report No. 82-21. Report 82-21 described the H.P. Foley Company drilling and coring program with respect to engineering design control, field activity control, quality inspection, and documentation of reinforcing steel damage, and addressed the Attorney General's concerns from a programatic standpoint.

The two specific incidents which Mr. Garrison referred to in his affidavit were reviewed during this inspection. Following each statement are the results of our inspection of these specific incidents.

a. Statement: "On one occasion, I drilled a 6" diameter hole through rebar in the reactor building of Unit 1, at an elevation below 710'. It was at a place where the steel tied together, and removed about 25-40 pounds of steel."

   Finding: H. P. Foley daily reports of core drilling and concrete expansion anchor installation were reviewed. One instance was identified in which Mr. Garrison cored a two foot deep, six inch diameter hole on June 22, 1979.

   The hole was located in the Unit 1 reactor building at elevation 688.6', five feet west of G-line and eight feet south of 14 line. Sargent & Lundy structural drawing S-251, Revision W, dated February 6, 1979, clearly specified the location of the core hole prior to the work being performed. This approval was based on conservative engineering analysis of reinforcing steel likely to be cut during the coring operation. NRC review of this engineering function is documented in Attachment B.

   Calculations were performed by Sargent & Lundy to estimate the amount of reinforcing steel actually cut by the coring operation. The calculations were based on Mr. Garrison's sketch of damaged reinforcing steel which he provided as part of the H.P. Foley daily report. They estimate that approximately 5-6 pounds of steel was cut and removed. The NRC inspector considers the Sargent & Lundy estimate to be valid.

   In this instance, the cutting of reinforcing steel does not constitute a nonconforming condition. This is because

Sargent & Lundy gave prior approval, based on an engineering assessment, to core drill the hole. (See reference made to "Office routed cores" in Inspection Report No. 82-21)

In conclusion, the NRC has verified that the activities surrounding and documentation regarding this core were conducted in accordance with CECo quality program requirements, and H.P. Foley procedures.

b. Statement: "On a second occasion, I drilled a 7" diameter hole in the reactor building of Unit 1 at elevation 735. I hit the 2" rebar, and as I continued to drill the rebar was splitting. That hole was drilled to a depth of 6 to 7 feet, where we hit a beam in the floor of a room where steam pipes were located. This hole was later grouted in, because it was improperly located."

Finding: H.P. Foley daily reports of core drilling and concrete expansion anchor installation were reviewed. One instance was identified in which Mr. Garrison cored a four foot-eight inch deep, seven inch diameter hole on April 28, 1979. The hole was located in the Unit 1 reactor building at elevation 753', two feet-three inches east of H line and 12.8 line. The hole had been abandoned and subsequently grouted back. Review of structural drawings did not reveal the presence of any structural steel in the vicinity of this cored hole.

On May 17, 1982, Mr. Garrison was interviewed by two Region III inspectors. During the interview with Mr. Garrison, he provided no additional details. He declined taking any plant tour to describe or point out the holes he referred to and, asked not to be contacted further.

On June 9, 1982, a tour of the plant was conducted with another former employee to identify locations of specific alleged construction problems. During the tour, this specific hole was identified by the individual as being the one referred to in Mr. Garrison's statement.

The location of the core hole was approved by Sargent and Lundy prior to the work being performed as evidenced by Sargent and Lundy structural drawing S-211, Revision U, dated July 19, 1978. This approval was based on conservative engineering analysis of reinforcing steel likely to be cut or damaged during the coring operation. NRC review of this engineering function is documented in Attachment B.

With specific regard to the cored holes, neither of these instances constitutes a nonconforming condition since approval based on an engineering assessment was given by Sargent & Lundy. (See reference made to "Office routed cores" in Inspection Report No. 82-21).

In conclusion, the actions of CECo and H.P. Foley Co. regarding the drilling of this core were in accordance with quality program requirements and applicable procedures.

2. April 13, 1982 Filing by the Attorney General of the State of Illinois

The purpose of this correspondence from the Attorney General was to comment on the Commonwealth Edison presentation in Washington, D. C. on March 31, 1982, concerning drilling and coring activities at LaSalle. The following documents the results of our investigation into each concern expressed by Judith S. Goodie of the Illinois Attorney General's Office.

a. Section I, "Cored Holes For Pipe And Conduit Passage"

(1) Statement: "There are no specific analytical criteria for the locating of passageway [sic] holes."

Finding: Cored holes are located according to construction requirements. Prior to the performance of any coring work, Sargent & Lundy engineers assess the effects of the maximum number of reinforcing steel bars which could be cut. This is a conservative assessment, based on the diameter of the core and the spacing of the reinforcing steel in the area to be cored. The assessment is conservative because the maximum number of bars which are assumed to be cut is always equal to or exceeds the number which is actually cut or damaged. The CECo report of May 7, 1982 discusses the mechanism for locating cored holes on pages 7 through 15, and on page 3 of CECo's response, dated March 31, 1982, to the Attorney General's Petition. The NRC inspection on April 8, 1982 independently confirmed that the reported Sargent & Lundy mechanism for routing cored holes is being followed and that it constitutes an acceptable method to accomplish the work (See Attachment B).

(2) Statement: "No written analytical assessment or structural calculations were made of rebar damage in the drilling of holes either before or after the holes were drilled."

Finding: The CECo report of May 7, 1980, pages 29 through 34, discusses this concern. Drawings on which the specifics of any damaged or cut reinforcing steel were designated, were continuously maintained and updated by Sargent & Lundy as information of damaged steel was received from the field contractors. With this information, the Sargent & Lundy engineers made an engineering judgment to assess the acceptability of the proposed coring work or completed drilling work. This engineering judgment "consisted of a review of the

location of the damaged reinforcing steel in relation to the design stress levels in the reinforcing steel and the existing design margins in the concrete elements." (May 7, 1982 CECo report, page 30.)

It was the opinion of the NRC inspection team during the Sargent & Lundy inspection of April 8, 1982, that the use of engineering judgment for this type of evaluation was appropriate and constituted standard industry technique. This conclusion was based on discussions with the responsible Sargent & Lundy engineers and review of the engineering calculations which were performed to substantiate the validity of the engineering assessments in nine selected concrete elements. These structural calculations were performed by Sargent & Lundy in response to the petition by the Illinois Attorney General. Attachment B to this report documents NRC engineering acceptance of the selection basis for the nine areas.

Because of a concern raised at the presentation in Bethesda, Maryland on March 31, 1982, regarding the selection of the nine representative areas, Sargent & Lundy performed structural calculations on all structural elements in Unit 1 areas and in those Unit 2 areas required for Unit 1 operation. The results of this program identified no areas in which the design margins have been reduced below 1.0. This substantiates the validity of the engineering judgments used throughout the LaSalle project.

(3) Statement. "No reporting requirement has been identified for rebar damage in the drilling of passageway holes at any time from 1976 to the present. It is not clear from the information provided on March 31, 1982 whether such reports were in fact made on a regular basis."

Finding: As discussed in Paragraph 2.a.(1) of this section and in Section 2 of Inspection Report No. 82-21, cored holes are of two types, office routed and field routed. In both instances, the core location is approved by Sargent & Lundy prior to the work being performed and recorded on either the mechanical or structural design drawings.

The site contractors are not required to report reinforcing steel which has been cut or damaged during the coring operation. This is because a conservative structural engineering assessment was performed prior to performance of the coring work. The CECo report of May 7, 1982 discusses this system on pages 7 through 15.

(4) Statement: "It is unclear, therefore, how specific instances of rebar damage in passageway holes could have been reported on the Rebar Hit Schedule (RHS) drawings submitted on March 31, 1982 as Exhibit 3A. Nor is there any information as to how many steel bars were discounted for each passageway hole."

Finding: The CECo and Sargent & Lundy program does not require that specific instances of reinforcing steel damage, due to coring operations, be recorded on Rebar Hit Schedule drawings. Just as cored holes are incorporated into the structural and mechanical design drawings, drilled holes for concrete expansion anchor installation are incorporated into Rebar Hit Schedule drawings. To reiterate, reinforcing steel which is damaged or cut during coring operations is not recorded on Rebar Hit Schedule drawings, but on the structural and mechanical design drawings.

The conservative estimate of the number of reinforcing steel bars which have potential to be cut during coring operations is based on the diameter of the core and the spacing of the reinforcing steel in the area to be cored. The number of bars to be "discounted" in any one area is a function of these two variables. Each new core area presents a new set of variables which must be individually assessed for each core. While each core hole is similar with respect to the variables involved, each is unique with respect to the number of reinforcing steel bars which will be cut or damaged. In order to perform the required engineering assessment for each cored hole which is noted on the structural and mechanical design drawings, the Sargent & Lundy engineer must have performed an estimate of the number of bars likely to be cut or damaged.

The CECo report of May 7, 1982, Table 2.4-1 and 2.5-1, documents the conservatively estimated total number of damaged bars in all Unit 1 safety related areas and in those Unit 2 safety related areas required for Unit 1 operation.

(5) Statement: "Two examples have been provided of instructional notes on individual design drawings where engineering judgment had determined that reinforcing steel should not be cut. In each case, the use of a metal detector was required. However, neither note expressly prohibited the cutting of rebar. Furthermore, there is no evidence of field verification of compliance with the instructional notes."

Finding: The two examples cited refer to areas where the Sargent & Lundy engineering assessment had "determined that [the use of metal detectors] was required to minimize the cutting or damaging [of] reinforcing steel during the installation of cored holes..." (CECo Report of May 7, 1982, page 11). It was not the intent of the referenced drawing notes to explicitly prohibit the cutting or damaging of reinforcing steel, but to minimize it to the extent possible.

Therefore, field verification of the requirement to use the metal detector was not mandatory because of the inherent conservatism in the original approval of each core. The use of metal detectors further increased the conservatism in the approval of each core.

(6) Statement: "Such notes were added to the drawings only after the engineers became concerned that a particular element could not tolerate many more damaged rebars. Implicit in this procedure is the assumption that previous passageway coring had in fact caused some significant amount of rebar damage. This is entirely consistent with the statement in the Garrison affidavit that we 'seldom failed to contact rebar' in drilling the larger diameter holes."

Finding: The intent of the referenced notes was to minimize the cutting or damaging of reinforcing steel during the installation of cored holes, not to explicitly prohibit the cutting or damaging of reinforcing steel. Any reinforcing steel that was damaged was documented and analyzed.

(7) Statement: "The two specific instances of severe rebar damage cited in the Garrison affidavit are acknowledged by Edison to have occurred. No explanation has been offered for the nonconformances, nor has any estimate been provided of the frequency of such occurrences."

Finding: The specific background and resolution of the two instances referred to can be found in Section 1 of this report. To reiterate, the core holes referred to by Mr. Garrison do not constitute a nonconforming condition. A conservative estimate of the number of cored holes in Unit 1 safety related areas and in those Unit 2 safety related areas required for Unit 1 operation can be found in the CECo report of May 7, 1982, Tables 2.4-1 and 2.5-1.

b. Section II, "Drilled Holes For Expansion Anchor Bolts"

(1) Statement: "From December 1976 until July 1979 the cutting of rebar was allowed in non-critical areas of safety related structures without restriction. Revisions 0, 1, 2. It is not known whether the impact of seismic events or loss of coolant accidents was factored into the definition of non-critical areas. Nor are there Quality Assurance specifications for the identification of critical and non-critical areas in the field."

Finding: A non-critical area is defined as an area in which the use of a metal detector was not required and reinforcing steel was permitted to be cut. Further, the areas were defined as those in which the reinforcing steel was not required for the structural integrity of the concrete element under the design loads. This included all normal operating, accident, and severe and extreme environmental conditions, including Loss of Coolant Accident and Safe Shutdown Earthquake. Therefore, cutting or damaging reinforcing steel in one of these defined areas was of no consequence from an engineering design standpoint.

As discussed, even though damage to reinforcing steel due to drilling in non-critical areas was not procedurally required to be reported for the period December 1976 to July 1979, verification has been made that the site contractors did, in fact, report reinforcing steel damage, regardless of the area in which it had occurred. This information was then incorporated into the Rebar Hit Schedule drawings. Therefore, the Rebar Hit Schedule drawings represent the total record of reinforcing steel damaged or cut at LaSalle due to drilling.

Table 38-2 of Sargent & Lundy Form LS-CEA, Revisions 1, 2 clearly specifies those areas in the field which require the use of a metal detector; hence, critical and non-critical areas are defined. Revision 0 of Form LS-CEA required the use of tungsten carbide tipped drill bits by all site contractors. It has been established that tungsten carbide drill bits are incapable of inflicting detrimental damage to reinforcing steel. (See Attachment D)

(2) Statement: "No reporting requirements existed for rebar cutting in non-critical areas of safety related structures from December 1976 to July 1979. Revisions 0, 1, 2, Table 38-2. It is therefore unclear how all rebar damage at the site could be verified on Exhibit 3A."

Finding: Resolution of this item can be found in Paragraph 2.b.(1) of this section.

(3) Statement: "From July 1979 to the present, one rebar cut for each four-hole plate was permitted in non-critical areas of safety related structures without prior approval. See Article 3.2.9d of Revision 3, 4, 5, 6, 7, 8."

Finding: The NRC finds nothing unacceptable with Sargent & Lundy Form LS-CEA, Revisions 3, 4, 5, 6, 7, and 8, Article 3.2.9d. Based on engineering design considerations, Sargent & Lundy has developed and presented an acceptable specification (LS-CEA) to control drilling activities for the installation of concrete expansion anchors.

It should be noted that, although Form LS-CEA Revisions 3 through 8 and engineering prudence do not require that damage to reinforcing steel due to drilling in the referenced situation be reported, it was the practice of site contractors to report all reinforcing steel damage. Consequently, the Sargent & Lundy Rebar Hit Schedule drawings represent the record of reinforcing steel damaged or cut at LaSalle due to drilling.

(4) Statement: "Reporting forms for rebar damage in drilled holes do not appear in the Specifications until July 1979. Revision 3. No information is given as to specific reporting procedures, even where reporting was required from December 1976 to July 1979. It is known that contractors were not required to distinguish between nicked steel and cut steel until July 1979. Verification that damaged rebar had been reported was not included in the Quality Assurance specifications until July 1979. Revision 3, Article 1.5.2g."

Finding: Our review of site contractors' quality procedures, which were implemented prior to LS-CEA Revision 3, has shown that each contractor was utilizing a form which each had implemented. See Paragraph 6 of this section and Inspection Report 82-21, Section 1. The reporting form, which was included for the first time as a part of Form LS-CEA Revision 3, was developed to provide a standardized form which all site contractors could use to report damaged or cut reinforcing steel due to drilling work.

The fact that the contractors were not required to distinguish between nicked and cut reinforcing steel until July 1979 adds conservatism to the Sargent & Lundy engineering analysis of damaged steel. This

conservatism stems from Sargent & Lundy's assumption, during the engineering analysis, that all steel was cut even if it was only nicked with a tungten carbide drill bit.

(5) Statement: "No provision is made in the reporting form for verifying that a metal detector was actually used, nor was such verification included in the Quality Assurance specifications. Revisions 0-8. Reference to reinforcing placement drawings was permitted, but not required, during the location of holes in critical areas of safety related structures, from 1979-1981. Revisions 3-8, Article 3.2.8. Such reference was not even recommended during 1976-1979. Revisions 0, 1, 2, Article 3.1, Table 38-2."

Finding: The use of a metal detector, as referenced by Form LS-CEA, Revisions 0-8, was specified by Sargent & Lundy in an effort to minimize the cutting or damaging of reinforcing steel. As discussed in Paragraphs 2.a.(5) and 2.a.(6), it was not the intent of the specification to explicitly prohibit the cutting or damaging of reinforcing steel in these instances, but to minimize its occurrence to the extent possible. It should be noted that all reinforcing steel damage, which occurred as a result of drilling, was reported to Sargent & Lundy by the site contractors for subsequent engineering assessment. Consequently, written verification of metal detector use on the reporting forms is not mandatory to assure a complete quality record of drilling activities at LaSalle.

c. Section III, "Cored Holes For Anchor Bolts"

Statement: "The drilling of cored holes for grouted anchor bolts began in July 1980. By this time a procedure had been instituted to notch the concrete elements to expose the reinforcing steel before drilling began. Even so, rebar damage was experienced and reported on RHS drawings. Rebar cuts due to cored anchor bolt holes were 'recently plotted on the drawings known as Exhibit 3A.'"

Finding: Cored holes for the installation of grouted anchor bolts, which partially penetrate concrete elements fall into two categories: (1) Mechanical and electrical equipment foundation anchor bolts; (2) Mechanical pipe support baseplate assembly anchor bolts.

The coring of holes for the installation of mechanical and electrical equipment foundation anchor bolts was planned prior to the work being performed. Based on the sample reviewed by the NRC inspector, in every instance a Sargent & Lundy engineer performed a conservative estimate of the

maximum number of reinforcing steel bars which could conceivably be cut or damaged. This conservative estimate was again based on the diameter of the proposed core hole and the spacing of the reinforcing steel. The location of all holes which were cored for the installation of mechanical and electrical equipment foundation anchor bolts were plotted on Cored Hole Schedule (CHS) drawings. Table 2.5-1 of the May 7, 1982 CECo report provides a summary of reinforcing steel damage due to cored holes for the installation of mechanical and electrical equipment foundation anchor bolts.

The coring of holes for the installation of mechanical pipe support baseplate assembly anchor bolts was controlled by Mechanical Drawing No. M-1100, Sheet 23. This drawing required the contractor to carefully notch the concrete to locate reinforcing steel to preclude any damage. We have verified that this note was interpreted by the responsible contractor to strictly prohibit any reinforcing steel damage. Our review has not indentified any instances in which reinforcing steel was damage during th coring of holes for the installation of mechanical pipe support baseplate assembly anchor bolts.

d. Section IV, "Scope of Data Presented"

Statement: "Edison's written Response refers to LaSalle Units 1 and 2, as do the Expansion Anchor Specifications. The 90 RHS drawings submitted to the NRC staff, however, relate to Unit 1 only. Some of the buildings at the LaSalle County Station house equipment for Units 1 and 2 jointly, for example the Auxiliary Building. It is unclear, therefore, how Exhibit 3A treats rebar damage in such buildings."

Finding: The NRC has identified that the site contractors' programs apply to all safety related drilling activities. This includes the auxiliary building.

The May 7, 1982 CECo report clearly makes reference throughout its text to "structural elements in all Unit 1 areas and in those Unit 2 areas required for Unit 1 operation." This consideration adequately addresses the concern regarding both Unit 2 and common plant areas which are required for Unit 1 operation.

3. May 3, 1982 Filing by the Attorney General of the State of Illinois

The purpose of this document was to submit information regarding additional allegations of reinforcing steel damage at LaSalle County Station. The Attorney General's transmittal contained specific quotes from the affidavits of three former LaSalle construction workers (see paragraph 1 of the filing) and made reference to one other additional affidavit. The Government Accountability Project of the Institute for Policy Studies

had previously supplied the four referenced affidavits to RIII. (see paragraph 2 of the filing).

Each affidavit was reviewed in detail to identify those concerns dealing with drilling and coring activities. The concerns which the four individuals expressed regarding subjects other than drilling and coring are addressed in separate sections of this report.

Three of the four affidavits contained information concerning drilling and coring work at LaSalle County Station. An excerpt from each affidavit (see paragraphs 3, 4, and 5 of the filing) which deals specifically with coring activities and the NRC findings with regard to each follows:

a. First Affidavit, dated April 21, 1982

Statement: "From personal observation I can confirm that several years ago around 1000 holes were core-drilled into the containment wall and the reactor vessel pedestal around the 694 foot elevation of Unit I at the LaSalle plant. Construction crews core-drilled right through the reinforcement bars... [W]hen I left they had not replaced or repaired the reinforcing bars they cut through ... I personally observed another example at the 761 foot elevation of Unit II. Construction crews had to install supports to hold up the control rod casings. Fitters from Reactor Controls, Inc. ('RCI') were core-drilling eight to ten inches down into the concrete floor, which I estimated was about 18 inches thick. The fitters were not taking the time to check for and detect the reinforcement bars, however. As a result, the fitters were hitting the bars. I saw the core bits pulling out chunks of steel from the floor reinforcement bars. Again, the supports were installed without replacing the reinforcement bars."

b. Second Affidavit, dated April 21, 1982

Statement: "[D]amage [due to coring activities] occurred in the pedestal that the reactor sits on ... Between the pedestal and the containment wall long tubes called downcomers come down from the drywell to release excess pressure. Several years ago the Nuclear Regulatory Commission required nuclear plants to install supports for the downcomers.... Walsh, the construction firm, installed the supports by boring holes into the primary containment wall and the pedestal itself on three different levels. Walsh drilled holes to install bolts on the plates that hold the supports. They did this about 500 times on the containment wall and 500 times on the pedestal. In the process, Walsh drilled holes up to three feet deep in the concrete....Walsh core-drilled right through the reinforcement bars (rebars) in the reactor

pedestal and containment wall concrete. I know these facts, because I personally observed the work. Further, last week I confirmed the number of holes with the guys who did the work."

c. Third Affidavit, dated April 21, 1982

Statement: "Probably the most serious construction deficiencies that I personally observed occurred during a February-March 14, 1980 stretch that I worked at LaSalle. The flaws involved the concrete in the containment wall and the reactor pedestal. We were helping to install supports for large tubes that came out of the suppression pool between the reactor pedestal and the containment wall. Chicago Bridge and Iron cut out stainless steel panels and then the concrete was core-drilled to install the supports. I personally saw holes at least a foot to twenty inches deep being drilled into the containment at the 710 foot elevation. In the process, many of the reinforcement bars were severed. I personally saw a half dozen rebars severed on each of two or three occasions during the first few days of core-drilling....The problem of shattered rebars is not limited to the pedestal. I saw rebars severed all over the plant during core-drilling."

d. Finding: The affidavits from all three individuals indicate that each are principally concerned with coring work which was performed within the primary containment. The coring work which was performed in the primary containment was necessitated by the installation of downcomer bracing, supports for the safety relief valve lines, and modifications to the KWU quencher system.

Because of the uniqueness of this work, CECo implemented a rigorous program of quality and engineering controls. The extent of the controls which were established by the program was commensurate with the importance of the work being performed. The coring work was of a critical nature because it was being performed inside the primary containment. This augmented inspection program consisted of additional Walsh quality control inspection and monitoring of inprocess work by Sargent and Lundy engineering. Additionally, the cores which were taken during the work were maintained at the site and several cores were examined by the NRC during this inspection. Further details regarding the specifics of the augmented inspection program and the results of its implementation are contained in the May 18, 1982 submittal by CECo. This office has reviewed the May 18, 1982 submittal and finds it to be accurate and acceptable.

Additionally, extensive inspection effort has been expended by this office to review this program and monitor the in process work within the primary containment. The results of

this effort can be found in IE Inspection Report Nos. 50-373/79-07, 79-08, 79-11, 79-12, 79-16, 79-18, 79-21, 79-29, 79-32, 79-34, 79-35, 79-36, 79-41, 80-09, 80-11, 80-13, 80-21, 80-23, 80-26, 80-29, 80-31, 80-35, 80-42, and 80-44. These inspection reports represent 342 hours of NRC Region III inspection effort relating to the installation of supports for the safety relief valve lines, downcomer bracing and the quencer system.

This office finds that the three individuals' statements are substantially correct in fact. But, this does not imply that the structural integrity of safety related structures at LaSalle has been compromised since our review has established that damage to reinforming steel was controlled and evaluated. Conversely, the information reinforces what we already understand to be true. This is evidenced by the favorable inspection results documented in this report, the referenced IE Inspection Reports, and our understanding of the CECo program and its implementation.

In response to the general comments made in paragraphs 6, 7, 8, and 9 of the Attorney General's submittal of May 3, 1982, we note the following:

(1) CECo has adequately addressed the quality and engineering controls which were implemented for all contractors to control both drilling and coring activities in all safety related structures at LaSalle. The CECo reports of May 7 and May 18, 1982 summarize these controls and the results of their implementation.

(2) During this inspection, inspectors independently verified the acceptability of the drilling and coring programs for each major contractor who performed safety related work at LaSalle. The results of this verification can be found throughout this report, Inspection Report No. 82-21 and the Inspection reports referenced in Paragraph 3.d.

4. <u>May 26, 1982 Letter to the Director of Nuclear Reactor Regulation from the Office of the Attorney General, State of Illinois</u>

The intent of this letter from the Attorney General's Office was "to comment on a few questions which [were] raised by Edison's Final Report of May 7, 1982 and which in the opinion of this office, remain to be addressed in the pending inquiry..." The following documents our response to each concern expressed by Judith S. Goodie of the Illinois Attorney General's Office.

a. <u>Statement (paragraph 1)</u>: "Edison's Final Report purports to address rebar damage in 'all structural elements in all Unit 1 areas and in those Unit 2 areas required for Unit 1 operation.' There is no indication on the record thus far that Edison or the NRC staff intends to investigate possible

damage to the integrity of Unit 2. It is obvious that Unit 1 has been reviewed first because of Edison's intention of keeping to its most recently revised startup schedule. However, we trust that the safety of Unit 2 will also be addressed before the NRC rules on our Section 2.206 request."

Response: Our inspection of the drilling and coring activities at LaSalle encompasses the work of all major site contractors in all safety-related structures (i.e. Unit 1, Unit 2, and common areas). This inspection effort consisted of procedure review, personnel interviews, observation of inprocess work, and review of quality records. Documentation of this effort is provided in this report and in Inspection Report Nos. 50-373/79-07, 79-08, 79-11, 79-12, 79-16, 79-18, 79-21, 79-29, 79-32, 79-34, 79-35, 79-36, 79-41, 80-09, 80-11, 80-13, 80-21, 80-23, 80-26, 80-29, 80-31, 80-35, 80-42, 80-44, and 82-21.

Further, because concrete drilling and coring work is still underway in Unit 2, the completion of the remaining work will be inspected as part of the NRC routine inspection program.

b. Statement (Paragraph 2): "Neither Edison nor Region III has addressed the question of how non-conformance reports were treated in the current investigation. At the hearing on March 31, 1982 Edison admitted that two incidents of rebar damage in non-conforming cored passageway holes, which were cited in Mr. Garrison's affidavit in our original petition, had in fact occurred. Yet to date Edison has not reported on:

(1) The procedures for reporting all non-conformances in cored passageway holes.

(2) The total number of non-conformance reports filed with respect to rebar damage in cored passageway holes.

(3) The manner, if any, in which non-conforming cored rebar damage was accounted for in the total assessment of rebar damage."

Response: The American National Standards Institute standard N45.2.10-1973, as endorsed by NRC Regulatory Guide 1.74, defines Nonconformance as "A deficiency in characteristic documentation, or procedure which renders the quality of an item unacceptable or indeterminate."

No instance was identified during this inspection in which the structural integrity or shielding capabilities of the LaSalle plant structures were rendered either unacceptable or indeterminate due to concrete drilling or coring work.

All cutting or damaging of reinforcing steel at LaSalle was either: (1) approved by Sargent & Lundy prior to the work being performed or (2) subsequently reported to Sargent & Lundy for engineering analysis. This means that cut or damaged reinforcing steel does not constitute a nonconforming condition when the designer (i.e. Sargent & Lundy) has taken into account the effect the damage or cut reinforcing steel has on the design.

c. Statement (Paragraph 3): "The only written control on rebar damage in cored passageway holes was the use of instructional notes on an unknown number of structural design drawings. Edison's Final Report gives two examples of such notes, which call for the use of metal detectors in two specific instances. A total of 971 cored passageway holes have been documented. Edison has not reported on:

(1) The total number of holes for which metal detectors were required in drawing notes.

(2) How many bars, if any, were assumed to have been damaged in the drilling of such holes.

(3) What, if any, verification procedures were employed by the contractors to ensure that metal detectors were in fact used, and that undesired rebar damage did not in fact occur."

Response: Detailed discussion concerning the use of metal detectors during both drilling and coring activities are found in Paragraphs 2.a.(5), 2.a.(6), 2.b.(5) and 6. of this section. Resolution of these concerns can be found there.

5. Personnel Interviews

Interviews with Walsh Construction Company, Morrison Company, and Reactor Controls, Inc. crafts personnel were conducted to assess their knowledge of their respective drilling/coring programs and discuss any specific problems which they may have encountered. Each was selected because of his knowledge of past and present drilling/coring practices and policies. Interview were held with the following personnel:

- Walsh Ironworker: Employed in 1974; performed drilling to install concrete expansion anchors in the Unit 1 and 2 reactor buildings since 1979.

- Walsh Laborer: Employed in 1976; performed coring work in the Unit 1 and 2 reactor buildings since 1979.

- Morrison Company Pipefitter: Employed in 1977; performed both drilling and coring work in the Unit 1 and 2 reactor buildings since 1979.

- Morrison Company Pipefitter: Employed in 1981; performed drilling for the installation of concrete expansion anchors in the Unit 1 and 2 reactor buildings since his employment.

- Reactor Controls, Inc., Pipefitter: Employed in 1976; performed drilling for the installation of concrete expansion anchors in the Unit 1 and 2 reactor buildings since 1979.

- Reactor Controls, Inc., Pipefitter: Originally employed by Morrison Company during the period 1976-1981; employed by RCI since 1981; performed drilling for the installation of concrete expansion anchors in the turbine and radwaste buildings from 1976-1981 and in the Unit 1 and 2 reactor buildings since 1981.

Each individual provided a written statement at the conclusion of the interview. In summary, it was a consensus of opinion that the drilling/coring work had and is being conducted in accordance with procedural requirements. Each individual was knowledgeable within the scope of his assigned responsibilities. Each stated that record was always made of any damage to reinforcing steel which occurred during drilling or coring activities in safety related structures.

In addition to these interviews, Inspection Report 82-21 documents the results of interviews with eight additional individuals representing H. P. Foley and CECO. The results of the interviews, as documented in Report 82-21, are consistent with the results of the interviews conducted during this inspection.

6. Procedure Review

The NRC inspectors examined procedures controlling the installation of concrete expansion anchors and for the core drilling of holes into or through concrete walls and slabs. Report No. 82-21 also documents the results of site contractor procedure reviews relative to drilling and coring activities. The contractors whose procedures were examined during this inspection were:

a. Morrison Construction Co. (MCCo). Mechanical and Piping Contractor

MCCo Procedure PC-42, Revision 0, dated March 1977 and entitled "Expansion Anchor Control Program for Nuclear Safety Related Work." is the MCCo procedure for the installation of concrete expansion anchors.

(1) Review of MCCo's records indicate that the first concrete expansion anchors were installed during the week ending March 25, 1977.

(2) Subsection 7.1 of the procedure establishes the use of a metal detector (R-meter) to determine reinforcement location prior to drilling.

(3) Subsection 8.1 requires that holes into concrete will be drilled with tungsten carbide drill bits.

(4) Subsection 15.1 requires that reinforcing contacts shall be reported to the engineering department by the quality control department on a weekly basis.

(5) The use of diamond carbide drill bits capable of cutting or penetrating reinforcing steel was authorized in September 1979 in Revision 2 of Procedure PC-42 and the first documented cut was in Report No. RT 1001 on September 26, 1979. The same revision requires documentation of damaged reinforcing on LS-CEA form 1.0 and forwarding of the form to Sargent & Lundy.

(6) Included with procedure PC-42 is Form PC 118A, "Checklist for Core Drilling" Line 8.1 requires check off of inspection of each core drilled hole for cut reinforcing Line 8.2 documents the hits. Line 8.3 checkoff that plates where hits or cuts occur are identified per PC-42 subsection 8.9.

b. Zack Company - HVAC Contactors

The Zack Company procedure for the installation of concrete expansion anchors is QCP 23, "Field Anchor Bolt Installation," Revision 0, dated October 13, 1977. Zack did not begin work on safety related system until June 1978.

(1) Subsection 4.7 of the procedure required the use of a metal detector to determine the location of concrete reinforcement prior to drilling holes in concrete.

(2) Subsection 4.9 requires that nicked or contacted reinforcement be documented on Form 1.0.

(3) Subsection 4.14 requires that cut reinforcing be documented on Form 1.0.

Form 1.0 is forwarded to the architect engineer through the owner.

(4) Zack checklist QCP-23A "Checklist for CEA Installation on line 12 has a checkoff for verifying that a metal detector was used. On line 15 there is a check off

for identifying reinforcing steel that was contacted during drilling.

(5) The Zack scope of work does not include core drilling into concrete. The largest bolt normally used is 3/4 inch. When a 3/4" bolt does not pass the torque pullout test the hole will be redrilled to one inch.

c. Mid-City Architectural Iron Company - Gallery Structural Steel Erection

This contractor's scope of work was limited to installation on nonsafety related platforms in the containment. However, they did install some concrete exparsion anchor to support structural steel. They did not perform core drilling.

(1) Section 6, "Expansion Anchor Program", of Mid-City's Quality Assurance Manual controlled the installation of expansion anchors in safety related concrete.

(2) Subsection 6.2.8 requires a metal detector be used and that if the metal detector indicates reinforcing the anchor assembly should be moved.

(3) Damage report Form 1.0, on Line 5, identifies damaged reinforcing steel and on Line 6 reports the depth of the damaged steel.

(4) The Safety Related CEA Installation Report, Form 2, on line 4, requires the identification of the metal detector used to locate reinforcing steel. On Line 5, the reinforcing steel damage is reported.

d. Reactor Controls, Inc. (RCI) - Reactor Vessel Components and Associated Equipments Installation

The RCI procedure for installing concrete expansion anchors and core drilling is CFIP-1, "Concrete Fastener Installation Procedure." The procedure effective date is July 14, 1980. RCI began installing fasteners in April 1981. The procedure follows the requirements of LS-CEA for metal detectors. It requires documentation and reporting #6 reinforcing steel contacts on Form 1.0.

The only area where RCI performs core drilling is for the control rod drive HCU frame supports at elevation 761. The inspector reviewed location drawings for the holes and the RCI QC data sheets indicating whether reinforcing was damaged, the size of the bar and its depth in the slab. In each case reviewed, the damaged steel was reported to the owner and Sargent & Lundy. No bar larger than #6 was indicated to have been contacted.

e. HP Foley Company - Electrical Contractors

The Foley procedure for controlling the installation of concrete expansion anchor and core drilling is HPFCo - WI - 601, "Concrete Anchor Installation." Revision 0 was issued December 7, 1976 prior to the commencement of drilling.

(1) Subsection 3.4 requires that metal detectors be used when drilling in the vicinity of reinforcement.

(2) Subsection 3.5 requires that the drill crew check each drilled hole for damaged reinforcing steel before installing the expansion anchor.

(3) Subsection 3.6 requires examination of cores from cored holes to determine whether any steel had been cut.

(4) Subsection 3.7 instructs the Foley Project Engineer to forward the location of the cut steel to the owner.

(5) The Daily Concrete Anchor Installation Report HPFCo - 016 requires on Line 5 an indication whether a metal detector was used. On Line 6, the metal detector serial number is recorded. Damaged concrete reinforcement is documented on Line 7.

f. Walsh Construction Company - Civil Contractor

Generally, Walsh did not install concrete expansion anchors and core drills were performed only as a result of design changes or when shown on a construction drawing. Drilling is controlled by QCP-16, "Expansion Anchor Installation," Revision 0, dated January 7, 1977.

(1) Subsection 5.8 requires use of metal detectors for holes larger than 1/4 inch.

(2) Form QCP-16A documents use of the metal detector and whether rebar was damaged during drilling.

7. Observation of Work

The inspector examined twenty-nine cores from the suppression pool area. The cores evidenced good aggregate distribution and consolidation. The findings were then compared with reports submitted to S&L. No discrepancies were noted.

In addition, one core drill in progress on the lower level of the auxiliary building was inspected. The reinforcing steel in the area had been exposed to minimize contact. One #18 bar was to be partially cut. Prior to cutting, the installation

was inspected by the contractor's field engineer and quality assurance, CECo field engineering and quality assurance, and the Sargent & Lundy site liaison field engineer.

Conclusion:

Based on the results of our inspection, we have concluded that (1) adequate procedures to control concrete drilling and coring are and have been in place at LaSalle, (2) these procedures are being successfully implemented, (3) the engineering disposition of damaged reinforcing steel by Sargent & Lundy was proper and complete, and (4) the completed drilling and coring represents no compromise to the structural integrity of the LaSalle plant structures.

This conclusion holds true for all site contractors throughout the time of their coring and drilling activities and is based on the inspection results presented in Report No. 82-21, Paragraphs 1. through 7. of this section, and Attachments B and C to this report.

SECTION II

Prepared by: J. F. Norton John F. Norton

J. R. Kniceley J R Kniceley

2. Concrete Discontinuities Were Observed in the Reactor Pedestals

Allegation: During the installation of baseplate assemblies in the Unit 1 and 2 wetwell to support bracing for safety relief valve and downcomer piping, discontinuities were discovered in the concrete of both reactor pedestals.

Finding: The Mark II Boiling Water Reactor (BWR) concrete containment consists of a primary and a secondary containment. The primary containment has a steel dome head and post tensioned concrete wall supported on a seven foot thick reinforced concrete basemat. The inner surface of the containment is lined with steel plate which serves as a leaktight membrane. The primary purpose of the secondary containment, which encloses the primary containment, is to confine any airborne radioactive materials which may potentially leak from the primary system, and provide a means for controlled elevated release to the atmosphere.

The primary containment is divided into the upper drywell area and the wetwell (sometimes referred to as the suppression chamber) area, which extends from the containment basemat to the heavily reinforced concrete slab separating the drywell/wetwell areas, a height of approximately 60 feet. The Wetwell is cylindrical in plan view, with an internal radius of 43 ft., 4 in.

The safety relief valve (SRV) and downcomer piping penetrate the drywell floor and outlet near the bottom of the wetwell. In the 1970s, an event at a European BWR facility precipitated a design review which eventually led to the adapting of structural modifications to the SRV and downcomer piping assemblies in BWR suppression chambers. The accident was caused by an SRV jamming open. This generated resonance not only in the pipe outletting steam, but in other piping and structural elements in the containment. The modifications included, among other things, adding lateral support bracing members, some of which anchor on the pedestal wall and others on the containment wall. Also, "T" quenchers were installed on the terminal ends of the SRV lines.

The reactor pedestal is a cylindrical reinforced concrete annulus centered in the containment. The outside diameter approximates 30 feet and the annular wall thickness is 4 ft. 10 in. Total height of the pedestal is approximately 82 ft. from the wetwell floor to the base of the reactor. All concrete surfaces in the wetwell are lined with 1/4 in. stainless steel.

The reactor pedestals for the two units were each constructed with four symmetrical 10 ft. 4 in. circular openings penetrating the annular walls in the lower inundation area. The openings were designed to

effect external and internal water mixing. The centerline elevation of the penetrations is 683 ft. 6 in., which is about 10 ft. above the wetwell floors. Design pool surface is about elevation 700. According to licensee records, Unit 1 pedestal concrete was placed in six pours from July 24, 1975 to October 30, 1975. Unit 2 concrete placement was in five pours from October 23, 1975 to February, 1976.

The design drawings for the pedestals were issued September 5, 1974. All loadings were considered in the design except dynamic pool loads, which were not known at the time.

Two major structural revisions were subsequently issued for the pedestals. On June 13, 1975 modifications were issued which provided for additional reinforcing steel in the pedestal walls and also called for reinforcement bars in the penetrations. When the pedestal concrete was placed, the outside face grid of reinforcement steel was continued across the penetrations. The grids consist of No. 11 bars (1.41 in. diameter) on approximately 6 in. centers vertically and horizontally. The bars near the inside pedestal face were stubbed in to the penetrations to provide the option of cadwelding bars across. The design rationale was that if it was decided to leave the penetrations open, the bars could be cut off. In the event it was decided to close the penetrations, cadwelding in additional bars would be accomplished to provide an integral inner grid across the penetrations. When the pedestal concrete was placed, the steel liner had been set on the outside face, and was poured against (used as a form). The liner was continuous across the outside face of the penetrations. However, the inside pedestal penetration face steel liner was absent. Thus, access was provided to enter the penetrations from inside the pedestal. Screenwire was placed around the pedestal/penetration interfaces to contain the concrete when the pedestals were poured.

On September 8, 1976, additional pedestal modifications were issued which provided for placing concrete in the pedestal cores and penetrations to elevation 699 ft. 10 in., which is the design pool level. Horizontal and vertical No. 11 bars were then cadwelded into the inner face pedestal steel across the penetrations. Licensee records indicate that in Unit 1 the concrete was placed in ten pours from January 24 to July 13, 1977. In Unit 2, ten placements were made from February 24 to August 29, 1977. The pour lifts were each approximately 2 ft. 8 in. (26 ft. 5 in. total height). The pedestal concrete is BA 45 mix (f'c = 4500 psi). The internal concrete is BA 40 mix. The inner core concrete contains one horizontal grid of reinforcing steel in each lift.

On December 4, 1979, revisions were issued to install 34 bracing baseplate assemblies, 24 for the SRV lines and 10 for the downcomer lines on each pedestal.

The baseplates were to provide bearing and anchor for lateral support members. Installation of the baseplates required cutting out sections of the 1/4 in. stainless steel pedestal liner. When liner cutouts were removed at the abandoned penetration faces, concrete discontinuities

were apparent. Licensee records indicate the concrete deficiencies were discovered March 6, 1980 in Unit 1 and May 21, 1980 in Unit 2. These discontinuities were observed by construction laborers and were brought forth in allegations of construction deficiencies.

In the course of addressing the concrete deficiencies, the Region III inspector reviewed licensee documentation relevant to the problem; examined pertinent NRC inspection reports; interviewed construction and licensee personnel; and reviewed technical literature. Documentation reviewed included the following.

a. Nonconformance Report (NCR) No. 405, in which the licensee identified and described the concrete deficiencies in the Unit 1 pedestal on March 6, 1980.

b. NCR No. 197, identifying and describing Unit 2 pedestal discontinuities, dated May 21, 1980.

c. Procedures detailing the method of repair. One procedure was issued with NCR No. 405, and Revision No. 1 was issued May 19, 1980. Revision No. 1 also embraced Walsh Grouting Procedure for SRV and downcomer base plates, dated March 10, 1980.

d. Engineering drawings of repair details.

e. Detailed pre-repair photographs of the concrete discontinuities.

f. Specifications and physical data of the repair material (EMBECO 636 Grout)

g. Licensee quality control documentation recorded during the repair process.

Three distinct types of discontinuities were evidenced in the photographs and other documentation of the Unit 1 pedestal. These were top voids, honeycombing and screenwire displacement.

Top voids, at the upper levels of the penetrations, were triangular in section along the direction of the penetration. The largest had vertical dimensions of 1 ft. 4 in. at the outer pedestal liner plate and feathered out horizontally at 3 ft. 1 in. In sections transverse to the penetrations, the voids were contained in approximately the top quadrants of the penetration peripheries at the outer liner plate. The voids were caused by failure to displace entrapped air and effectively consolidate the concrete. Accomplishing this was complicated by the 4 ft. 10 in. pedestal wall thickness and by the inner and outer face grids of No. 11 reinforcement bars (with cadweld sleeves on the inner grid bars) on about 6 in. vertical and horizontal centers.

Honeycombing occurred in two of the Unit 1 penetrations near the outside penetration face centers. It was contained between the outside face reinforcing grid and the steel liner plate. The honeycombing

was caused by inadequate concrete consolidation. One penetration had about 4.5 square feet of honeycomb area at the liner plate face and the other evidenced about 2.0 square feet. Maximum depth of the cavities perpendicular to the liner plate was about 5 in. Average depth was approximately 3 in.

As mentioned previously, screenwire was used at the pedestal/penetration interfaces to contain the concrete when the reactor pedestals were poured. The screenwire was not secured adequately at the outside periphery of the penetrations in Unit 1. This resulted in wadded up ridges of screen displacing concrete during the placements. The screenwire discontinuities ranged up to about 1 ft. wide along the liner plate face and 6 in. deep perpendicular to the liner plate around the circumference of the penetrations.

In Unit 2, top voids were also discovered. They were generally as those previously described for Unit 1. The largest void had vertical dimensions of 1 ft. 8 in. at the outer liner plate and wedged out at 4 ft. 2 in. along the top of the penetration. Again, in sections transverse to the penetrations, the voids were contained in approximately the top quadrants of the penetration peripheries at the liner plates. No honeycombing or screenwire displacement discontinuities were found in the Unit 2 penetration faces.

In the repair process, the outer pedestal liner plate was removed to expose the concrete discontinuities. Chipping hammers were then applied to remove unsound concrete and wadded screen wire. The areas were then flushed and cleaned by water blasting. After preparation, the areas were grouted back to design configuration with EMBECO 636 grout. This was done in accordance with engineered repair procedures and QC inspection and documentation.

Three other concerns relating to the pedestal concrete discontinuities were also expressed in the allegations. The first of these was a concern that all of the concrete deficiencies may not have been located. A review of design/construction chronology and construction methods makes clear how the discontinuities occurred. No deficiencies in the annular pedestal concrete were evident except the screenwire displacement in Unit 1 previously described. This was generally contained between the outer face reinforcing bars placed across the penetrations and the inside face of the 1/4" thick stainless steel liner on the outside circumference of the pedestal. The areas averaged approximately 3 in. deep transverse to the liner plate at the edge of the 10 ft. 4 in. diameter penetrations, then, feathered out at about 10 to 12 inches in a direction perpendicular to the circumference lines of the penetrations.

The top voids (in all 8 penetrations) and the honeycombing (in two of the Unit 1 penetrations) were all discontinuities in the mass concrete placed in the internal pedestal and penetration areas. This situation was observed by an NRC inspector and the following is excerpted from NRC Inspection Report 80-11 and addresses the pedestal concrete quality in Unit 1:

"Suppression Pool Modifications

The inspector toured the Unit 1 suppression pool and observed the work in progress. The contractor was removing the pool liner as required by the revised design. The concrete under the liner plate appeared solid, well compacted and no evidence of honeycomb was visible in the areas observed. Construction drawings specify the removal of 54 sections of liner plate in the pool wall and 31 from the reactor pedestal. Approximately 15 percent of the liner plate had been removed."

In the installation of 24 baseplate supports for the SRV lines and downcomers, a total of 192 holes were cored in the Unit 1 reactor pedestal. All of the cores were logged and retained by the licensee. The Region III inspector examined 29 of these cores. All cores examined evidenced good aggregate distribution, matrix qualities and consolidation characteristics.

When the pedestal liner plates were removed for installation of baseplate bracing supports and the concrete deficiencies discovered, additional liner plate was removed as required to trace the unsound concrete in the penetration faces. There is no evidence to substantiate that additional deficient concrete exists where plating has not been removed.

The second concern voiced was that in the repair process, reinforcing steel was heavily damaged or cut through by jackhammers when the unsound concrete was removed. Subsections 3, 4 and 5 of the repair procedures specified the following relative to reinforcing bar damage:

- Inspect reinforcing steel to assure that it is not damaged.
- Reinforcing with dents 1/8" in depth or less shall have the dents ground smooth.
- Reinforcing with dents deeper than 1/8" in depth shall be replaced by cadwelding in a new bar if so deemed necessary by the engineer (Sargent & Lundy).

Quality control documentation recorded during the repair process indicated the procedural items were inspected for compliance.

Also, the chipping hammers used were relatively lightweight models which are designed to be handled in overhead and lateral positions without undue operator fatigue. These hammers are capable of chipping and dressing concrete, but do not have adequate inertia to significantly damage reinforcing bars when operated with average care to avoid such damage.

The third concern was that concrete repaired with grout (or by other repair methods) is not as strong as the original monolithic structural element would have been if no deficiencies had existed. Extensive literature addressing research and testing by the concrete industry

and by major constructors such as the U.S. Army Corps of Engineers, the Bureau of Reclamation and other engineering entities involved in maintenance and repair of concrete and concrete structures has been published. Generally, these data conclusively substantiate that when unsound concrete is properly repaired in accordance with carefully engineered approved practices, original design structural integrity can be maintained, and even enhanced in many cases. This has also been substantiated in the experience of the Nuclear Regulatory Commission.

Conclusion: Concrete discontinuities existed in the Reactor Pedestals of both units as alleged. The deficiencies were identified by the licensee in nonconformance reports. The NCRs were properly dispositioned and repairs were engineered by Sargent & Lundy. The discontinuities were repaired under CECo QC surveillance which is documented. No additional followup action is planned.

3. 55-Gallon Drum Embedded in the Unit 1 Basemat Slab

Allegation: A 55-gallon drum was encased in the concrete slab under the Unit 1 containment.

Finding: On June 9, 1982, the Region III inspector interviewed the alleger in an attempt to obtain specific data relative to the location of the alleged embedded barrel in the basemat of Unit 1. The individual stated that the barrel is in Unit 1, and that it was buried in concrete which was placed in July or August of 1975. He also provided the identity of the Walsh Construction Company foreman responsible for pre-pour cleanup for the particular placement.

The individual related the following story: He was working as a cleanup laborer on the 5:30 a.m. shift. There were two shifts of reinforcement steel placers at the time. Reinforcement steel had been placed and tied proximate to the 55-gallon drum, which was laying on its side. To remove th barrel would have required either cutting it in pieces with a torch or removing the bars. There was pressure to expedite the cleanup operations because concrete placement was scheduled to begin about 9:00 a.m. The Walsh foreman came on duty about 7:30 a.m. The individual said he approached the foreman and told him about the drum. The foreman then left the immediate area for about 15 minutes, then returned. He then stated, "You didn't see a thing. We don't have time to remove it before the concrete gets here." The individual said he then left the area, but feels sure the 55-gallon drum was encased in the pouring operations.

The individual started working on pre-pour cleanup in January of 1975. The Region III inspector examined Walsh Construction records for the Unit 1 basemat pours. The Unit 1 basemat was placed in two pours, the south half by pour Nos. 1R4A and 1R4B placed October 29, 1974 and the north half by pour Nos. 1R4C and 1R4B placed November 17, 1974. Also, an NRC inspection report was located which addressed the south half pours which is excerpted in the following:

"Observation of Concrete Work Performance

The inspector observed concrete placement (pours No. 1R4A and No. 1R4B) for the south half of the Unit 1 containment basemat, and the following were determined:

(a) Forms appeared to be tight, strong, and clean.

(b) Rebars appeared to be spaced properly.

(c) All items on Walsh Pour Checkout Card, Form No. QCP-9A, were signed off as acceptable.

(d) Slump and air content tests and temperature (ticket No. 6579) were determined to be within the range as specified. The label on air content test equipment (I.D. B7786) indicated calibration was current.

(e) Four (4) placement crews were performing the work. Each crew consisted of nine (9) persons, and three (3) to four (4) vibrators were used."

It was also observed in examining the Pour Checkout Cards (QCP-9A) for the basemat placements that the individual identified by the alleger was not the Walsh Foreman who signed off for pre-pour clean-up inspection.

A visit with the individual was again held on June 11, 1982 and the aforementioned data reviewed. When faced with the conclusive data that the basemat was not placed during his employment, he then indicated it may have been in the drywell floor of Unit 1. Subsequent examination of construction records regarding this pour revealed it was placed March 3, 1976. Again, the foreman identified by the individual was not the foreman who verified prepour clean-up. Further checking revealed that during July and August of 1975, the Unit 1 pours were in the containment walls from about elevation 681 to 717. The individual had indicated he remembered for sure it was a concrete slab with at least a 20 foot span. The containment walls are four feet thick below elevation 728'-8".

Walsh records indicate employment of the foreman identified by the individual began in February 1975 at LaSalle County Station. On November 11, 1975, he was promoted to General Foreman. He held this position until June 16, 1976 when he left the site. He is presently employed at the Braidwood, Illinois Nuclear Construction site.

On June 11, 1982, he was interviewed by Messrs. James Kniceley and John Norton of the Region III NRC office. During the course of the interview, the individual's story was related to the foreman. The two following paragraphs are excerpted from the sworn statement obtained from him:

"I was in charge of the clean-up crews to clean up before concrete pours. Mr. Dean Plese was my Superintendent. I worked on Unit 2 and I began work at the 710 elevation level. We cleaned up before wall pours. My job was to see that everything was cleaned up before concrete could be poured. I never worked on Unit 1.

Mr. Plese and I personally witnessed clean-up before we signed off that the area was ready. I do not recall anyone telling me that there was a 55-gallon drum left in the area before a pour. We cleaned everything and I am not aware of any debris left behind. Mr. Plese was very strict and did his job well. He was always walking around checking up on things. If anyone told you of any debris or 55-gallon drum in any area that I worked, they are lying because it is not true."

On June 14, 1982, the Region III inspector again examined Walsh pour records. The records indicated that the foreman identified by the individual signed off for 54 pre-pour clean-ups on Unit 2 and none for Unit 1.

Although it is not believed that a barrel is encased in Unit 1 concrete, the licensee was requested by the NRC to evaluate the effects of such a barrel in the Unit 1 basemat and drywell floor concrete. Design engineers Sargent & Lundy reviewed structural drawings. Their study concluded that reinforcing steel spacing in the drywell floor precludes the space required for a 24 in. diameter by 36 in. 55 gallon drum. Furthermore, the basemat also contains a relatively small percentage of area that could accomodate the drum due to reinforcing steel density. An engineering assessment was made by Sargent & Lundy for these areas postulating a void with the dimensions of a 55 gallon drum. Loading combinations considered included dead load, live load, pressure load, safe shutdown earthquake load, temperature load and pool dynamic loads. The results of the assessment indicate that the stresses, considering a void created by the drum, are less than the allowables committed to in the LaSalle FSAR.

Conclusion: No credible evidence could be developed to substantiate that a 55-gallon drum is embedded in Unit 1 concrete.

4. Debris in Concrete

Allegation: Cleanup activities prior to concrete pours were inadequate and debris was left in concrete. The allegations indicated the debris included paper cups, soda pop cans, beer cans and 2x4 boards.

Finding: The inspector reviewed pertinent NRC inspection reports, interviewed licensee and contractor personnel, toured the plant area with an alleger, reviewed selected QC and construction records of various concrete placements, and reviewed all nonconformance reports (NRCs) of structural concrete repairs. Also a report addressing Unit 1 reactor containment concrete integrity prepared by the licensee in late 1981 was reviewed.

On May 27, 1982, an alleger took the inspector on a plant tour. During this tour, an area of Unit 1 containment at about the 750 ft. elevation near the outboard main steam valves was pointed out and alleged to contain an embedded 2x4 board.

Subsequently, Walsh Construction concrete records were reviewed for two pours which contained the area identified - Pour No. 2RCW9B placed September 23, 1976 and 2RCW 10B placed October 4, 1976. The dates of these placements were then compared with the alleger's labor record cards. It was found that on September 23, 1976, the alleger was chipping concrete in the auxiliary building. On October 4, 1976 the alleger was not on site.

A selected sample of 75 Pour Checkout Cards (QCP-9A) was examined. The sampling included placements in the Reactor Vessel Pedestals, Containment Walls, Reactor Building columns and interior walls, columns, walls and floor slabs in the Auxiliary Building, and the Unit 2 RHR pump foundation. Cleanup was signed off as inspected and approved by the contractor and by CECo on all cards. In examining the pour checkout card for the RHR foundation pour, it was noted CECo reminded the Walsh General Foreman to watch cleanup better.

Nonconformance reports covering repairs for all structural concrete were reviewed. A total of 44 NCRs were written, nine of which were relevant to the debris allegations. Of these nine, two involved form spreader blocks inadvertently left in place, and one involved styrafoam blocking inadvertently left in place, when the pours were made. Therefore, six NCRs were generated because of embedded debris. The un-removed spreaders and styrafoam blocking is not considered to be a debris problem.

A search was made for NRC inspection reports addressing structural concrete placement. A total of 15 reports addressing various aspects of concrete activities were examined. Of these, four reports were considered relevant to debris cleanup effectiveness.

The reports are discussed in the following:

a. Report Nos. 75-05/75-05, 75-06 and 77-13/77-12: The reports identify four pours which were inspected by NRC before and during placement. Specific pours addressed are 1RCW3C (Unit 1 containment), 1R9E, 1R10E, and 2RB19W8. Statements are recorded for all of these pours that the placement areas were inspected by NRC inspectors to assure no debris existed prior to placement. All areas were found acceptable.

b. Report No. 75-06: This report documents an NRC review of licensee audits related to concrete placement activities during the period of March to July 1975. The audits were stated to be comprehensive and complete.

c. Report No. 77-11/77-10: Region III inspectors wrote a noncompliance report on concrete pours 1RB21W2 and 2RB2W1. They identified voids attributed to the following, "This condition was caused by debris, i.e., paper, wood polyurethane material and scrap wire which were not cleaned out prior to concrete placement". The location was Reactor Building J-line wall to elevation 840 ft. and 15-line wall.

The concrete deficiencies were repaired and the noncompliance was closed out in Report No. 78-08/78-07 (See pages 3 and 4).

d. IE Report 82-11: This report details a comprehensive in-service concrete inspection embracing American Concrete Institute section 201 and surpassing it in some aspects. Placement Nos. 1R7A, 1RCW15A, 1RB19W and 1RB1W3 were evaluated. The inspection engulfs much more than the embedded debris aspect, but it is inclusive. The summarizing statement regarding the inspection follows:

> "It is the evaluators' conclusion that the results of the detailed inspection of these four concrete placements indicate an acceptable and functional level of concrete serviceability at LaSalle. This conclusion is enhanced by the favorable results of other concrete placement inspections in the reactor and auxiliary buildings which were conducted during this and previous civil inspections."

Also, a report was reviewed which the licensee prepared November 10, 1981, a portion of which addressed the structural integrity of Unit 1 containment concrete. The licensee accomplished a survey of the outside face (50% of the concrete surface area) of the containment wall. The survey showed that the concrete quality was excellent and no voids were detected.

The following observations are submitted based on the Region III inspector's personal experience accumulated through several years as a Civil Design/Construction Engineer involved with major concrete structure types such as locks, dams, spillways and other water resource type structures as well as nuclear construction.

Concrete, when placed in areas such as walls, columns and beams which have a relatively large heighth to width ratio tends to push low-density, lightweight debris laterally to the form faces. This is because normal fluid concrete weighs approximately 145 lbs/cu. ft. compared to a few lbs./cu. ft. for empty pop and beer cans, paper cups, sawdust, wood chips, and other light extraneous material. The first concrete in the placement is usually introduced toward the central area of the forms due to reinforcing steel grids near the form faces. Because of this phenomenon, any extraneous lightweight material is usually immediately apparent in the construction joint face when form stripping is accomplished. Any consistent or gross negligence in removing these types of extraneous materials would soon become evident.

The following was determined during the course of the inspection:

a. The individual who pointed out the area of the alleged embedded 2x4 was not at the pour area immediately prior to concrete placement. Therefore, it is not known whether or not a piece of timber he said he observed earlier was removed in the pre-pour cleanup.

b. A total of seven concrete repair jobs effected by the licensee were recorded. This included six NCR repairs and one repair associated with the NRC inspection finding. These seven repairs represent placement of 476,879 cu. yd. of structural concrete at the site.

c. A review of the nonconformance reports indicate that the deficiencies were identified by the licensee. The NCRs were properly dispositioned and repairs were engineered by Sargent & Lundy. The deficient areas were repaired under licensee QC surveillance which is documented. Also, the record review mentioned previously in this section indicated pour checkout cards were appropriately employed and completed.

d. Interviews held with the concrete superintendent, the Walsh Construction Quality Control Manager, a General Foreman responsible for pre-pour cleanup, a licensee Quality Control Inspector and others gave indications that pre-pour cleanup was properly accomplished.

Conclusion: Structural concrete deficiencies attributable to extraneous embedded materials existed as alleged. Some of the deficiencies were identified by the licensee in nonconformance reports (NCRs), and others were identified by Region III NRC inspectors. The resulting NCRs were properly dispositioned, and repairs for the deficient areas appropriately engineered by the design AE firm. Repairs were accomplished with licensee QC surveillance, which is documented. No evidence could be developed to indicate concrete debris exists which has not been identified. No additional followup action is planned.

SECTION III

Prepared by: R. E. Lipinski E. E. Norelius for R. E. L
I. N. Jackiw E. E. Norelius for I. N. J
J. R. Kniceley J. R. Kniceley

5. Improper Concrete Work in the Screenhouse

Allegation: When the concrete was poured for the screenhouse, it hardened faster than it was supposed to. Chloride was added to the concrete to speed up the process.

Finding: Chloride is used in industry in small amounts, not to exceed 2 percent, in the form of calcium chloride to accelerate hardening of concrete. When used in this proportion it has no detrimental effect on concrete. Strictly speaking, chloride cannot be completely eliminated from concrete because of its presence in nature in various forms of chemical compounds. It is present in water, in aggregate, in sand, etc.

Review of the FSAR revealed that only part of the screenhouse should be considered as a Category I (safety-related) structure. This is the part which is protecting the pipeline which carries the water for the core spray cooling system (CSCS) and should be designed and constructed as a safety related structure. Review of the FSAR also indicates that the design of the screenhouse did incorporate all the loads as appropriate for the Category I structure.

In order to resolve the allegation that chloride is present in the concrete the inspector performed the following:

a. Review of Records

Mix records for the following dates of pour and portions of the screenhouse were reviewed:

Date	Location
4/15/75	Valve Pit
4/10/75	D Line Wall
4/9/75	Valve Pit
2/13/75	Partial Wall
12/20/74	Screenhouse
3/3/75	Wall
3/1/75	Screenhouse Wall

These are the parts of the screenhouse which are considered to be safety related becuase they house the equipment specified above.

The records indicate that no calcium chloride was added to the concrete mix. Furthermore, examination of records of tests performed on samples from the above pours indicate that all samples exceeded the minimum specified compressive strength which is 3500 psi at 90 days.

b. Statement from Concrete Superintendent

The Concrete Superintendent provided a written statement indicating that there is no chloride used in the construction of the lake screenhouse. Instead, the concrete consists of a richer mixture of cement and aggregate (BA-40 or BA-45) which designates concrete of compressive strength of 4000 or 4500 psi respectively and this accounts for quicker setting. Review of the test cylinder records indicated that the compressive strength achieved by the samples exceeded far above the minimum specified strength of 3500 psi at 90 days, in most cases being above 5000 psi and in some cases exceeding 7000 psi.

c. Chemical Analysis of Cored Samples

The inspectors requested that three samples be cored from the area which is surrounding the CSCS pipe. These were subjected to a chemical analysis performed by Wiss, Janney, Elstner and Associates, Inc., an independent Consulting and Research Engineering Company. This consulting company was selected by the NRC inspectors.

Chemical analysis results received from the independent testing lab showed that for the three core samples, the chloride content was in the range of 0.04 - 0.05% by weight of concrete. Wiss, Janney, Elstner and Associates, Inc. concluded that the amount of chloride ion content which was found in the concrete samples does not indicate that calcium chloride was added as an accelerator to the original concrete mix.

Conclusion: This allegation regarding addition of chloride to concrete has not been substantiated. No further followup actions are planned.

6. Misaligned Containment Wall

Allegation: One section of the outer containment wall is off-whack by about four inches.

Finding: On May 27, 1982, a tour was conducted with an alleger who pointed out the location in the reactor building where the outer containment wall appeared to be misaligned with respect to a nearby RHR shutdown cooling equipment room structure.

The inspector reviewed design drawings and specifications and interviewed licensee personnel. Section 30-30 of drawing S-254 shows the

as-built condition showing the reactor building wall butting into the reactor containment corbel. To an uninformed individual, the open space at the wall-corbel interface might appear as bulging. The inspector verified that both the corbel and wall are constructed per the drawings.

Conclusions: It is concluded this outer containment wall was constructed in accordance with design.

7. Improper Masonry Wall Construction and Mortar Quality

Allegation: High density masonry walls were constructed with insufficient amounts of mortar between the blocks and in some cases no mortar was used. Also, in many cases the reinforcing wire was not installed between the blocks. In addition, the mortar used in the construction of the masonry walls was not properly proportioned. Too much sand was used in the mixture.

Finding: This inspection was directed towards quality control and workmanship. The inspection consisted of the following:

a. Interviews with the applicant's personnel responsible for QA/QC of the walls.

b. Interview with the workers who were employed on construction of the walls.

c. Review of construction records pertaining to quality of materials and workmanship.

d. Review of tests on cored samples taken from the walls as a result of issuance of IE Bulletin 80-11.

On June 12, 1982 an interview was conducted with an employee of Walsh Co. who was knowldegeable in masonry wall construction. He stated that he was employed by Walgren as a bricklayer from October 1978 until June 1979. From June 1979 he worked as a foreman for Walsh. In January 1979, he was promoted to the position of supervisor. His duties involved field inspection and documentation of quality control of material and construction. In that capacity he was responsible for quality of construction of safety related masonry walls. According to him, the drawings and specifications pertinent to the masonry concrete walls (CMU) were originated by Sargent and Lundy. All of the safety related walls were designated on the drawings.

He inspected mixing of mortar twice a day, usually in the morning and in the afternoon. Proportioning of the mix was 1 part of cement to 3 parts of sand by volume for a mixer of 12 cu ft. capacity. He stated that this procedure resulted in a uniform mix throughout the job and was observed for all masonry construction. The in-process

testing of mortar was for every 25,000 blocks of construction and the specified minimum compressive strength was 2500 psi. The walls were reinforced with horizontal truss type reinforcing, consisting of 3/16 inch diameter side rods and No. 9 cross ties, for every other course of the blocks. The reinforcing was manufactured by AA Wire Products Company, Chicago, Illinois. Testing of the mortar was conducted by A and H Engineering Company.

He stated that it would be practically impossible to leave out mortar from the middle of the wall without being noticed by the quality control personnel because each layer of mortar has a thickness of about 3/8 inch and after several layers a depression would be formed in the middle of the wall thus revealing lack of mortar. Furthermore, construction was continuously supervised by the foremen who were responsible for the quality of the workmanship.

When asked about quality of construction before and after the period of time when he was in charge of quality control inspection he stated that to the best of his knowledge he did not observe any change in the procedures.

On June 14, 1982, four former employees of Walgren, the contractor in charge of masonry walls at the LaSalle Nuclear Generating Station, volunteered to provide the inspectors with first hand knowledge of construction practices regarding safety related masonry walls. They were working as bricklayers in various parts of the plant and therefore, were familiar with the construction of the masonry walls. According to them, during the peak of construction, the workers were pressured to rush with the job to the point that a proper workmanship could not be exercised. As a result, in many cases, the amount of mortar placed between concrete blocks was insufficient or sometimes omitted altogether. Also, the reinforcing which was called for on the drawings was skipped. The personnel who were responsible for quality control were not always present during construction of the walls.

The interview was followed by a tour through the plant during which the aforementioned workers pointed out several masonry walls where to their knowledge proper workmanship was not applied.

In most cases, according to the former bricklayers, the deficiencies were in the upper part of the walls or in those sections where because of presence of other equipment, such as cable trays or piping, the walls were not easily accessible. During the tour, the former workers pointed out two walls in the Unit 1 reactor building where they believed deficiencies existed. Examination of similiar walls in Unit 2 revealed that in a relatively large pipe penetration, local small voids were present due to absence of mortar between blocks located inside of the walls. The walls thus examined were located at elevation 710'-0" between column lines G and J and on column lines 14 and 15.

In order to further verify how wide spread the dificiencies in workmanship were, the NRC inspectors requested that several blocks be removed from the Unit 1 wall thus forming an opening of approximately 3'-0" long, 20 inches wide and 12 inches deep exposing joints between several blocks. The inspectors verified that horizontal reinforcing was present as specified on the drawings and that the mortar was present between the blocks. Furthermore, cores were taken from two other walls. One was in the Unit 1 turbine building, in the wall at floor elevation 735'-0" on column line 13.8, about 5 feet east of line S. This wall was first identified by another alleger during a tour conducted on May 27, 1982. During that tour this wall was identified as one where mortar was omitted from blocks on the inside of the wall. This wall was constructed from high density blocks, and was 4'-7 5/8" thick and about 30 feet high. The core taken was 3'-8" long and 4" in diameter. No voids due to absence of mortar were found.

The other core was taken from the Unit 1 TIP room at floor elevation 740' 0" between column lines 15 and 16. This wall was also made of high density blocks and was 2'-11 5/8" thick. Again, no voids due to absence of mortar were found, and the inspectors noted that reinforcing was present between the blocks.

Based on the taking of cores from the walls and review of records there was no evidence that the walls were constructed in violation of the specifications or design requirements as stated on the drawings. The records of the material tested indicate that they are acceptable. The examination of the openings in the walls either by removal of the blocks or by coring indicate that reinforcing wire and mortar were present according to the design criteria and to normal engineering practice. In view of the above, the inspectors conclude that the structural integrity of the safety related masonry walls will not be adversely affected by the local deficiencies and that they will perform their design function.

The inspector further reviewed licensee records and noted that in response to IE Bulletin 80-11, the licensee identified the walls which are considered to be safety-related and described the design criteria used for analysis of these walls. In view of the ongoing review of these criteria against the corresponding criteria proposed by the NRC staff, a license condition was imposed in the LaSalle license. In a letter from Commonwealth Edison to A. Schwencer, dated April 24, 1981, the licensee stated that the QA/QC information for the safety related masonry walls is available at the LaSalle County site for review by the NRC staff.

The inspectors also reviewed the records pertaining to the construction of the concrete block walls for a period from February 23, 1979 through June 8, 1979. These records consisted of verification of quality control of mortar mix and contained such information as time of mortar mixing, how clean was the sand and water and penetrometer

readings to verify the initial set of the mortar. The other information contained in the records referred to placing of the blocks, i.e., quality of joints, joints construction and thickness and joint reinforcing.

In all cases, the records were found to be in proper order and there was no evidence of workmanship of unacceptable quality.

The inspector noted that as a result of the issuance of IE Bulletin 80-11, the status of some of the masonry walls had been changed from non-safety related to safety related. In view of the fact that these walls were originally constructed as non-safety related, there were few QA/QC records for the walls and the applicant was requested to evaluate their integrity. In response to this NRC inquiry, the licensee initiated a program to verify their acceptability. To accomplish this task, one sample was cored for every 5000 square feet of the wall and tested by A&H Engineering Company. The cores were taken in 3 1/4 inch diameters and then cut to the size of 2 inch by 2 inch cubes to be used as samples. The samples were then tested as per specifications. Ninety-six samples were taken altogether. This included 52 of combination of solid block and mortar cores and 44 of block units. Of the 96 samples, 11 failed on the first test. The failures were attributed to disturbing of mortar during preparation of the cubed samples. For the samples which failed, the tests were repeated with another sample taken from the area as close as practicable to the original sample (usually within 5 feet). All of the examined samples for which the records have been reviewed exceeded the minimum specified compressive strength of 2500 psi.

Conclusion: Based on the inspection findings above, the inspectors concluded that the masonry walls at LaSalle were constructed in accordance with design requirements.

SECTION IV

Prepared by: J. Creed J.R. Creed

J. Knideley J. Knideley

8. Inadequate Security

Allegation: Forced long work hours by security personnel have led to violations of the licensee's security plan.

Finding: Several members of the guard force were interviewed to obtain specific information. Four NRC inspectors were onsite during the period June 14-17, 1982. The licensee's general implementation of the security program was assessed, records were reviewed, and control points were observed. Some examples of violations of the licensee's security plan were identified. These were brought to the management's attention and corrective action was initiated. The details are considered to be Safeguards Information not subject to public disclosure.

Conclusion: It was found that some security personnel had worked many hours of overtime. The extent to which this may have contributed to the violations observed is not clear. The licensee was in the process of increasing the numbers of guards. The details of the findings regarding the security inspection will be reported in a separate report (Inspection Report 50-373/82-39.) While examples of violations occurred, we concluded that the security system at LaSalle provides an overall adequate level of protection.

SECTION V

Prepared by: J. Neisler C E Norelius for JN

J. R. Kniceley J R Kniceley

9. Improper Vibration of Concrete

Allegation: In 1975 - 1976 concrete was not properly vibrated.

Finding: No specific information was provided by the alleger as to locations where this may have occurred. Therefore, general program requirements and records were reviewed. Review of concrete placement procedure, QCP9, "Concrete Placement Control" shows a requirement in Subsection 3.3.5 for the use of mechanical vibrators for consolidation of fresh concrete while Subsection 3.3.1 describes the methods for vibrating previous pours to assure the establishment of a bond between pours or layers of concrete.

NRC examination of concrete placement records for pours made during 1975-1976 shows that the compaction (vibration) of each of the pours was inspected and found to be acceptable by quality control inspectors. The placement records examined consisted of pours in containment walls, reactor building columns, auxiliary building walls, columns and slabs.

The inspector's review of training records shows that the concrete contractor was conducting training in the correct methods of vibrating concrete at frequent intervals prior to and during the period of questionable vibration practices. Training sessions were conducted on:

4/17/74 for 17 personnel
9/16/74 for 9 personnel
10/26/74 for 50 personnel
1/29/75 for 32 personnel
2/28/75 for 29 personnel
4/9/75 for 8 personnel
4/28/75 for 11 personnel
5/17/75 for 17 personnel
5/21/75 for 12 personnel
6/6/75 for 9 personnel
6/16/75 for 10 personnel
7/22/75 for 35 personnel
8/12/75 for 18 personnel
10/13/75 for 6 personnel
10/15/75 for 24 personnel
12/8/75 for 8 personnel
2/9/76 for 11 personnel
9/21/76 for 18 personnel

Since procedures for placement of concrete were developed and implemented before the concrete was placed, the procedures included instructions for vibrating the concrete. Personnel were trained in

the proper use of external concrete vibrators. Each pour was supervised by engineers, superintendents, foremen, and inspected by quality control during and subsequent to the pour. The NRC inspector examined 29 drilled cores from concrete poured during 1975-1976 and did not detect evidence that the concrete was improperly vibrated.

Conclusion: This allegation was not substantiated.

SECTION VI

Prepared by: R. D. Lanksbury

10. Secondary Containment Test

Allegation: In 1978 a reactor building pressure test blew the roof off.

Finding: The reactor building encloses the reactor and the primary containment and provides secondary containment integrity when the reactor is in service. The reactor building up to and including the operating floor (refueling floor), is of reinforced concrete. The operating floor is the uppermost level in the reactor building and consists of a structural steel super-structure enclosing the floor and supporting the sheet metal siding and decking. The decking is covered with insulation and composition roofing as a water proofing covering.

As reported by the resident inspector in inspection report 50-373/81-30, sometime during the weekend of July 18-19, 1981, while LaSalle was in the process of conducting the Secondary Containment Leak Rate Test, the reactor building was inadvertently overpressurized, causing damage to the reactor building composition roofing and insulation. This damage was reported to the NRC by Commonwealth Edison pursuant to 10 CFR 50.55(e) in Report 81-05, dated 8/18/82 (the event was verbally reported on 7/20/81). Several NRC inspectors, including the Senior Resident Inspector, inspected the structural area above the operating floor. No deficiencies were identified.

Secondary containment integrity testing was performed on April 14, 1982 and again on April 24, 1982, with both tests being acceptable and meeting all FSAR and technical specifications requirements. Both of these tests were witnessed by an NRC inspector to verify compliance with all requirements, as reported in inspection reports 50-374/82-20 and 50-373/82-26, respectively.

The 1978 date given in the allegation appears to be in error. In 1978, the walls, decking, and roof of the reactor building were under construction and no testing was performed until 1981. This has been verified through review of photographs of the site that are taken on an appoximate monthly basis and by discussions with NRC and construction personnel who were present during this time period.

Conclusion: The inspector found that during a Secondary Containment Test in 1981, the reactor building was overpressurized causing damage to the roof. The inspector reviewed records and verified that the roof was repaired and two successful tests were conducted in April 1982. No further followup action is planned.

11. Improper Installation of Hanger Supports

Allegation: Metal in hanger supports is identified by a color code. When construction (the piping erection contractor) had the right size supports, but not enough of the right color, the hanger colors were changed. This practice was observed in an area near Unit 1 reactor.

Finding: The inspector interviewed representatives of the licensee and piping erection contractor organization who are responsible for the work and materials in question. It was stated that no color coding system is in use for the identification of hangers, or other piping support elements, by material or other characteristic. This position was reinforced by reviewing the contractors procedure for "Erection of Supports-Restraints and Final Installation Verification," and touring the yard lay-down area where hangers are stored. The procedure makes no reference to any color coding scheme. No evidence of color coding could be identified in the lay-down area.

During the inspection it was determined that all safety related hangers installed by the piping installation contractor are supplied by a single vendor, with the exception of those hangers which are fabricated or modified at the site. The same material is used in the fabrication of all safety related hangers, and each hanger is supplied with a letter of conformance which identifies that material. The same manufacturer also supplies the material (with letters of conformance) which is used by the contractor for field fabrication or modification of hangers.

One practice was identified which is included in this report because it could conceivably be misconstrued and result in the basis for this allegation: All hangers are painted to provide suitable corrosion protection. The type of paint to be used is specified by the architect engineer. It appears that all hangers supplied in the past were painted with a paint containing lead. The station design, however, does not allow for the use of lead bearing protective coatings inside of the reactor containment. As a result, all hangers designated for use inside of one of the containments, and which are painted with lead bearing paint, are stipped to base metal at the site and repainted with a lead free paint which is a different color. This practice may have been misconstrued as misapplication of the hangers themselves.

Conclusion: The inspection findings do not support the allegation. No further followup action is planned.

12. Water Leakage Through Auxiliary Building Basement Wall

Allegation: The J-Line wall in the Auxiliary Building basement is leaking water, and has been for some time.

Finding: Interviews with the licensee's Lead Structural Engineer and the cognizant contractor's Quality Assurance Supervisor, in addition to review of two existing non-conformance reports (NCR's) established that this condition exists and that the licensee had been aware of this condition since about August 1979. The inspector toured the wall and the surrounding area and observed:

a. The wall was dry, however, latent water stains on the concrete wall confirmed that leakage had occurred in the past.

b. A series of floor drains existed in a parallel line approximately six feet from the walls. The drains were capable of collecting and removing leakage.

c. No safety related equipment was located in the immediate vicinity of the wall.

Several days later the inspector verified that minor leakage (surface moisture) was occurring. Contractor personnel were in the process of repairing the leaking portions of the wall. The findings and repair methods were discussed with regional based construction inspectors. It was concluded that the repair methods were in conformance with industry practice and were acceptable.

It was also noted that no requirements concerning leakage were included in the design of this wall.

Conclusion: Although the allegation was substantiated, the situation represents no threat to the operability of safety related equipment; it is being repaired by the licensee. No further followup action is planned.

13. Reactor Building Settlement

Allegation: The northeast side of the Unit 2 reactor building is settling, and has already settled four to five inches.

Finding: The reactor building for Unit 2 is common with Unit 1, so this matter was included in items requiring a prompt review. The inspector interviewed licensee and structural contractor personnel who are responsible for the monitoring and evaluation of settlement of structures which are important to safety. Site records of Reactor Building settlement data and applicable portions of the Final Safety Analysis Report (FSAR) were also reviewed.

The responsible licensee and contractor personnel were not knowledgeable of settlement or differential settlement of four to five inches. The Reactor Building settlement data for Units 1 and 2 through May of 1981 had previously been compared to the theoretical predictions in

the FSAR by NRR. It was concluded that the recorded data was in reasonable agreement with the FSAR predictions. The inspector then compared the settlement data of May 1981 through May 1982 to predicted values. It was noted that Reactor Building settlement has apparently stabilized at approximately 2.4 inches. No indications of unacceptable differential settlement were noted in the NRR or inspector reviews.

Conclusion: While settlement of the reactor building has occurred, it has not occurred to the specific amount alleged. Settlement is within the limits specified in the FSAR.

SECTION VII

Prepared by: F. W. Reimann

R. D. Lanksbury

J. R. Kniceley

14. Acts of Sabotage (Broken Gauges and Controls and Flammable Liquids in Fire Extinguishers)

Allegation: Equipment was damaged during installation. In 1979 gauges were broken and flammable liquids were being placed in fire extinguishers.

Finding: The Station Construction Fire Marshall stated that he had no knowledge of flammable fluids being placed in fire extinguishers. The Operations Fire Marshall stated that he had heard rumors of this practice, but that in the last five years, no evidence to support the rumor has surfaced. He also stated that he has no knowledge of any individual who had found the rumor to be a fact.

The inspector reviewed the site annual fire extinguisher maintenance records for 1980, 1981, and 1982. The maintenancep rocedure requires disassembly of the fire extinguishers, and, therefore, should reveal the existance of flammable or other foreign substances in them. No flammable or foreign substances were recorded in the records. A review of the site security records for 1979 revealed no acts of sabotage that would collaborate this allegation.

The Operations Fire Marshall also stated that fire extinguishers had been used on several occasions for training exercises in fighting actual fires and to extinguish fires. No cases of flammable liquid in fire extinguishers were noted as a result of this usage.

The matter of instrumentation being damaged during or after installation is not new to the NRC. Inspection Report 50-373/80-15 addresses this issue and contains a noncompliance for inadequate control of damaged instruments. This matter has been resolved between licensee management and Region III, the Office of Nuclear Reactor Regulation (NRR), and the Office of Standards Development. The improvement of cleanliness and controls to protect installed instruments was discussed in several meetings with the licensee, and in some cases, the licensee's contractor management. The issue was considered resolved when reasonable assurance was provided to the NRC that adequate tests, surveillance, and calibrations were planned prior to fuel load to assure that instruments were in good working condition at the time of fuel load. Such preoperational tests have been performed, and a routine surveillance program is required during plant operations.

The inspector's observations of approximately 150 safety related instruments randomly selected during plant tours of Units 1 and 2 on June 17 and 23, 1982 resulted in no adverse findings. Degraded or potentially degraded instruments were properly identified in accordance with applicable licensee procedures.

Conclusion: Some problems with broken and damaged gauges had been identified in the past. Because of licensee corrective actions and subsequent testing, these matters are considered closed. No further action is necessary.

SECTION VIII

Prepared by: H. M. Wescott H. M. Wescott

R. N. Sutphin C. T. Nordin for RNS

J. R. Kniceley J. R. Kniceley

15. Welder Unqualified or not Properly Certified

Allegation: There were problems with the certification of welders by the architect engineer in 1977 and 1978. When section III documents of welder certification were checked, it was found that approximately 1% of the documents were done by uncertified welders.

Welders at the site of specified national origins were performing safety related welding and were not certified (or qualified) welders.

Findings: The inspectors interviewed the responsible members of the architect engineer's QC Division, and reviewed the AE's scope of work in regard to welding, the procedures which control that scope of welding work, QC records of AE activities, internal audit reports, and documentation of welding activities performed at the site. Licensee individuals who control welding and welder certification at the site, and site procedures and records were also reviewed. It was found that the AE had many responsibilities in the area of weld design, weld procedure certification, and the review of welding data.

The AE was not, however, responsible for the certification or maintenance of the certification of welders. Responsibility for the review, approval, or acceptance of welder certifications is retained by the licensee. There were occasions when the licensee involved the services of the AE in the resolution of items relating to the welder certifications of contractors or vendors. The AE personnel who assisted in the resolution of problems appeared to be qualified.

The inspectors reviewed licensee records and procedures and noted that all welders who performed work at the site were tested prior to employment, and appeared to be qualified and capable individuals. Each welder is assigned a unique welder identification number, and that number is recorded with the remainder of the pertinent data for all welding which is safety related. A review of welding specifications, procedures, quality control procedures, welder certification procedures, and resulting records of work performed, quality documentation, and welder certification failed to identify any deficiencies in this area.

In addition to the above inspections, the records of 13 selected welders of the national origin identified were reviewed. No deviations, departures, or practices different from the remainder of the certification program were noted. No adverse findings were noted.

SECTION IX

Prepared by: R. N. Sutphin E. E. Murlin for RNS

J. Kniceley J. Kniceley

16. Inadequate Training of Quality Control Personnel

Allegation: Quality Control personnel employed in the mid-1970's in the QC Division of the Architect Engineering firm (Sargent and Lundy) were not given adequate training to perform assigned work, and the required certifications were not maintained.

Finding: The inspector interviewed the Head, Quality Assurance Division, the QC division Project Quality Control Supervisor, a Quality Assurance Team Leader, and six current and former employees of the QC Division.

The architect engineer's internal procedures and documentation for training, qualification, and certification were reviewed and no deficiencies were identified. The results of two internal Quality Assurance audits, which were conducted to evaluate the conformance of the Quality Control, Mechanical (DMD), Electrical (EPED), and Structural (SPED) divisions to procedural requirements, were reviewed.

The alleger supplied the names of other individuals who, in the alleger's opinion, would confirm the concerns. One individual who worked in the same capacity and during the same time period when the alleged problem occurred stated that she had no knowledge of the conditions cited, and provided a statement that in her case training was acceptable and good. The other individual could not be located.

The following are details of the inspection of the Sargent & Lundy Training Program:

a. GQ-1.04 states in paragraph D.1, "An individual shall not perform any quality related activities prior to approval of his employee qualifications statement" (Form GQ-1.02.2)

b. GQ-2.04 states in paragraph A.1, "Training shall be required to the extent necessary to assure that each individual shall achieve and maintain a suitable knowledge of the requirements of the QA program and QA procedures pertinent to the individual's position and to assure that suitable proficiency is achieved and maintained".... "Personnel who direct, manage, supervise or perform quality related activities as outlined in the S&L QA Manual shall require training when one or more of the following criteria apply: when an individual is promoted, or transferred, or loaned..., when an individual is hired, exhibits an inadequate proficiency when revisions are made, etc."

c. GQ-2.05 states in part - paragraph 1.0, "The objective of the technical training shall be to indoctrinate and train personnel who are performing quality-related activities, to assure that suitable proficiency is achieved and maintained."

Paragraph 3.0 B.1, "Training may include, but not be limited to, the use of NRC regulations, industry codes and standards, and S&L administrative and technical standards, as they apply to the design of nuclear power plants."

Paragraph 3.0 C.1 "A trainee's proficiency record shall be maintained in the department/division training file."

S&L General Quality Assurance Procedures GQ-1.04, GQ-2.04, and GQ-2.05 covering employee experience records, qualification statements, QA/QC indoctrination and training, and technical training provide an adequate basis for training and certification of S&L personnel, and the maintenance of required proficiency for personnel working in quality related activities.

An Internal Audit No. G-23 conducted on December 16, 1976 covered the following:

a. Organization chart for the Quality Control Division

b. The description of organizational functions and responsibilities of the QC Division

c. The roster of personnel for Quality Control Division

d. The Position Descriptions for the Personnel of QC Division

e. The employee experience records of QC Division personnel

f. Employee qualification statements for QC Division personnel

No deficiencies were found during the audit; this audit appeared to be complete.

To determine whether divisional personnel of PMD (Mechanical), EPED (Electrical), and SPED (Structural) are complying with the requirements of the QA procedures, internal Audit No. G-58 conducted on June 28-30, 1981 covered the following:

a. GQ-1.01, Rev. 4 - S&L Plan of Organization

b. G-1.03, Rev. 3 - Organization of Position Descriptions

c. GQ-1.04, Rev. 2 - Employee Experience Records and Qualification Statements

d. GQ-1.04, Rev. 4 - QA Training

e. GQ-1.05, Rev. 3 - Technical Training

f. ME-1.14, dated December 16, 1976 - Mechanical Department Technical Training Program

g. ESI-230, dated April 27, 1977 - Electrical Department Technical Training Procedure

h. SAS-24, Rev. 0, dated December 3, 1976 - Technical Training of Structural Department Employees

No deficiencies were identified during this audit.

Records of the QA Training Section for 1977 show that QA training was given to 82 of the 102 employees in the Project Management Division, 61 of the 71 in the Electrical Project Engineering Division, and 22 of the 28 in the Structural Project Engineering Division. A substantial number of persons from the three engineering divisions had received retraining in the first half of 1978 for those QA procedures which have been revised.

Employee Qualification Statements and Employee Experience Records were reviewed for a selected group of individuals, including specific individuals who were alleged to be unqualified. These were found to be properly documented showing conformance to the company's procedures.

Conclusion: Sargent & Lundy had a continually improving training and certification program during the 1970's. Records were maintained and appropriate qualifications, training, and certification programs were in effect. Adequate programs for work orientation and training were in effect and seemed to be effectively implemented. There were no deficiencies found in the program or in the related documentation for selected individuals.

17. Coverup of Deficiencies

Allegation: An architect engineer supervisor with responsibility at the LaSalle Project allegedly told an employee that he knew of many mistakes in the plant and how they were covered up.

Finding: The alleger provided no specifics regarding the nature of the alleged mistakes or portion of the plant affected.

It is noted that throughout the history of the project, a continuing program of inspections, audits, and investigations have been conducted by the NRC. Deficiencies that have been identified during these inspections have been documented and corrective action had been taken.

The individual who expressed the concern gave the names of two other former architect engineer employees who could provide additional information. One of the individuals could not be located. In a telephone interview with an investigator, the second individual stated that any problems she found were corrected through her supervisor, she was not aware of any attempts to cover up or hide anything, and she was not aware of statements attributed to any supervisor which were alleged.

The inspector interviewed six architect engineer and licensee managers in addition to reviewing eleven AE procedures, audits, and standards which control the AE scope of work. No adverse findings were identified.

Conclusion: The allegation could not be substantiated.

18. Nonconforming Material

Allegation: Metal for a valve was rejected because it did not meet the carbon content requirements. The original number was changed to meet the requirements.

Finding: The inspector interviewed personnel and reviewed records pertaining to valve specifications. The inspector also reinterviewed the alleger by phone to try to obtain more details regarding this allegation. The alleger could not provide any additional information.

In the absence of specific details, the inspector interviewed personnel and reviewed records pertaining to certification of material. The following records and documents were reviewed:

a. Quality control documentation lists (QCDL) for 4 specifications:

(1) J-2937 Control Valves (Section III)
(2) J-2938-02 Globe Valves (Section III)
(3) J-2961 Piping System Prefabrication
(4) J-2950-01 OFF Gas Valves (Section III)

The quality control documentation lists provide a record of documentation that is required, received, reviewed, and transmitted to CECo and the site. It covered technical administration requirements and detailed item requirements.

b. Q.C. correspondence files and document record files for 3 specifications:

(1) J-2937 Control Valves (Section III)
(2) J-2938-02 Globe Valves (Section III)
(3) J-2950-01 OFF Gas Valves (Section III)

QC Correspondence and Document Record files contained copies of all record transmittals, reviews, comments, concerns, rejections, requests for missing documents or information, responses, acceptances, and final record copy.

c. Quality control inspection certificates, certified material test reports, and ASME Code (Form NPV-1) data reports for several sample valves. Certified material test reports contained detailed data on the chemical composition and mechanical properties for each item or part number, as required. Several certified material test reports for valves covered by the 3 specifications referenced were reviewed and checked for irregularities or nonconformances and none were found. No changes or missing data was evident.

d. Audit reports. Audit Report No. 34 of audits performed on July 25 through 28 and 31, and August 1 and 2, 1978, covered 16 items associated with the QCDL review and acceptance process.

Two items were questioned in a sample data package relating to a wall thickness measurement and a review of a radiograph. However, these did not affect the chemical composition or mechanical properties of the valve material. Subsequent information indicated the particular questions raised applied to a valve installed in the off gas system for Unit #2.

Subsequent to this question on a valve body wall thickness, the valve was located in the field and a complete ultrasonic wall thickness examination performed. The report of the examination, Report No. 28, dated June 16, 1982, indicates that the smallest reading found was 0.60". The minimum required was 0.58", so the valve body appears to meet the specification. The radiograph in question was reviewed and found to have been covered by a repair and reradiograph identified as 17-10 R on the radiograph examination report dated December 7, 1978. At this time these two questions have been addressed in a satisfactory manner.

All of the data on the certified material test report for sample valve No. 2N62-F042A was checked against the 8 ASTM specs that applied and all data were found to be within specified limits.

Conclusions: All data observed and checked regarding material chemical composition and mechanical properties appeared to be correct original data and was within specified limits. Therefore, without specific identification, it is concluded that this allegation has not been substantiated.

19. Conflict Between Specifications and NRC Requirements

Allegation: Specifications made by the Architect Engineer were in conflict with NRC requirements.

Finding: Since the alleger was not able to provide specific details regarding this allegation, the Region III inspector reviewed procedures and records relating to the handling of specifications. The following procedures and records were reviewed:

a. General QA Procedure GQ-4.01, Procurement Specifications

b. Project Instructions (La Salle) PI-LS-03, Project procurement and conformed technical specification requirements; and PI-LS-IS, Processing of Commonwealth Edison nonconformance reports and Sargent & Lundy Engineering Change Notices.

c. Audit Reports (Internal) No. 22 8/12-13/76, G-23 12/23/76, 34 8/22/78, and G-58 7/27/78;

d. Status of Project Specifications, reports dated 6/1/76, 5/1/82, and 6/1/82;

e. Interoffice Memoranda Re: Reviews & Comments on Specification J-2937 (as a sample), dated 1/11/73, 11/14/74, 11/25/74, 12/3/74, 1/17/73, 4/7/73, 9/23/73, 10/27/73, and 3/5/76.

The S&L General Quality Assurance Procedure GQ-4.01, Procurement Specifications, includes reference to several pertient standards and requirements including NRC's 10 CFR 50 Appendix B, NRC's Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction);" NRC's Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants;" NRC's Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants;" ANSI/ASME N45.2, "Quality Assurance Program Requirements for Nuclear Facilities;" ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants;" ANSI N45.2.13, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants."

Instructions were documented on the manner in which revisions could be made to the specifications. Revisions must be prepared, reviewed and approved in the same manner as the original issue as required by the nature of the revision.

The procedures states that technical requirements must be specified in each procurement specification either directly or by reference to specific drawings, specifications, codes, regulations, standards, procedures, or instructions, by their specific titles, numbers, and revision and/or due dates, which describe the items or services being procured.

Specifications must identify, or provide for later identification of test, inspection and acceptance requirements, and any special instructions and requirements for activities such as designing, identification, fabrication, cleaning, erecting, packaging, handling, shipping and extended storage.

Specifications require that the contractor have a documented quality assurance program that implements applicable portions of Appendix B of 10 CFR 50, as well as all other applicable nationally recognized codes and standards. Specifications also require that the contractor include appropriate quality assurance program requirements in all sub-tier procurement documents.

LaSalle Project Instruction PI-LS-03 establishes the requirements for project procurement and conformed technical specifications and amendments thereto for the LaSalle County Station - Units 1 and 2. This Project Instruction supplements the requirements of GQ-4.01 for procurement specification and establishes specific requirements for preparation of conformed technical specifications.

Conclusion: The inspector found that Sargent and Lundy programs for developing specifications, for their review and approvals, for revisions, for compliance and for adequacy are well defined and followed; the audits reviewed showed compliance with the necessary requirements. Detailed reviews of the files for 3 specifications developed no items of concern or lack of compliance with NRC requirements. In the absence of a specific specification identified to be potentially in conflict with NRC requirements, no information was identified to substantiate the allegation.

SECTION X

Prepared By: R. Lanksbury

F. Reimann

H. Wescott

F. Jablonski

J. Kniceley

20. Calibration of Torque Wrenches

Allegation: Records for torque wrench calibrations by the site piping contractor (Morrison) were falsified.

Findings: The contractor's program for controlling, issuing, calibrating, and record controls for torque wrenches used for construction of safety related equipment was evaluated. It was found that approximately sixty torque wrenches of the dial indicator and slip (clicker) type were maintained for use by Morrison personnel. The wrenches are stored in one of two locked tool cribs, and are logged out by serial number for each job. Verification of calibration is maintained in the field by the presence of a gummed calibration sticker which identifies the wrench, the date of latest calibration, and the due date for calibration verification. Each wrench is calibrated weekly, except for wrenches which are removed from the tool cribs for maintenance or repair. In the event of damage to or loss of a wrench, procedures require that each job on which the wrench was utilized since its previous calibration be rechecked.

Once each week the wrenches are returned to the Morrison construction office for calibration verification. This verification is performed using a bench testor which serves as a transfer standard. The accuracy of the transfer standard is verified using calibrated standard weights (traceable to the NBS) by the licensee (CECo) every six months. The physical work of verifying torque wrench calibrations, performing minor wrench repairs, and recording calibration data is accomplished by two craft personnel (millwrights), employed by Morrison. The same individuals have performed this task for at least the past two and one half years. The same QC inspector has been assigned to inspect the quality of the work, and maintaining the records of calibrations, repairs, and nonconformance reports (which are required if a wrench is found out of specification or broken) for at least the five previous years. The inspector is also employed by Morrison.

Morrison's procedures and records indicate that torque wrenches were used for installing concrete expansion anchors (which support components, supports, and restraints), in bolting flanges, valve parts, and similar mechanical fastenings. Bolted fastenings which have critical preload requirements (such as jet pump support beams, large

equipment assemblies, etc.) are not torqued (they are preloaded by elongating the bolt using hydraulic pressure), as is standard industry practice.

A sample of about 1,500 of the approximately 6,000 records of weekly torque wrench calibrations were reviewed. Fifteen records of semi-annual verifications of the torque wrench calibrator by CECo were also reviewed. Of this sample approximately 70% of the records were found to exhibit one or more of the following questionable record keeping practices (Attachment E is a sample of the torque wrench calibration record PC-139. Numbers have been added to assist in referencing to the report text):

a. Three cases where the records of calibration for a single wrench were photocopied from the results obtained for a different calibration interval. The master used to make the copies was found to be the next record, chronologically, after the copies. The number of consecutive weeks for which results were photocopied ranged from three to twelve weeks.

b. Frequent instances where the information in Form PC-139 columns 3, 4, 5, and 6 were written in ink, but all or combinations of items 7, 8, 9, 10, or 11 were photocopied, indicating that signatures verifying that the work had been correctly done and performed were affixed prior to the work being performed.

c. Frequent instances where the value recorded in column 2 of Form PC-139 was repeated in columns 3 and 4 rather than recording actual data. Errors were then recorded as 0. There were also frequent instances where the values recorded in columns 3, 4, 5 and 6 were repeated week after week with no variance. This occurrence is unlikely based on the difficulty of repeating an exact reading.

d. A small number of instances (less than 1%) where the actual recorded data was out of tolerance a few percent, and later altered to acceptable values. Reading of the final recorded values was not physically possible because they required the calibration device be read to one tenth of a foot-pound or inch-pound on a scale calibrated in five foot-pound (inch-pound) increments.

e. In a small number of cases (less than 1%), the actual data recorded exceeded allowable tolerance limits by a few percent. A few mathematical errors were also found. This data was reviewed and approved by the QC inspector (items 7 through 9 on Form PC-139) and not challenged.

f. The records of torque wrench calibrator records were found to contain approximately six calibration point records which were 5% or less out of tolerance. The records were reviewed and approved, and not identified as being out of tolerance, nor were corrective actions taken.

Following the discovery of the above questionable records, the inspection was divided into three separate initiatives: (a) the re-evaluation of work performed in the field to determine if actual unacceptable work on safety related equipment existed (this initiative had actually started prior to this time and was underway when the questionable records were identified); (b) a further evaluation of the suspect records and subsequent engineering analysis to determine whether the worst case out-of-tolerance conditions indicated in the records could result in conditions adverse to safety; and (c) an examination of other Morrison records for Measuring and Test Equipment (M&TE), and an evaluation of torque wrench programs of other site contractors to quantify the extent of the observed records problem.

a. Work Re-Evaluation

The NRC inspector randomly chose three safety related valves for verification of proper torquing. The inspector witnessed the torque verification. Two of the three valves were found to have bolted fastenings which were verified to be at least equal to the minimum torque requirement. The third valve was found to have two of the four bolts which hold its motor operator to the valve yoke to be loose (not threaded to the hand tight condition). This condition was not related to torque wrench accuracy. Morrison's records for this valve (mechanical joint checklist) indicated that the loose bolts had been torqued to 50 foot-pounds and verified by a QC inspector. Maintenance records indicated that no work had been performed on the valve since its installation. A second group of bolted fastenings on the same valve was determined to conform to at least the minimum torque requirement. The loose bolts were retorqued.

As a result of this finding, CECo, at the NRC's request, committed to verify the integrity of all bolted joints relating to the operability of safety related air and motor operated valves. Bolted joints which form a pressure retaining boundry were exempted on the basis that their integrity had been tested when the system hydrostatic test was performed.

(1) Valves inside of the containment have since been verified to be torqued to the minimum required value. Of the approximate 6,000 bolts checked by CECo, four motor operator to yoke bolts on four different valves were found to be only hand tight and one bolt was found to be approximately 10 ft-lbs below the required torque. These bolts have been retorqued to the required values.

(2) Valves outside of the containment will be checked by January 15, 1983.

A random sample of 10% of Morrison's torque wrenches were selected for a recalibration test which was witnessed by an NRC inspector. In all cases the data was in disagreement with the

latest routine calibration records, and the trends established in the previous several weeks. In all but two cases, however, the results were acceptable. The out-of-tolerance results for the two wrenches were less than 5% beyond acceptable limits. Nonconformance reports were initiated for the out-of-tolerance wrenches. The differences between NRC witnessed results and earlier records is considered corroborative of finding c. above.

NRC investigators and inspectors interviewed five Morrison individuals primarily responsible for torque wrench calibration testing and QC acceptance. They denied that they had ever made up a record without performing any calibration at all. It was determined, however, that calibration data was not always taken as required by procedures. For example, those individuals who performed calibrations said that on certain occasions they may have checked only three of the five calibration points on a wrench. They also stated that if the readings were within the acceptable range, they sometimes recorded the acceptance value rather than the actual reading. The QC inspector stated he did presign blank calibration data forms which were used by the millwrights. This was done as a short-cut to speed up the QC acceptance process. The millwrights provided conflicting statements at different times as to whether or not they entered data on presigned forms. QC routinely reviewed these forms later for completeness, but not necessarily for technical accuracy. Everyone interviewed stated they never changed any numerical data on any calibration form, and they are not aware of anyone who has. Forms and procedures have now been changed and QC now witnesses all calibrations.

The largest single category of work involving the use of torque wrenches was the installation of concrete expansion anchors (CEA's). In addition to the verification of proper installation of CEA's for wrenches with suspect calibrations (discussed above), it was determined that the installation quality was additionally verified immediately following initial installation (at least six hours after but not more than two weeks after) by an independent testing contractor retained by CECo. The QC effort has continued as the anchors were installed. The independent test contractors program of inspecting CEA installation consisted of sampling one CEA on each base plate, or 10% of the bolts, whichever was greater. The adequacy of CEA installation was further verified by the licensee's actions in response to NRC Inspection and Enforcement Bulletin 79-02 (which specifically addresses failures of CEA's). This included verifying and providing information on various aspects of the design criteria utilized and verification of proper installation, including installation torque, test torque, embedment length and anchor size. Their response and actions are documented in the final report, dated February 8, 1982, provided by CECo.

As part of their commitment to the NRC, CECo checked the torque values for any item whose initial torque was questionable based upon suspect calibration records for the torque wrench utilized. A portion of these checks were witnessed by an NRC inspector. A total of 67 hangers with approximately 392 CEA's and the bolting of two valves were checked. The two valves were checked as part of the verification noted in a.(1)., above. Of the 392 CEA's checked, fourteen CEA's on nine hangers failed to meet the specified verification torque. Upon retorquing the fourteen CEA's and again checking the verification torque, two CEA's failed. These two CEA's will be replaced per established procedures. It was noted by CECo that upon recalculation of the loading forces on the two base plates involved that they would have functioned without these two CEA's.

b. Engineering Evaluation

Two NRC experts in mechanical fasteners and two NRC consultants, one an expert in mechanical fasteners and one an expert in code requirements, conducted an evaluation of the engineering implications of deficiencies which could arise as a result of torque wrenches which were either out of calibration or potentially out of calibration. The experts independently evaluated the problem and met with CECo and Sargent and Lundy representatives who were performing a similar evaluation.

At the meeting, CECo addressed three primary areas of concern: (1) Motor Operator to Yoke Bolts, (2) Expansion Anchor Bolts, and (3) Torque Wrench Calibration Accuracy. The following paragraphs discuss each area of concern.

(1) Motor Operator To Yoke Bolts

This concern originates with the problem identified by the inspector on the two loose motor operator to yoke bolts. As previously stated, CECo agreed to check for tightness all non-pressure boundary bolts on safety-related valves, with motor or air operators, that could affect valve operability. While one of the consultants included in his recommendations specific provisions for checking both upper and lower torque values, the NRC staff required that these bolts be checked only to assure that they met minimum torque values. Valves within containment have been checked, as discussed above, and valves outside containment will be checked by January 15, 1983. An NRC inspector reviewed the licensee's procedures for re-checking torque valves and found them acceptable.

(2) Expansion Anchor Bolts

Sargent and Lundy presented their study of anchor bolt seating torques and their justification for the values chosen for their verification torques. Studies made of the torque, holding power, and slip characteristics were presented and copies of a portion of their report on their

findings were given to those present. It was explained that the preload on the anchor dissipates, due to concrete creep, over a period of time after installation and that the preload is not required for the anchor to withstand cyclic loading but is used to verify that the anchor bolts are properly installed. This was the justification provided for the 60% of installation torque value assigned to the verification torque. All parties present agreed that the above was technically satisfactory. CECo stated that when anchor bolts are found to turn prior to the verification torque, they would be taken all the way up to the installation torque and then rechecked. If they failed to meet the verification torque the second time, a nonconformance report (NCR) would be written to disposition that anchor bolt.

(3) Torque Wrench Calibration Accuracy

The "worst case" of torque wrench miscalibration was established from the recorded data and rechecks of the torque wrenches in question, and was stated by CECo to be less than 15% (including allowed calibration error). It was noted by the NRC consultants that typically torque wrenches are accurate to approximately ±10% and that analytical calculations are considered accurate at ±20%. It was also established by CECo that the known applications of the questionable torque wrenches did not require close tolerances (i.e., primarily anchor bolts).

It was also noted that torque does not produce an accurate bolt preload. Where an accurate bolt preload is required, it is normally accomplished using a more accurate method such as by hydraulic tensioning (e.g., jet pump holdown beam bolts, reactor vessel head bolts, etc.). It was concluded by the NRC experts and the NRC consultants that from a design standpoint, the effect of the out-of-spec torque wrenches does not have a significant impact on the bolted components ability to perform its intended function. Thus, providing no bolt applications are found where questionable torque wrenches were used and where an accurate preload is required, it was concluded that the inaccuracies of the torque wrenches is not a significant safety concern with respect to design.

As a result of these analyses, it is concluded that the worst case out of tolerance conditions (less than 15%) which could result from the conditions observed to exist would not result in a reduction of safety margins. The NRC consultants' reports are included as Attachments F and G. The final results of CECo's analysis were also submitted to the NRC experts and consultants for evaluation. All parties have agreed that the reported results, along with the inspector's site verification of the procedures used, are acceptable to demonstrate that the licensee's program met the NRC staff's requirements.

c. Additional Record Review

A review of records of and interviews with representatives of other site contractors were performed to determine whether the deficiencies observed in the Morrison torque wrench program could affect work in other contractor scopes of work. It was found that with two exceptions, other contractors who use torque wrenches used equipment supplied by other manufacturers, and that the calibration programs for them were independent of the Morrison program. The two exceptions to this finding are Reactor Controls, Inc. (RCI) and the Zack Company (Zack). RCI had a contractual agreement since 1980 by which Morrison performed their torque wrench calibrations. Prior to 1980 the RCI calibrations were performed by Gage Laboratories (an independent corporation). No questionable findings were found in the Gage records.

The Morrison records for RCI wrenches were found to have many of the same findings as Morrison's own records. However, RCI had performed a documented independent review of results, and their records did not show out of tolerance results. It was also found that RCI routinely performs and documents a check of one torque wrench against another (both calibrated) both before and again after each use. The above findings were corroborated by CECo during their audit of RCI M&TE documentation. It is also noted that for CEA installation work performed by RCI, the same type of independent verification program was in effect as noted above for Morrison.

CECo performed an audit of Zack M&TE records and found that the only calibration work performed by Morrison for Zack was for torque wrenches. Zack's use of torque wrenches was limited to CEA installation and equipment holddown bolts. In December 1980, Zack instituted a recheck of all work performed previously. Therefore, only records from December 1980 to the present were reviewed. This review revealed problems with repetitive data, as noted above, but no other problems were noted. As with Morrison and RCI, the CEA installation was sampled by the independent testing contractor retained by CECo.

A review of approximately 10% of other types of Morrison M&TE documentation was also performed by the NRC. Documentation problems similar to those encountered for torque wrenches were found in less than 20% of the sample. Evaluations of the magnitudes of errors which may have resulted, and, particularly, the observed condition of equipment which was recalibrated in the presence of NRC inspectors indicate that the findings relate to poor record keeping rather than deficiencies in installed plant equipment.

d. Licensee Audit

As a result of the questionable status of Morrison's QC records and the records of those contractors utilizing Morrison's calibration service, the NRC requested CECo to perform an audit of 100% of Morrison's, records for M&TE (including those calibrations which Morrison performed for RCI and Zack), and additionally to

audit a minimum of 75 to 100 documents for each type of Morrison's QC records other than M&TE records. The results of this audit, conducted between June 29 and July 8, 1982, are included as Attachment H to this report.

CECo's twelve man audit team concluded that the documentation problems identified by the NRC were limited to calibration records. Their evaluation of the reports where torque wrenches were found to be out of calibration or had questionable calibration data or records concluded that they had no adverse impact on plant construction. This conclusion was based on the records reviewed and the results of the independent testing agency's results of bolt torquing inspections. It was found that the independent testing agency examined 6,756 Morrison installations between 1977 and June 1982. Of these, 81 (1.2%) were rejected for low torque values during the six year period. In addition, they stated that examination by Quality Assurance (QA) of follow-up checks conducted by Project Construction of other calibration areas (e.g., thermometers, hygrometers, calipers, etc.) did not identify any cases where the installed condition of Unit 1 equipment appeared questionable.

CECo stated that many qualified individuals had looked at this area but that without being specifically attentive to looking for record alterations they would not have identified them as a quality issue. In addition, their examination of the overall surveillance and audit process indicated that their auditing methodology involving large numbers of similar records needs to be changed to provide that larger samples be taken to achieve a better assessment. The CECo QA department committed to implement this change.

The NRC received this report on July 9, 1982. On July 12, 1982, the NRC met with CECo management to discuss the report. The NRC also performed an independent sampling audit of Morrison's QC records and reviewed some of the detailed data which supported the report findings. NRC inspectors independently reviewed calibration records for torque wrenches, micrometers, dial indicators and temperature measuring devices. A sample of pressure test records were reviewed. Also, a sample of several other types of QA records maintained by the licensee (welding travelers with associated nondestructive testing records, nonconformance reports, receipt inspection reports, welder qualifications) were reviewed. A review of the audit qualification records for seven (7) of the principal auditors performing this audit was also performed. The qualifications met the requirements of ANSI-N45.2.23. This independent sample tended to confirm the validity of the CECo audit. The scope of the audit was compared to the commitments made by the licensee, and documented in the J. G. Keppler to C. Reed letter of July 8, 1982. While it was found that the sample size for documents handled by individuals who generated other questionable documents was not as large as CECo had committed to, the Region III staff concluded that the existing audit and followup inspection by the NRC has resulted

in the identification of all technical problems which exist. The licensee has been directed to complete a 100% check of documents handled by individuals who generated records which were found to be suspect.

Conclusion: The allegation has been substantiated by the findings described above. As a result of the analyses and evaluations performed and the licensee's actions in retorquing bolts where questions remained, it is unlikely that conditions exist which would reduce the margins for safe operation in regard to the installed equipment. Based on the CECo audit and the NRC's review, we believe the technical issues related to Morrison's activities have been properly identified and resolved. The licensee has committed to additional auditing of a wider sampling, and the NRC staff will continue to pursue QA program deficiencies which have been identified. A separate report is being prepared which will contain the enforcement sanctions resulting from unacceptable QC practices.

CATEGORIZATION OF ALLEGATIONS

CATEGORY 1 ITEMS - ITEMS OF SIGNIFICANCE WHICH MAY EFFECT UNIT 1 OPERATION

1. IMPROPER CORING AND DRILLING ACTIVITIES
2. VOIDS IN REACTOR PEDESTAL
3. 55 GALLON DRUM IN CONTAINMENT BASEMAT
4. DEBRIS IN CONCRETE
5. IMPROPER CONCRETE WORK IN THE SCREENHOUSE
6. MISALIGNED CONTAINMENT WALL
7. IMPROPER MASONRY WALL CONSTRUCTION AND POOR MORTAR QUALITY
8. INADEQUATE SECURITY
9. INADEQUATE CONCRETE VIBRATION
10. SECONDARY CONTAINMENT TEST EVENT
11. IMPROPER INSTALLATION OF HANGER SUPPORTS
12. AUXILIARY BUILDING WALL LEAKING
13. EXCESSIVE REACTOR BUILDING SETTLING
14. ACTS OF SABOTAGE IN 1979
15. WELDERS UNQUALIFIED OR NOT PROPERLY CERTIFIED
16. INADEQUATE TRAINING OF QUALITY CONTROL PERSONNEL
17. COVERUP OF DEFICIENCIES
18. NONCONFORMING MATERIAL
19. CONFLICT BETWEEN SPECIFICATIONS AND NRC REQUIREMENTS
20. FALSIFICATION OF TORQUE WRENCH CALIBRATION RECORDS

CATEGORY 2 - RESOLUTION REQUIRED (BUT NOT IMMEDIATE)

1. ADVANCED KNOWLEDGE OF NRC INSPECTIONS
2. ARCHITECT ENGINEERS INABILITY TO CLEARLY COMMUNICATE
3. IMPROPER INSTALLATION ACTIVITIES IN UNIT 2
4. IMPROPER MANAGEMENT ATTITUDE
5. INSTALLATION OF DAMAGED EQUIPMENT
6. EVENT RELATING TO UNIT 2
7. CONDITION OF UNIT 2

CATEGORY 3 - REFER TO LICENSEE; STATE; OSHA; OR OTHER AGENCY/ NO FURTHER INVESTIGATIVE ACTION REQUIRED

1. NRC INSPECTOR CONDUCT
2. IMPROPER INSTALLATION OF PIPING
3. INADEQUATE WORKER SAFETY
4. WASTE AT LA SALLE
5. DEFECTIVE CIRCULATING WATER PIPE
6. INSTALLATION OF PARTS NOT IN ACCORDANCE WITH PRINTS
7. LOOSE BOLTS ON BEAMS IN UNIT 2 TURBINE BUILDING
8. BULGE IN CONDENSER PIT CONCRETE WALL
9. ALCOHOL AND DRUG USE

EVALUATION REPORT ON THE ATTORNEY GENERAL OF ILLINOIS ALLEGATIONS FOR LA SALLE PLANT

STRUCTURAL ENGINEERING BRANCH

BACKGROUND

The allegations made by the Office of the Attorney General, State of Illinois, on March 24, 1982, in the matter of La Salle County Nuclear Generating Station, Units 1 and 2, (Reference 1) can be summarized as follows:

1. That thousands of drilled holes may have been cut through the reinforcing steel and that the potential degradation in structural quality may cause failure of the structures and systems.
2. That the concrete roof of the Off-gas building was actually 8 inches thick instead of 12 inches that the specifications called for.

On March 31, 1982, a meeting was held in Bethesda, Maryland. Participants included NRC staff members; representatives of the Commonwealth Edison Company; Sargent and Lundy; and a representative of the Attorney General, State of Illinois. Mr. Denton of NRR conducted the meeting and the applicant presented his response to the petition made by the Attorney General of Illinois. Discussions among participants ensued and a transcript of the entire proceedings has been taken (Reference 2). The applicant later made some comments and clarification on this meeting transcript (Reference 3). At the end of this meeting the applicant left us for reference a copy of the specification for concrete expansion anchor work and a set of 109 engineering drawings showing the number and locations of drillings through concrete (References 8 and 9).

On April 7, two staff members of the Structural Engineering Branch went to the La Salle plant site to observe and to gather information on practice of drilling holes through concrete elements. The staff also attended the meeting at Sargent and Lundy in Chicago on April 8. Preliminary findings of this trip were documented in a trip report (Reference 4).

Other reports by IE and Region III concerning the issue of hole-drilling and

the Off-gas Filter Building have also been sent to us for review (References 5, 6 and 7). We have also reviewed the final report on allegations regarding rebar damage and Off-gas Building roof thickness submitted by the applicant (Reference 10).

DISCUSSION

The applicant has kept a complete record of cored and drilled holes passing through concrete elements, this includes permanent records for all reported damaged rebars due to drilling. We have verified at the plant site several groups of drilled holes through the use of the set of drawings that have been provided to us and believe that the record of drilled holes is reasonably accurate. In spite of the fact that thousands of holes have been drilled and thousands of rebars have been hit, the actual damage is believed to be too small to affect the structural integrity of the plant. Furthermore, there are no holes cored completely through the primary containment walls.

We have reviewed the applicants' quality control procedures and documentation procedures for cored holes either passing through or partially penetrating concrete elements and for damaged reinforcing steel due to drilling operations for concrete expansion anchors. These procedures require accurate records of drilled and cored holes and damaged bars, are consistent with good engineering practice, and are therefore acceptable. Although recorded details concerning damaged bars in drilled holes prior to 1976 were insufficient in our opinion, the conservative approach of assessment by taking hit bars as completely cut compensates for this deficiency.

We looked into the method of engineering assessment performed by the applicant. We are satisfied that the applicant distinction between a "nicked" and "cut" bar was appropriate as implemented; in particular that partially cut bars were regarded as completely cut because the residual strength of a partially cut bar is uncertain and unreliable. We questioned the basis of selecting sample groups and panels for assessment and subsequently agreed that the sampling based on

density of holes was appropriate. We have also audited and spot-checked engineering calculations performed to assess the significance of cut bars and found them to be acceptable.

In regard to the thickness of the roof of the Off-gas Building, we have visited the building and witnessed some crucial field measurements. Based on this field data and on the reports of IE and Region III (References 5, 6 and 7) we believe that the thickness of the roof slab is definitely 12 inches and not 8 inches as alleged.

CONCLUSION

In conclusion, we confirm our earlier observation with the following findings:

1. The controls and engineering evaluation of the effect of drilled and cored holes were such that there is reasonable assurance that they will not result in unacceptable degradation of structural elements.

2. The roof of the Off-gas Building is 12 inches thick.

In view of the above we are of the opinion that the allegations filed by the Attorney General of the State of Illinois are without merit.

REFERENCES

1. "Request to Institute a Show Cause Proceeding and for Other Relief" transmitted through a letter dated March 24, 1982 from T. C. Fahner, Attorney General of Illinois to Secretary, U. S. Nuclear Regulatory Commission. Attachments include affidavits by E. Garrison and D. G. Bridenbaugh.

2. Transcript of the March 31, 1982 meeting held in Bethesda, Maryland, in the matter of Commonwealth Edison Company, La Salle County Nuclear Generating Station, Units 1 and 2. Attachments include: (a) Off-gas Building Roof Report, dated March 30, 1982, (b) Exhibits 1-8 at the meeting, and (c) Response to Petition made by the Office of the Attorney General, State of Illinois, in the matter of Reinforcing Steel Damaged during the Installation of Cored Holes and Concrete Expansion Anchors, La Salle County, Units 1 and 2, by Commonwealth Edison Company, dated March 31, 1982.

3. Comments and Clarification on Meeting Transcript, March 31, 1982, transmitted by Commonwealth Edison Company on April 22, 1982 to U. S. NRC.

4. "Trip Report - Visit to La Salle Plant and Meeting on Hole-Drilling and Cut Rebars in Concrete" by R. E. Lipinski and S. P. Chan, April 14, 1982.

5. "Assessment of the Off-gas Filter Building at La Salle Nuclear Station" by R. E. Shewmaker, April 8, 1982.

6. "Assessment of the Response by Region III to Allegations concerning the Off-gas Filter Building at the La Salle Station" by E. C. Gilbert, April April 16, 1982.

7. Region III Inspection Report No. 50-373/82-21 (DETP).

8. Sargent and Lundy: Specification for Concrete Expansion Anchor Work (Form LS-CEA) Rev. 0, September 23, 1976; Rev. 1, December 7, 1976, Rev. 2, November 29, 1978; Rev. 3, July 20, 1979; Rev. 4, September 7, 1979; Rev. 5, December 10, 1979; Rev. 6, February 13, 1980; Rev. 7, October 27, 1980; Rev. 8, May 13, 1981.

9. Sargent and Lundy: Engineering drawings showing locations of drilled holes and reinforcing steel bits. Partial list of CHS, RHS and RCS Series drawings.

10. Commonwealth Edison Company: Final report in response to petition made by the Office of the Attorney General, State of Illinois, in the matter of reinforcing steel damaged during the installation of cored and drilled holes and the matter of the Off-gas Building roof for La Salle County, Units 1 and 2, May 7, 1982.

Attachment C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

APR 27 1982

Docket No. 50-373

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the special safety inspection conducted by Mr. F. C. Hawkins of this office on March 24 and April 6, 1982. of activities at LaSalle County Station, Unit 1, authorized by NRC Construction Permit No. CPPR-99 and to the discussion of our findings with Mr. C. Schroeder and others at the conclusion of the inspection. This report also refers to the continuation of that inspection conducted by Messrs. F. C. Hawkins, S. P. Chan and R. E. Lipinski at the LaSalle site on April 7, 1982, and at Sargent and Lundy Engineers in Chicago, Illinois on April 8, 1982.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

No items of noncompliance with NRC requirements were identified during the course of this inspection.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you (or your contractors) believe to be exempt from disclosure under 10 CFR 9.5(a)(4), it is necessary that you (a) notify this office by telephone within ten (10) days from the date of this letter of your intention to file a request for withholding; and (b) submit within twenty-five (25) days from the date of this letter a written application to this office to withhold such information. If your receipt of this letter has been delayed such that less than seven (7) days are available for your review, please notify this office promptly so that a new due date may be established. Consistent with Section 2.790(b)(1), any such application must be accompanied by an affidavit executed by the owner of

the information which identifies the document or part sought to be withheld, and which contains a full statement of the reasons which are the bases for the claim that the information should be withheld from public disclosure. This section further requires the statement to address with specificity the considerations listed in 10 CFR 2.790(b)(4). The information sought to be withheld shall be incorporated as far as possible into a separate part of the affidavit. If we do not hear from you in this regard within the specified periods noted above, a copy of this letter and the enclosed inspection report will be placed in the Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

C. E. Norelius, Director
Division of Engineering and
Technical Programs

Enclosure: Inspection Report
No. 50-373/82-21(DETP)

cc w/encl:
Louis O. DelGeorge, Director
of Nuclear Licensing
R. Cosaro, Site Construction
Superintendent
T. E. Quaka, Quality
Assurance Supervisor
R. H. Holyoak, Station
Superintendent
B. B. Stephenson, Project Manager
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII
Mary Jo Murray, Office of
Assistant Attorney General

UNITED STATES ... COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-373/82-21(DETP)

Docket No. 50-373

License No. CPPR-99

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: LaSalle County Station, Unit 1

Inspection At: LaSalle County Station, Unit 1, and Sargent & Lundy Engineers in Chicago, IL

Inspection Conducted: March 24 and April 6-8, 1982

Inspector: F. C. Hawkins
March 24 and April 6-8, 1982

4/19/82

Accompanying Personnel: S. P. Chan
April 7-8, 1982

R. E. Lipinski
April 7-8, 1982

Approved By: C. C. Williams, Chief
Plant Systems Section

4/19/82

Inspection Summary

Inspection on March 24 and April 6-8, 1982 (Report No. 50-373/82-21 (DETP))
Areas Inspected: Special joint inspection conducted by IE Region III and NRR in response to alleged indiscriminate concrete drilling/coring which resulted in damage to embedded reinforcing steel. This inspection involved a total of 49 inspector-hours by one Region III inspector and two NRR representatives.
Results: No items of noncompliance or deviations were identified.

DETAILS

Persons Contacted

Commonwealth Edison Company (CECo)

R. Cosaro, Project Construction Superintendent
L. DelGeorge, Director of Nuclear Licensing
W. Garrigan, Supervising Staff Auditor
J. Gieseker, Project Construction Engineer
J. Harchut, Project Construction Engineer
*M. Morris, Structural Engineer
I. Netzel, Quality Engineer
T. Quaka, Site Construction QA Manager
*C. Schroeder, Nuclear Licensing Administrator
D. Shamblin, Staff Assistant

Sargent & Lundy Engineers (S&L)

*L. Dolder, QA Coordinator
*S. Kazmi, Supervising Design Engineer
*K. Kostal, Assistant Manager - Structural Department
*T. Longlais, Structural Engineering Department Head
*V. Reklaitis, Structural Project Engineer

Walsh Construction Company

M. Dougherty, QA Manager

Other personnel were contacted as a matter of routine during this inspection.

*Denotes those attending the exit interview on April 8, 1982.

Functional Areas Inspected

This inspection was conducted in response to alleged indiscriminate concrete drilling/coring which resulted in damage to embedded reinforcing steel. Information of the specific concerns were transmitted to the NRC by the Attorney General of Illinois in the form of a 10 CFR 2.206 request. This report addresses only the contention regarding damage to reinforcing steel during drilling/coring activities.

The scope of the inspection was twofold:

Phase I, which was conducted at the LaSalle site by Region III, investigated the programmatic approach to assure control of drilling/coring activities. Specifically, Phase I consisted of review of procedures, interviews with cognizant personnel, and review of quality records.

Phase II, of the inspection was conducted at S&L by Region III and NRR representatives. The expressed purpose of the S&L assessment was to verify proper and complete engineering disposition of field supplied data pertaining to damaged reinforcing steel.

A. Phase I

The scope of work for three site contractors was evaluated: H. P. Foley Co., Commercial Concrete Drilling and Sawing Co. (a Foley subcontractor), and Commonwealth Electric Co.

The contractual relationship between Foley and Commercial Concrete was reviewed. Commercial Concrete acted as the drilling/coring subcontractor to Foley for the period December 1977 through December 1979. During this period Commercial Concrete used the Foley procedures and the applicable S&L Specification to accomplish all drilling/coring work. For that reason, the programmatic appraisal of both companies was based on the review of the H. P. Foley drilling/coring program.

Additionally, the examination indicated that Commonwealth Electric was responsible for installation of temporary lighting and had commenced drilling activities on March 7, 1980. The review indicated that Commonwealth Electric had exclusively used carbide-tipped drill bits for the work. Past experience has shown that carbide-tipped drill bits are not capable of inflicting damage to reinforcing steel. Consequently, work performed by Commonwealth Electric is not considered relevant and was not included as part of this inspection.

1. Drilled Holes

Typically, drilled holes are provided for the installation of concrete expansion anchors which vary from 1/4" to 1" in diameter. The corresponding depth for holes of this size varies from 1-1/4" to 8", respectively. Drilled holes penetrate only partially into the concrete section.

To facilitate evaluation of the Foley drilling program, S&L "Standard Specification for Concrete Expansion Anchor Work" (Form LS-CEA) and H. P. Foley "Concrete Expansion Anchor Installation" procedure (No. WI-601) were reviewed. Each revision to both documents contained provisions to control drilling activities and identify reinforcing steel which may have been damaged during work operations. It is our assessment that the extent of control for drilling/coring work was commensurate with the level of activity in progress at all times during construction. The following revisions to each document were reviewed:

WI-601

Revision 0, December 7, 1976
Revision 1, November 21, 1977
Revision 2, January 31, 1978
Revision 3, May 8, 1979
Revision 4, October 23, 1979
Revision 5, August 6, 1981

Form LS-CEA

Revision 0, September 30, 1976
Revision 1, December 7, 1976
Revision 2, November 29, 1978
Revision 3, July 20, 1979
Revision 4, September 7, 1979
Revision 5, December 10, 1979
Revision 6, February 13, 1980
Revision 7, October 27, 1980
Revision 8, May 13, 1981

Foley Procedure No. WI-601 includes a daily report work form (No. HPFCo-016) on which any reinforcing steel which is damaged during drilling is reported. Following completion of form HPFCo-016, WI-601 requires that the form be forwarded to S&L for engineering review. This is the mechanism through which the necessary engineering assessment is accomplished for each piece of reinforcing steel which is damaged during concrete anchor installation. The specifics of any drilling damage to reinforcing steel is tabulated and plotted by S&L on Reinforcing Hit Schedule (RHS) drawings.

Approximately 200 of the Foley daily reports (No. HPFCo-016) were reviewed. Each was properly completed and in cases where reinforcing steel damage had occurred, proper notation of the damaged area was made on the form by the driller. Transmittal records of the forms to S&L for engineering evaluation were also verified.

2. Cored Holes

Cored holes typically range in size from 3" to 12" in diameter. In this application, cored holes pass completely through the concrete section to allow the passage of an electrical component (e.g., conduit). The routing of cored holes for electrical components is determined during the initial design phase (office routed) or in the field by the electrical contractor (field routed).

Office routed cores are designated on the structural design drawings and an engineering assessment is made of the effects of reinforcing steel likely to be damaged during the coring operation. This is accomplished prior to the release of the drawings for construction purposes. Field routed cores are requested by the contractor via a Field Change Request (FCR). The FCR is submitted to S&L prior to the coring operation. Approval of both the field routed core and the office routed core is based on an engineering evaluation by S&L. The core locations are indicated on the structural design drawings. It is important to note that both office and field routed cores are approved by the designer prior to the commencement of any coring operations.

3. Audit/Surveillance Activities

Three CECo audits of H. P. Foley concrete expansion anchor activities were reviewed. The audit numbers were 1-79-72, 1-80-22, and 1-80-45. The results of CECo surveillance inspection Nos. 79-237, 79-462, 79-571, 81-597, and 82-167 were also reviewed. Each audit and surveillance inspection was well planned, the findings well supported, and the resulting corrective actions appropriate.

In addition, a summary of Foley internal audit report Nos. 1 through 5 were reviewed. The summary indicated that the audits were conducted systematically and the findings were of substance.

4. Training

Records of twelve Foley training sessions on concrete anchor installation procedure No. HPFCo-WI-601 were reviewed. Each package consisted of a lesson plan and list of attendees. The training sessions were conducted in a timely fashion by qualified individuals.

5. Personnel Interviews

Interviews with H. P. Foley and CECo personnel were conducted to assess their knowledge of the Foley drilling/coring program and discuss any specific problems which they may have encountered during its implementation. The selected personnel were chosen because of their knowledge of past as well as present drilling/coring practices and policies. Interviews were held with the following personnel:

- Foley Labor Superintendent
- Foley Labor General Foreman
- Three Foley Concrete Drillers
- Foley Quality Assurance Manager
- CECo Quality Assurance Manager
- CECo Quality Engineer

Each individual categorically stated that, in his opinion, concrete drilling/coring by H. P. Foley and Commercial Concrete Companies had and is presently progressing in an orderly and well controlled manner. Each individual was knowledgeable within the scope of his assigned responsibilities.

B. Phase II

The documentation of the NRR assessment of S&L on April 8, 1982, is forthcoming and that report will be issued through their office upon its completion.

C. Conclusion

Based on the results of our review, we have concluded that (1) adequate procedures to control concrete drilling/coring are and have been in place at LaSalle; (2) these procedures are being successfully implemented; (3) the engineering disposition of damaged reinforcing steel by S&L was proper and complete; and (4) the completed drilling/coring represents no compromise to the structural integrity of the LaSalle plant structures. This issue is considered closed.

Exit Interview

The Region III inspector and NRR representatives met with licensee representatives during the conclusion of the inspection on April 8, 1982. The scope and conclusions of the inspection were summarized during the exit interview.

EVALUATION OF TECHNICAL REPORT

Report Title: Assessment of the Effects of Nicked Reinforcing Steel

Report No. SD & DD 77. Rev. 1

Author: H. Singh
Sargent & Lundy Engineers
Chicago, Illinois

Date of Report: November 1981

References:

1. "Probabilistic Analysis on nicking on Rebars at Clinton Power Station" by M. Amin and T. Y. Su, Sargent and Lundy Engineers, SAD-395, April 1982

2. "Influence on Nicking on Rebar Ductility at Byron and Braidwood Stations" by M. Amin and T. Y. Su, Sargent and Lundy Engineers, SAD-398, March 1982

Reviewer: S. P. Chan, NRC/SEB

Date of Review: June 15, 1982

SUMMARY

In the course of study to assess the effect of nicked reinforcing steel with the tungsten carbide drill bit during the installation of concrete expansion anchors, two major concerns were addressed on the effect of nicked steel bars; reduction in bar ductility and reduction in bar strength.

Tests at the Illinois Institute of Technology in 1978 demonstrated a ductility reduction to 2.2% in nicked bars with report mill ductility of 9.4%. Ever since it has been assumed that the bar ductility after nicking would be approximately 2%. Additional information on probabilistic analysis is contained in References 1 and 2 which indicated that more than 94% of all nicked bars at Byron and Clinton plants will have ductilities greater than 5.5%

This report presents the analytical approach for assessing the required bar ductility in both flexural members and shear walls at ultimate load. The analytical approach demonstrates that the minimum required bar ductility for flexeral members designed to: (a) ultimate loads in accordance with ACI 318 is 4.5% and (b) for impact loads is 5.5%. Appendix A to the report provides supporting calculations for these results. Design charts for flexural members and shear walls have been developed to assess the capacity of nicked reinforcing steel, showing no-nick zones of the concerned structure and the additional reinforcements required.

Appendix B to the report provides information of additional tests on the effects of nicked reinforcing steel and excerpts of a paper entitled, "Experience with Concrete Anchors on Northeast Utilities Construction Projects" presented at ASCE Specialty Conference on Construction Practices at Penn State University, September 16-18, 1981. The paper concludes that nicked reinforcing steel does not effect the ultimate strength of the bar.

DISCUSSION

The S&L report deals with the analytical approach used in calculating the required ductility in flexural elements at ultimate loads considering moment redistribution and other requirements of ACI 318. The ductility or ultimate strain is calculated from the moment/curvature relationship established by the proposed method of analysis and basic assumptions. The effect of strain hardening of steel reinforcement is considered only for determining the area where plastic hinge is formed and not for strength calculation. We feel that this analytical approach is reasonable and acceptable.

In case that the reduced ductility of a nicked bar is thought to be less than the calculated required ductility, the strength of the area should be re-analyzed by incapacitating the suspicious bar.

The report addresses, in Appendix B, the test results of rebars drilled with carbide bits. Four drill bit sizes (3/8", 1/2", 3/4", and 1") and five bar sizes (Nos. 5, 8, 11, 14, and 18) were used in the test program. Three specimens for each combination pair provided a total of 60 tests. A control specimen of each bar size was taken from the same bar as the other specimens. The control bars were not drilled but tensile tested for comparison purpose and baseline data. Drilling was done in a downward, vertical position with continuous pressure for not less than 15 seconds. The following observations had been made:

1. Reduction in ultimate strength due to drilling ranged from 0% to 2% for all bars except that of N5 rebars from 3% to 6%.

2. All bars broke above the specified yield and ultimate strengths. The control bar had an ultimate strength 25% greater than was required.

3. The deepest penetration into the body of the bar was approximately 1/8" while the defect diameters ranged from 7/16" to 9/16".

The test results reported in Appendix B have identified the largest nick which can be inflicted on reinforcing steel by a tungsten carbide drill bit. The utility recommendation, of 4/29, that 1/2" dia. x 1/8" deep defects should be acceptable.

These tests and observations, reported in Appendix B, were performed for Millstone III Project and were similar to those at the Clinton and Braidwood sites. Justification has been established that the test results are applicable to LaSalle.

CONCLUSION

The analytical model and assumptions used in this report for the assessment of bar ductility are reasonable and logical. The proposed method of analysis is consistent with the requirement of ACI Code 318 and is acceptable. The minimum required bar ductility for flexural members as calculated by this analytical approach (viz., 4.5% for ultimate non-impact loads and 5.5% for impact loads), are also acceptable and may be used as guidelines for estimating ductility requirements in case of moment redistribution.

We believe that the effects of nicked reinforcing steel by the tungsten carbide drill bit are negligibly small and will not have any significant effect on the structure integrity of reinforced concrete flexural and shear wall elements.

APPROVED BY

Attachment E

12-17-79 Date

MORRISON CONSTRUCTION COMPANY
LaSalle County Station
TORQUE WRENCH CALIBRATION CERTIFICATION

Form No. PC-139
Rev. 0 / 12-79

DATE 11/30/80 (1)

MCCo ID No. of Torque Wrench E-35-131 1/4" OR INCH POWERS

MCCo ID No. of Torque Wrench Tester BMTT 1067

TYPE OF TORQUE WRENCH: (X) Clicker Type ( ) Torqometer

TORQUE TESTER SETTINGS	WRENCH READINGS		ERROR	
	Clockwise	Counterclockwise	Clockwise	Counterclockwise
30 (2)	29 (3)	28.7 (4)	1 (5)	1.3 (6)
70	70	68	0	2
110	112	108	2	0
150	155	150	5	0
190	200	190	10	0

TOLERANCES

Clicker-Type

Clockwise = + 4%
Counterclockwise = ∓ 6%

Torqometer

Clockwise = + 2%
Counterclockwise = ∓ 2%

Instrument certification was accomplished by use of: Calibrated torque testers BMTT-1067 per procedure PC-31C Rev-0

Comments:

MCCO
Q.C. ACCEPTANCE
DATE 11/30/80 BY (9)
(8)

(10) TEST CONDUCTED BY

(11) WITNESSED BY

(12) P.237T

Attachment F

TECHNOLOGIES

World Headquarters
The Benson East
Jenkintown
Pennsylvania 19046
215-572-3000

June 28, 1982

Mr. I. N. Jackin
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL. 60137

Dear Mr. Jackin:

Accompanying this letter is a trip report covering a visit I made at the request of C. D. Sellers of U.S.N.R.C., Bethesda, Md., in connection with technical problems relative to the installation of bolts at the LaSalle County Nuclear Generating Station on June 25 and 26, 1982.

The report covers engineering discussion and certain recommendations and a conclusion I drew after examination of the available evidence relative to problems presented to me.

Yours sincerely,

A. Craig Hood, Manager
Advanced Metallurgical Products

ACH/pml

enc.

cc: C. E. Norelius - NRC
C. D. Sellers - NRC

JUL 1 1982

SPS TECHNOLOGIES

World Headquarters
The Benson East
Jenkintown
Pennsylvania 19046
215-572-3000

June 28, 1982

SUBJECT: Visit to U. S. Nuclear Regulatory Commission Region 3 and LaSalle County Nuclear Generating station on June 25 and 26, 1982

This report covers the visit I made to the above locations on 6/25 and 6/26, 1982 relative to a problem regarding tightness of bolts on the area in containment and outside containment of the Unit one reactor. I was contacted by C. D. Sellers of the U.S.N.R.C. on June 24 and asked to accompany him to U.S.N.R.C. on June 25. I had previously had discussions with Mr. Sellers on mechanical fastening technology at Bethesda. SPS Technologies is a major producer of precision mechanical fasteners and offers consulting advice relative to fastener design and installation upon request from industry and government agencies.

I was initially briefed on 6/25 at the Region 3 office. The problem is made up of two elements. The first refers to an allegation that records relating to calibration of torque wrenches used at the reactor site may have been in error over a period of several months. In addition, a check by one of the reactor inspectors of NRC of a valve attachment within a containment revealed two loose bolts that could be rotated with the fingers.

The second element related to the engineering implications of the calibration errors and the loose bolts. The first element will be dealt with by the NRC and Commonwealth Edison, the utility operating the nuclear station. I will only deal with the second element of the problem.

The reactor is currently on zero power and is planned to be started up for the first time preparatory to going to full power on Tuesday, June 29, 1982.

A meeting was held at the LaSalle Nuclear Station at 1:00 P.M. on 6/25. A copy of the list of those in attendance accompanies this report. Also, an outline of the meeting subjects and information relating to the valves in the containment area is enclosed.

SFS TECHNOLOGIES

World Headquarters
The Benson East
Jenkintown
Pennsylvania 19046
215-572-3000

SUBJECT: Visit to U. S. Nuclear Regulatory Commission Region 3 and LaSalle County Nuclear Generating Station on June 25 and 26, 1982

1. Motor Operator/Yoke Connection Tightness Check

Quality Assurance management from C. E. Co. said that all bolts in the subject valve attachments would be rechecked with a click torque wrench by turning the bolt head clockwise. I suggested that a dial type wrench be used and that the data for each bolt be recorded in terms of the torque it took to get the bolt head turning. Where the torque was too low, it could be brought up to the required minimum. Where it was too high, it could be backed off and retightened to the proper value.

C. E. Quality Assurance management agreed to do this for the 24 valves within containment immediately. They agreed to do it for the approximately 125 valves outside containment within 6 months.

It was stated that many of the bolts were ASTM-A193-B7 (Cr-Mo 125 ksi UTS).

C. E. Engineering management pointed out that many of the bolts were only called out to be "wrench tight" while others had no upper and lower tolerance for torque. After much discussion, C. E. agreed to establish clamping load requirements for these joints and specify an allowable upper and lower torque range for each bolt. Where certain bolts are not accessible with a torque wrench, C. E. Engineering and Quality Assurance indicated they would establish an engineering procedure andjustify their action. I suggested that they could use a snug fit and an angle of turn as an alternative in inaccessible areas.

2. LaSalle County Expansion Anchor Installation and Inspection Values

This element relates to the installation of expansion-type anchor bolts in concrete at various locations at the reactor site.

TECHNOLOGIES

World Headquarters
The Benson East
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Pennsylvania 19046
215-572-3000

SUBJECT: Visit to U. S. Nuclear Regulatory Commission Region 3 and LaSalle County Nuclear Generating Station on June 25 and 26, 1982

2. LaSalle County Expansion Anchor Installation and Inspection Values (cont'd.)

Mr. T. Longlais, head of Structural Engineering for Sargent and Lundy made a presentation on their study of anchor bolt seating torques for these bolts. Sargent and Lundy are the firm of architects and engineers used by C. E. Co. in the construction of the nuclear station. Studies made of the torque, holding power, and slip characteristics were presented. Copies of pp. 21-24 of a report on their findings along with Figs. 2.12, 2.23 and 2.24 were given to the group. A copy is enclosed. Their studies showed that once the expansion ring makes contact with the concrete inside the hole that further applied preload has little effect on the performance of the bolt. They did establish a preload tension above that expected to cause slip and these are used in anchor bolt installation. They also justified that a rechecking at 60% of installation torque is satisfactory and that bolts not achieving the 60% are taken to the original torque value.

I indicated that I had no problem with this engineering approach.

3. Torque Wrench Calibration Accuracy ($\pm$ 2% + $\pm$ 6%)

Concern was expressed by NRC personnel in the meeting that the errors discovered in the torque wrench calibration charts had contributed to improperly tightened bolts. It is possible to determine two things. First, the "worst case" of torque wrench effect on accurate torques can be established from the data and a recheck of the wrenches in question. C. E. Quality Assurance said they believed that maximum error was not greater than 11-12%. Second, this worst case condition can be applied to a range of torque values acceptable to C. E. engineers for each joint in

SPS TECHNOLOGIES

World Headquarters
The Benson East
Jenkintown
Pennsylvania 19046
215-572-3000

SUBJECT: Visit to U. S. Nuclear Regulatory Commission Region 3 and LaSalle County Nuclear Generating Station on June 25 and 26, 1982

3. Torque Wrench Calibration Accuracy (cont'd.)

question. C. E. Engineering management agreed to calculate the range of clamping loads and corresponding torque spread. Any joint believed to be outside the acceptable range will be retightened after consultation with NRC Region 3 inspectors and engineering personnel.

I indicated that this procedure was acceptable from an engineering standpoint.

Meeting on 6/26 at NRC Region 3 Headquarters

On the day following the meeting at LaSalle, NRC personnel from Region 3 and from NRC, Bethesda headquarters, met to discuss the results of the previous day.

It was the decision of NRC Region 3 personnel that they would request C. E. to evaluate all points in question as a result of the calibration problem prior to allowing the reactor to go above zero power. I concurred with this position.

In conclusion, after hearing the engineering statements of C.E. Co. and their consultants and other discussions with NRC personnel, I believe that all actions the licensee has taken and plans to take, as set forth in this trip report, are in the right direction.

A. Craig Hood, Manager
Advanced Metallurgical Products

ACH/pml
encs.

6/28/82

LASALLE COUNTY STATION
COMMONWEALTH EDISON COMPANY/NUCLEAR REGULATORY COMMISSION MEETING

JUNE 25, 1982 - 1:00 P.M.

PARTICIPANTS: Nuclear Regulatory Commission (NRC)
NRC Consultants
Commonwealth Edison Company
Sargent & Lundy Engineers

1. Motor Operator/Yoke Connection Tightness Check
2. LaSalle County Expansion Anchor Installation and Inspection Values
3. Torque Wrench Calibration Accuracy ($\pm$2% + $\pm$6%)

ATTENDANCE LIST

COMPANY: LA Salle DOCKET NO. 373

Date: 6/15/82 ☐ Pre-Inspection Conference ☐ Post Inspection Conference

NAME (Please Print)	TITLE (Please Print)	ORGANIZATION (Please Print)
R. M. Wescott	Project Inspector	NRC R-III
A.C. Hood	NRC Consultant	SPS Technologies
C D Sellers	Materials Engineer	NRC/NRR/MTEB
B.B STEPHENSON	PROJECT MANAGER	C.E. Co.
W.J. SHEWSKI	Mgr QA	C.E Co
T. E. QUAKA	SITE QA SUPT	CECo
E.E. FALB	SITE CONST. ENGR.	CECo.
RA BRAUN	SITE QA SUPERVISOR	CECo
WE VAHK	CECo Constr	CECo
M.J. MORRIS	Proj. Engineering	CECO
B.L. PANDIT	MECH PROJ ENGR	S&L
V. REKLAITIS	STRUCT. PROJ. ENG.	SARGENT & LUNDY
T. Conglais	~~HEAD~~ STRUCTURAL ~~ENGINEERING~~	" "
B. Shelton	PROJ ENG MGR	CECo
[illegible]	MECH PROJ ENG	S&L
RD Lanksbury	Reactor Inspector, Rgn III	NRC
D. TERAO	MECH. ENG	NRC (MEB)
Ernest Podolnick	Consultant	for NRC
D.L. Shamblin	Proj. Constr. Site Supt.	Commonw. Edison Co.

Rx Containment Motor Operated Valves
Tightness Check

Basic Requirements - Manufacturers Recommendation

Specific Tightness Requirements

1. Torque Value
2. Tightening Sequence
   a. G.E. - Atwood & Morrill
   b. Anderson Greenwood

Non-Specific Tightness Requirements

- Standard Mech. Practice (Wrench Tight)
  a. Anchor Darling Company
  b. Fisher Controls
- Standard Torque Chart Referral

Rx CONTAINMENT MOTOR OPERATOR/VALVE YOKE BOLTING TIGHTNESS REQUIREMENTS

VALVE NUMBERS	S&L SPEC.	MANUFACTURER	TORQUE SPEC.	TORQUE SPEC. REFERENCE
1B33-F023A 1B33-F023B 1B33-F067A 1B33-F067B	J-2500	G.E./Atwood&Morrill	110 Ft.-Lbs.	Instr. Manual Part IX Step 12
1B21-F016 1E12-F009 1E51-F063 1G33-F001 1G33-F100 1G33-F101 1G33-F102 1G33-F106 1WR179 1WR180	J-2938.01	Anchor-Darling	Wrenchtight	Instr. Manual - Confirmed w/John Chappell, A/DV Eng'g. Manager
1VP113A 1VP113B 1VP114A 1VP114B	J-2940	Fisher Controls	Wrenchtight	No Instr. Manual Reference; Info. Obtained From Glenn Hyatt
1B21-F001	J-2950.01	Anderson-Greenwood	75±10 Ft-Lbs.	Drawing #NO3-6492-500, Rev. B
1B21-F002			"	"
1B21-F005			"	"
1E12-F099A			"	"
1E12-F099B			"	"
1E51-F076			150±10 In-Lbs.	Drawing #NO3-6498-510, Rev. C

2.4 Phase D - Anchor Preload Relaxation Tests

2.4.1 Introduction

When a concrete expansion anchor is installed, a preload will be induced on the anchor as a result of torquing the bolt or nut. It is known that a portion of the preload in the anchor dissipates over a period of time after installation. The purpose of these single anchor relaxation torque tests was to investigate the loss of anchor preload over time (relaxation).

The tests were performed on single anchors of varying types, and diameters, installed at various embedded depths and with various torques in concrete and mortar (Types N & M). The specific testing requirements are outlined in Table 2.4.

2.4.2 Test Apparatus

Single anchors were installed in unreinforced concrete (no reinforcement within a minimum depth of ten anchor diameters) and in Types N and M mortar.

2.4.3 Procedure

Single anchors were installed in concrete and mortar in accordance with manufacturers' recommended installation procedures. Installation torques are shown in Table 2.4. The nut or bolt of the anchors was loosened 1/8 of a turn and then retorqued to its original position. The torque required to return the nut or bolt to its original position was then

recorded as a measure of remaining preload in the anchor. One anchor for each set of tests performed was tested with a load cell under the nut or bolt head to establish a torque-tension relationship.

The anchors were retorqued at intervals of 12 hours, 24 hours, 7 days, 14 days and 28 days after initial installation. The anchor load and the average anchor torque versus time were plotted for each set of tests. (Figures 2.23 and 2.24 show typical load and torque plots.)

2.4.4 Results

The results are represented by Figures 2.23 and 2.24. The loss of preload at the end of 28 days was as little as 13% for a 3/4" diameter anchor embedded in mortar and as much as 54% for a 1/2" diameter anchor embedded in concrete. Overall, it appears that less relaxation occurred for anchors embedded in mortar than for those embedded in concrete.

2.5 Conclusions

2.5.1 Phase A - Static Tension Tests on Single Anchors

The static tension tests on single anchors have provided a clear understanding of the anchor behavior under loading and the effect of various parameters on that behavior. It is noted that the prestressing of the anchor at the time of testing does not affect the ultimate load carrying capacity of the anchor.

2.5.2 Phase B - Cyclic Test on Anchored Plate Assemblies

The wedge, sleeve and shell type anchors tested in concrete and block walls exhibited insignificant anchor displacement when subjected to seismic or pipe transient loadings. It can, therefore, be concluded that anchors embedded in concrete can withstand cyclic loads up to 25% of manufacturer's ultimate capacity with a simulated OBE condition and 50% of manufacturer's ultimate capacity with a simulated SSE condition. It has been shown that anchors embedded in concrete block and mortar can withstand cyclic loads. The tests were conducted at load levels of 25% of the measured mean ultimate static capacity or greater.

It should be noted that anchor preload is not required for the anchors to withstand cyclic loading. The preload in the

anchors tested was generally not greater than 500 lbs. (0 preload) which is equivalent to tightening the nut or bolt approximately 1/8 of a turn after "hand" tight.

2.5.3 Phase C - Static Tension Tests on Anchored Plate Assemblies

The results of tests on a flexible base plate with four expansion anchors show that the prying action is of the order of 15-20 percent of the applied load. This increase is much lower than the expected increase in an assembly with regular steel bolts where the prying action force is calculated to be 110 percent. The reduction in the prying action force is due to the effective lower stiffness of expansion anchors installed in concrete.

2.5.4 Phase D - Anchor Preload Relaxation Tests

From the typical curves showing load or torque versus time (Figures 2.23 and 2.24), it can be seen that the anchor preload losses are most pronounced in the first 24 to 48 hours. However, it should be noted at this point that the relaxation phenomenon should not be of great concern when viewed in light of the cyclic test results which showed that preload is not required to withstand cyclic loading.

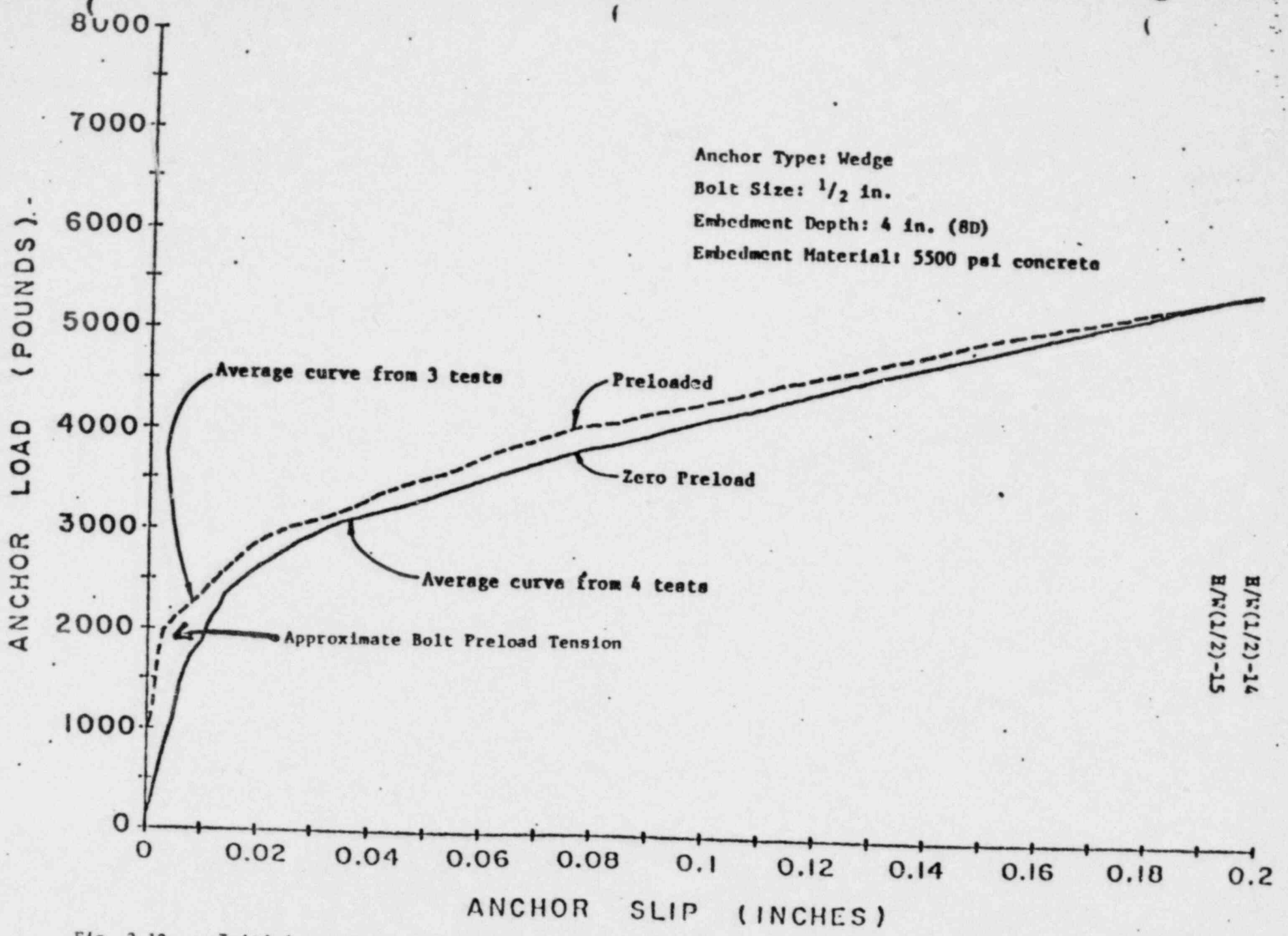

Fig. 2.12 Initial portion of load slip behavior for anchors with different bolt preloads

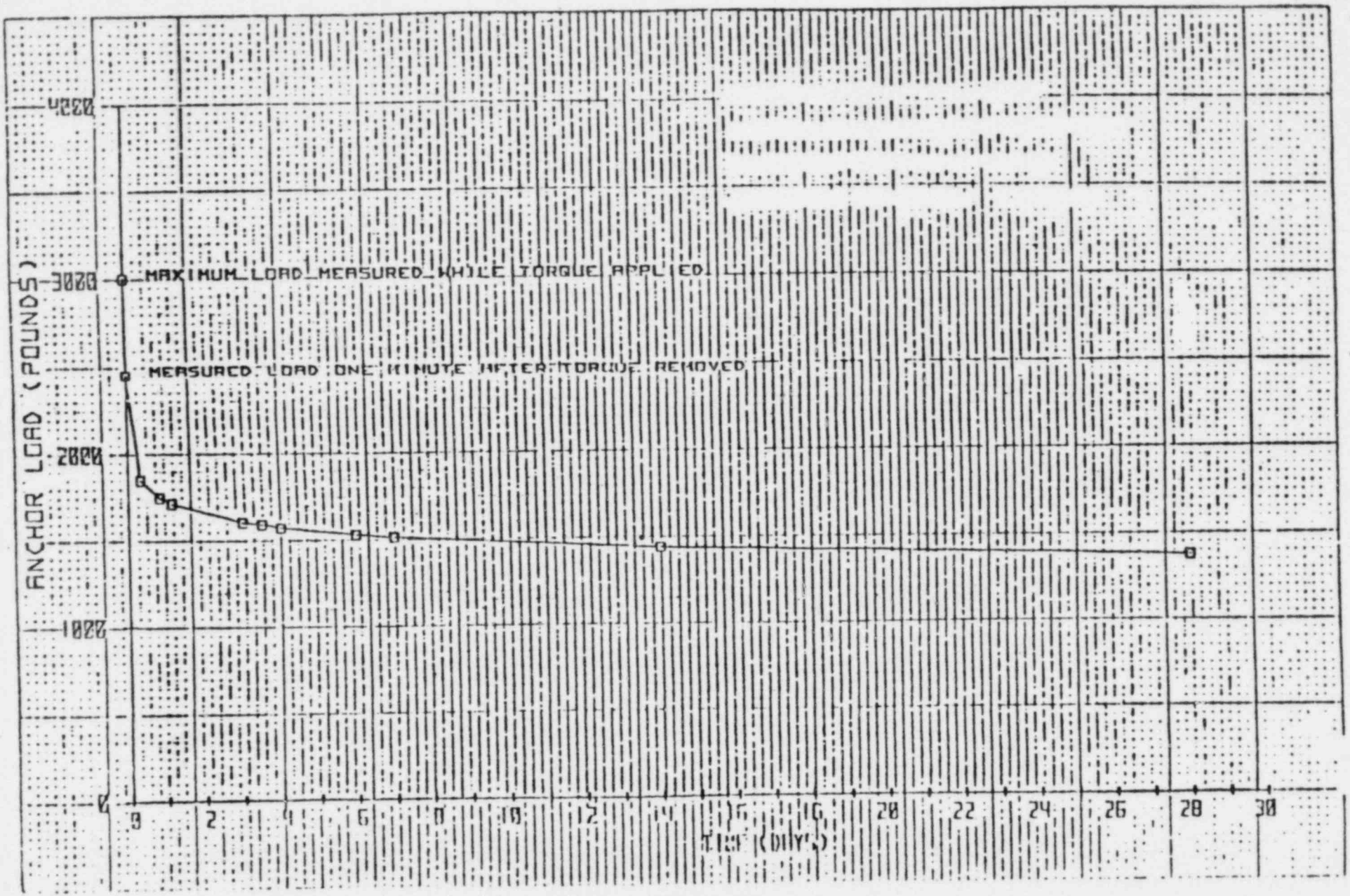

FIGURE 2.23
ANCHOR PRELOAD VS. TIME

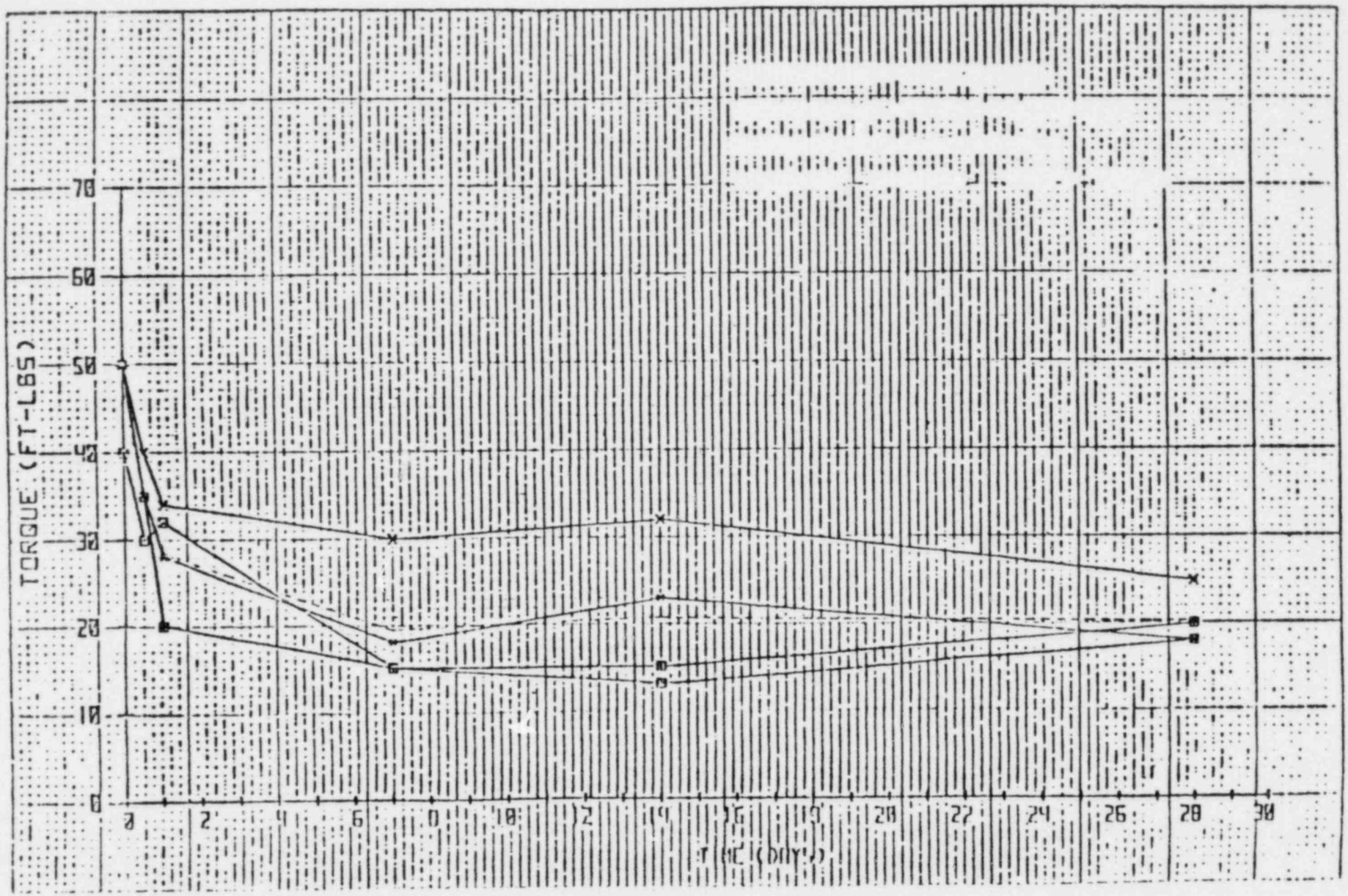

FIGURE 2.24
TORQUE VS. TIME

E. C. Rodabaugh Associates, Inc.
4625 CEMETERY ROAD • HILLIARD, OHIO 43026
614/876-5719

June 26, 1982

Mr. Dave Terao
Mail Stop P-522
Mechanical Engineering Branch
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Meeting at LaSalle Plant, June 25, 1982 on Technical Aspects of Torque Wrench Calibration

Dear Dave:

A copy of the Agenda is included as Enclosure 1. My comments concerning each Agenda item follow.

1. Motor Operator/Yoke Connection Tightness Check

Roger Lanksbury (NRC, Region III) found (6/18/82) that two nuts on a motor operator/yoke connection were "finger tight". (A possible reason is that it was necessary to remove the operator for clearance during construction and all of the nuts were not re-tightened during replacement of the operator.) The Applicant has committed to checking all safety-related motor operated valves inside containment (24 valves) immediately and all safety-related motor operated valves outside containment within 6 months.

Enclosure 2 is a listing of motor operated valves inside containment. It includes a column headed "Torque Spec." Apparently, this information was obtained in the last week or so and, prior to then and presumably during the time of the alleged lack of accurate torque wrench calibration, there was no specified torque for these particular bolts when, or if, they were loosened during plant construction and then re-tightened. A discussion of the technical significance of the torque wrench accuracy, if in fact a torque wrench was used in the re-tightening, is irrevelant.

Considerable discussion occurred at the meeting as to what was meant by "Wrenchtight". In my opinion, "Wrenchtight" is an adequate instruction to any competent workman. However, the Applicant agreed to put "A value" in place of "Wrenchtight". Once that specified torque is established (as it is for two of the four valve manufacturers as shown in Enclosure 2), the technical significance of the torque wrench accuracy becomes relevant.

From a technical standpoint, torque during bolt tightening is a method of trying to achieve an appropriate clamping force. As brought out by Mr. Hood of SPS Technologies, the relationship between torque and clamping force, for a given nominal bolt geometry (size, threads per inch, thread tollerances, etc.) can vary widely depending upon the coefficient of friction. My own experience (on flanged bolt bolting) indicates that if one first calibrates torque versus clamping force using a reasonably good thread and other mating surfaces lubricant, then if one continues to use that same lubricant and nominal bolt geometry, the clamping force can be predicted from the torque within a scatter of ±30% with about 90% confidence. However, if the lubricant is not controlled, then from what I have read it appears that the clamping force, for a given torque, could easily vary by a factor of 4 or more.

Fortunately, the safe operation of bolted connections of the type under consideration are not very sensitive to preload clamping forces. To the extent that loads are cyclic, it is desirable to have the preload higher than those loads so that the bolt stresses will remain essentially constant and fatigue of the bolts will not occur. In addition, even if no loads are anticipated in paper calculations, most power plant equipment operate in an environment of at least low-level vibration and bolts should be tightened sufficiently so that the nuts do not vibrate loose in service. These considerations establish minimum desired preloads.

The maximum preload may be significant if, by any chance, prying effects are not properly evaluated in calculating loads. However, if loads are appropriately calculated and bolt sizes are selected to accommodate those loads, then maximum preload is not significant in a loading

sense. Some bolting materials such as SA 193 Grade B7 can be susceptible to stress corrosion cracking and excessive preload may contribute to such cracking. (A perhaps necessary co-condition is lack of adequate heat treatment control in manufacturing the bolts.)

In summary, a preload not less than calculated loads is desirable and, regardless of calculated loads, the bolts should be tight enough to prevent loosening due to vibration. Overtightening could be a problem if stress corrosion cracking is a possibility. With controlled lubrication, a specified torque could be useful. Without controlled lubrication, "Wrenchtight" is probably better than specified torque. In view of the large variations of clamping force for a given torque, torque wrench accuracy of ±20% is adequate.

2. LaSalle County Expansion Anchor Installation and Inspection Values

Sargent and Lundy's description or their basis for setting installation and inspection specified torques for concrete anchors (specifically, Hilti Wedge Anchors) indicates to me that torque wrench accuracy of 20% is adequate.

3. Torque Wrench Calibration Accuracy (±2% + ±6%)

The ±2% is the calibration fixture accuracy and the ±6% is the limits within which a torque wrench must be accurate as determined by use of the calibration fixture. In principle, the torque wrench accuracy is ±8%. Based on personal use of torque wrenches some years ago, I suspect that in "tight" places where the torque wrench can not be pulled on in the same manner as in the calibration, additional inaccuracies could occur. However, in view of the wide scatter in torque versus clamping force without controlled lubrication, whether the torque wrench is accurate to ±8% or ±20% (as used in tight places) is a relatively trivial consideration.

Incidentally, there is a good discussion of preloading of bolts in flanged joints in the ASME B&PV Code, Section VIII, Division 1, Appendix S. That Appendix gives an equation that gives an indication of what "wrenchtight" means in terms of clamping stress in the bolts.

$$S = 45000/\sqrt{d} \qquad (1)$$

where

S = probable bolt stress developed manually when using standard wrenches

d = nominal diameter of bolt.

Equation (1) was developed by E. C. Petrie in 1937. It is limited to $d \geq 1/2"$ and probably to $d \leq 2"$; well lubricated threads and other mating surfaces; NC threads below d=1", the 8 thread per inch series above d=1".

Yours Very Truly

Everett

E. C. Rodabaugh

Enclosures 1 and 2

cc S. E. Moore

Commonwealth Edison
LaSalle County Nuclear Station
Rural Route #1, Box 220
Marseilles, Illinois 61341
Telephone 815/357-6761

Attachment H

July 8, 1982

Mr. James G. Keppler
Regional Administrator
Directorate of Inspection and
Enforcement - Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

SUBJECT: LaSalle County Station Units 1 and 2
Review of Morrison Construction Company Records

Dear Mr. Keppler:

The purpose of this letter is to provide you with copies of the Quality Assurance and Project Construction reports of the subject review.

Commonwealth Edison believes that these reports demonstrate that, although various deficiencies have been observed, the adequacy of construction of the plant is satisfactory.

If there are any further questions in this matter, please immediately contact C. W. Schroeder, Nuclear Licensing Administrator for LaSalle at (312) 294-3962 (office) or (312) 329-1087 (home).

TOM QUAKA (HOME) 312-668-5089

Very truly yours,

Cordell Reed

Cordell Reed
Vice-President

CR/CWS/djp

Q.P. FORM 18-1.2
DATE 8-4-81

Commonwealth Edison Company

QUALITY ASSURANCE MANUAL

AUDIT REPORT

G.O. Audit of Morrison Construction Company
LaSalle Project

Type Audit: ☐ Program Audit ☐ Product Inspection Point
☐ Records ☒ Special

To: J. Hamilton, Morrison Construction Company - Project Manager

Project LaSalle Visit Date 6/29 - 7/8 1982 Report Date 7/8/82

System N/A Component Identification N/A

Material Description N/A

Vendor Morrison Construction Co. Location LaSalle

Subcontractor N/A Location N/A

Contacts See Report

P.O. No. 181110 Spec. No. J-2530

Recommended Inspections: 6 mos 3 mos 1 mo
Other: As Scheduled

Notes:

See Report

Auditor K. J. Hansing Date 7/8/82
K. J. Hansing - Lead Auditor
Reviewed [illegible] Date 7-8-82

cc: Manager of QA
[redacted]
Manager of Projects

Director of QA (Engr-Constr)
Site Constr. Supt. or Proj. Engr.
Site Quality Assurance
Project Manager
Project Engineering Mgr.
(List others as required)
Auditee
Lead Auditor

AUDIT REPORT FOR

MORRISON CONSTRUCTION COMPANY

LASALLE PROJECT

JULY 8, 1982

PURPOSE

The purpose of this audit was to evaluate Morrison Construction Company's (MCCo) calibration practices and procedures, various quality control records, records control and the Quality Control review functions. Additionally, the audit, inspection and surveillance activities in the area of MCCo calibration were evaluated to determine if required reviews were performed and to assess the credibility of their reviews and associated documentation.

SCOPE

This audit examined the following areas:

1. CECo audits of MCCo calibration activities
2. CECo auditor qualifications
3. Credibility of CECo audits
4. CECo surveillances of MCCo calibration activities
5. Authorized Nuclear Inspector's involvement in MCCo calibration activities
6. MCCo audits of MCCo calibration activities
7. NRC involvement in MCCo calibration activities
8. CECo or MCCo NCR's generated which address calibration activities
9. Requirements and performance of Quality Control reviews
10. Dispositioning of out-of-calibration or damaged calibrated tools and equipment.
11. A broad scope review of other documentation activities to assess whether the types of documentation problems identified in the torque wrench calibration activities existed in other areas
12. Training, skills, and qualifications of MCCo personnel performing calibration functions
13. MCCo review of calibrations performed by others for them
14. Records control
15. Accountability of records
16. Performance of calibrations
17. Calibration procedure adequacy
18. Storage and maintenance of calibrated tools and equipment
19. A review of all calibration records not recently reviewed by the NRC as part of the investigation.

AUDIT TEAM

The G.O. QA audit team consisted of the following personnel:

Principal Team Members

D. A. Brown	Auditor	QA Supervisor-Braidwood
S. M. Jaquez	Auditor	QA Engineer-Braidwood
S. J. Reutcke	Auditor	QA Engineer-LaSalle
R. D. Vine	Auditor	QA Engineer-LaSalle
R. E. Waninski	Auditor	QA Engineer-LaSalle
K. J. Hansing	Lead Auditor	QA Supervisor-Byron
G. F. Marcus	Management Observer	Director Quality Assurance (Engr/Constr)

Others

G. M. Maksimuk	Auditor-Data Taker	QA Engineer-LaSalle
A. M. Montalto	Auditor-Data Taker	QA Inspector-LaSalle
E. A. Kram	Auditor-Data Taker	QA Engineer-Braidwood
R. F. Smeets	Observer	QA Engineer-LaSalle
M. J. Wendell	Observer	QA Engineer-LaSalle

Auditing activities conducted by the data takers and the LaSalle personnel were performed under the direction of a principal team member and were reviewed by an off-site member of the audit team.

PERSONNEL CONTACTED

The following personnel were contacted during the course of this audit:

M. Wherry	MCCo	QC Supervisor
D. Kanakares	MCCo	QC Inspector
D. Kozlowsky	MCCo	QC Inspector
P. Granby	MCCo	QC Inspector
B. Hamilton	MCCo	QC Inspector
B. Angell	MCCo	QC Inspector
R. McClosky	MCCo	QC Inspector
T. Harrington	MCCo	(Former) QC Inspector
K. Hamilton	MCCo	Project Manager
B. Balz	MCCo	Head Doc. Clerk
J. Terry	MCCo	QC Inspector
J. Bitner	MCCo	QC Inspector
D. Shamblin	CECo	Proj. Constr. Super.
A. Patak	Continental Insurance	MCCo. ANI
W. Caldwell	Hartford Insurance	CECo's ANI

ENTRANCE MEETING

An entrance meeting was conducted on June 29, 1982 with the following personnel in attendance:

*K. J. Hansing	CECo QA	Lead Auditor
*R. Waninski	CECo QA	Auditor
*S. M. Jaquez	CECo QA	Auditor
*S. J. Reutcke	CECo QA	Auditor
D. L. Shamblin	CECo PCD	Site Proj. Const. Supt.
D. A. Brown	CECo QA	Auditor
*K. J. Hamilton	MCCo	Project Manager
*M. Wherry	MCCo	QC Supervisor
*W. E. Vahle	CECo PCD	Lead Mechancial Engineer
R. A. Braun	CECo QA	QA Supervisor
R. D. Vine	CECo QA	Auditor
*T. E. Quaka	CECo QA	Site QA Superintendent
*D. J. Skoza	CECo PCD	Engineer
D. J. Kanakares	MCCo	QC Inspector
G. F. Marcus	CECo QA - GO	Management Observer

The scope of the audit was discussed at this time.
* denotes present at exit meeting.

EXIT MEETING

An exit meeting was conducted on 7/8/82. The audit results were discussed and acknowledged by those present. The following personnel were present at the exit meeting in addition to the personnel indicated above:

A. R. Huffman	MCCo	QA Engineer
R. T. Rose	CECo PCD	Lead Struct. Eng.
J. J. Maley	CECo	Manager of Projects

OVERALL AUDIT ASSESSMENT

It is the conclusion of this audit team, based on the extensive review of MCCo quality records, that the documentation problems originally identified by the NRC are limited to calibration records. It was further concluded that the duplicating of some signatures and data did not appear to be an attempt to falsify quality records but rather was a time saving measure used to handle the numerous, repetitious calibration records as they viewed the calibration functions as a go, no-go type check.

Relative to the adequacy of the surveillance and audit activities, our evaluation indicates that many qualified individuals had examined this area and that without being specifically attentive to looking for record alterations they would not have identified this as a quality issue. Examination of the overall surveillance and audit process, however, does indicate that our auditing methodology involving large numbers of similar records needs to be changed to provide that larger samples be taken to achieve a better assessment. As a result, Quality Assurance has committed to implement this change.

Evaluation of the reports where torque wrenches were found to be out of calibration or had questionable calibration data or records were deemed to have no adverse impact on plant construction. This is based on the records reviewed and the results of independent inspections of bolt torquing by the Independent Testing Agency involving Morrison. The inspection, conducted by CECo.'s Independent Testing Agency, examined 6,756 Morrison installations between 1977 and June 1982. Only 81 of the 6,756 installations were rejected for low torque values during the 6 year period. The reject rate of less than 1.2% supports the conclusion that torque wrench calibration was not a problem and helps explain, at least in part, why this calibration documentation matter had not surfaced.

In addition, examination by Quality Assurance of follow-up checks conducted by Project Construction of other calibration areas has not identified any cases where the installed condition of Unit #1 equipment appears questionable.

SUMMARY

1. Assessment of Audit Questions 1-12

In order to ascertain if there were any actions which could have discovered the "deficiencies" in Morrison calibration records which are the subject of this audit, the following areas were investigated.

1) Adequacy of site QA audits & surveillances
2) Involvement of CECo & MCCo ANI's
3) Trends of applicable CECo & MCCo NCR's
4) Adequacy of MCCo internal auditing
5) Past NRC site inspection activities
6) Adequacy of CECo G.O. QA audits

The effort was directed at determining if this problem went undetected due to oversight by many parties or a widespread problem not adequately addressed. The results in these areas can be generalized as follows:

(1) MCCo's ANI and CECo's ANI did not document evidence of examining calibration reports as this was their normal practice in the past.

(2) Region III Inspections (57 in 1981 & 1982) did not uncover any problems with MCCo quality control records. Five of the fifty-seven site visits involved review of MCCo quality records. One of these visits involved an NRC investigation of an allegation of lack of control of quality records.

(3) No CECo NCR's have had to be issued for concerns relating to MCCo calibration records. Therefore, no trends were overlooked.

(4) No MCCo NCR's have had to be issued for concerns relating to calibration records. Therefore, no trends were overlooked.

(5) Morrison internal audits of calibration activities have been extensive and did not identify any calibration record problems.

(6) CECo Site QA surveillance acitivites consisting of 27 calibration surveillances, which specifically covered thirteen calibration records since 1979, were found to have adequate coverage and frequency.

(7) Twenty-four Site CECo QA audits were done on schedule with qualified auditors and recorded objective evidence is sufficient to support the conclusions of those audits. Of the audits conducted since 1979, nine covered site calibration activities.

(8) G.O. audits addressed the areas of Morrison calibration activities and did not discover similar calibration documentation problems.

(9) Site CECo QA examined 64 MCCo calibration reports during its surveillances and audits since 1979.

(10) None of the Morrison calibration documents found deficient in the NRC investigation were specifically examined in any CECo audit or surveillance.

In summation, many qualified individuals examined the calibration documents as well as many other MCCo quality documents in accordance with quality auditing and surveillance plans; however, by the nature of the sampling process inherent in the audit and surveillance process they did not specifically discover the calibration record deficiencies. Furthermore, without being sensitized to look for record alterations or repetition either by allegations, specific training and experience or as a result of previous problems, it is doubtful that the MCCo calibration document type problems would be identified in the normal audit and surveillance process.

2. Assessment of audit questions 13-16

In order to determine the adequacy and accuracy of Quality Control reviews of quality records, the following record types were examined:

a. Calibration records
b. Material procurement documents
c. Material receipt records
d. Installation documentation/travelers
e. NDE reports
f. NCR's
g. Test reports
h. Procedures and work instructions

An evaluation was made of the adequacy of the Quality Control reviews for the above listed record types by examining the Quality Assurance manual and procedure requirements and by interviewing the Quality Control personnel who perform the reviews.

It was found that the requirements and meaning of the Quality Control reviews were not clearly delineated in the procedures for several of the record types. As a result of interviewing the MCCo Quality Control personnel performing the actual reviews, it appears that these Quality Control personnel are knowledgeable in how to do the reviews and overall are doing adequate technical reviews. The lack of delineated requirements for Quality Control reviews is identified as a deficiency in Attachment A of this report.

To verify accuracy of Quality Control reviews, a sample of each above listed record type was examined. With the exception of calibration records all of the record types reviewed were found complete and acceptable. The deficiency relating to calibration records is identified in Attachement A of this report.

To determine proper disposition of out-of-calibration or damaged calibrated tools and equipment, MCCo NCR's were reviewed and compared to the MCCo Deleted Instrument Log and Hold Tag Log entries involving calibrated tools and equipment. Through this review it was determined that MCCo.'s Quality Assurance Manual requirement, Section 10.10, was not being consistently followed in that nonconformances were not generated each time a calibrated tool or piece of equipment was suspect and put on hold. The need for issuance of an NCR for suspect equipment became a requirement on October 8, 1979. MCCo Procedure PC-31 does not include this requirement and thus appears to be a contributory factor in the inconsistent issuance of NCR's for suspect calibration tools and equipment. This deficiency is identified in Attachment A of this report.

3. Assessment of Audit Questions 17-22

A review of all MCCo calibration reports, not previously reviewed by NRC, was performed to determine the types and extent of documentation problems. This review identified a generic problem relating to reproduction of signatures and data and is identified in Attachment A of this report.

Also, in an effort to determine if the problems found in the calibration area were present in other MCCo Quality Control documentation, a review of the following documents was performed:

a) Receiving inspections
b) Weld data, and traveler packages
c) As-Built drawings with checklists (small bore)

d) Component support checklists
e) Mechanical revision directives
f) NCR's
g) NCR Log
h) Final line walk reports
i) Pressure test reports
j) ASME Code Data Reports (CECo Supplied)
k) NDE Reports
l) Quality Control surveillances
m) On-Site audits
n) Purchase requisitions
o) Welder qualifications
p) Quality Control inspector qualifications
q) Calibration personnel certifications

This extensive review found only isolated instances of the use of white-out, incomplete information and stamping of Quality Control approval signatures. Also, the isolated cases of white-out identified were judged to be corrective actions and not an attempt to falsify. Furthermore, the use of white-out on quality documents was not specifically prohibited at the site until February 1982 when a letter to this effect was sent to all site contractors as a result of a recent NRC penalty assessment against the Zimmer Project. As a result, it is the conclusion of this audit that the documentation problems only involve the calibration area and the quality control document review activities of a specific Quality Control Inspector. Also, one case was identified during the review of Quality Control Inspector qualifications where the certification of an inspector was not signed but otherwise appeared in order. This deficiency is identified in Attachment A to this report.

Since most of the problems identified in the calibration area were traceable to one Quality Control individual, an expanded review was performed of other records processed by that person. A selection of records from the following record types reviewed by that MCCo Quality Control Inspector were examined:

a) Receiving and storage inspections
b) NCR's
c) Installation records
d) Receiving reports
e) Heat treatment records
f) Purchase requisitions
g) Pressure test memoranda & test results

This examination found no repetitious occurrences of the documentation problems noted in the calibration area. This further supports the audit conclusion that the generic documentation problems are isolated in the area of calibrations.

4. Assessment of Audit Questions 23-26

A review of training and qualification records for the millwrights and fitters performing calibrations indicated that all were properly trained and the training was documented. The training appeared adequate and further the required skills were within craft capability. Eye examinations of Quality Control Inspectors performing calibration activities were reviewed to assure they had required vision capabilities and all were found acceptable. Additionally, the millwrights and fitters performing calibrations were eye tested for near and far vision acuity and all passed. An examination of the areas where calibration is performed was conducted to assure that adequate lighting was present; all areas were found acceptable.

A review of MCCo's handling of calibrations performed by the Operating Station Maintenance Department indicated that MCCo does not have adequate guidelines to assure review of test results for acceptability of the calibration. This deficiency is identified in Attachment A to this report.

5. Assessment of Audit Questions 27-33

MCCo has established and implemented measures to control access to the documentation vault by using a listing of personnnel authorized to remove documents from the vault and requiring logging of documents being taken in or out and identification of the person involved. Also, since June 29, 1982, a security guard has been posted at the vault to countersign issuance of documentation, monitor the review of such issued documentation to assure no alterations are made and assure that Unit #1 and common documentation with Unit 2 is only reviewed in the presence of the guard. Additionally, all Unit #1 documentation in the vault is secured with locks on the file cabinets. This special security arrangement is to be maintained until conclusion of the NRC investigation.

Prior to the actions taken on June 29, 1982, the CECo Project Construction Superintendent had verbally communicated to the MCCo Project Manager, on June 24, 1982 that actions should be taken to insure that no documentation would be altered. This requirement was communicated to the Quality Control personnel verbally on June 24, 1982 and again on June 25, 1982 in the form of written correspondence from the MCCo Quality Control Supervisor to the Quality Control personnel. During the week of June 21, 1982 when the MCCo Quality Control Inspector involved in the calibration documentation problem became aware of the fact that duplicated signatures were unacceptable on calibration reports, he decided to go back and initial those calibration reports he had issued with his signature duplicated. This initialing of signatures which occurred for about two hours ended on June 24, 1982. It was the auditor's opinion that the addition of the initials was done to correct an omission as opposed to falsifying records.

In the area of records accountability this auditor determined that MCCo.'s logging systems are adequate to verify accountability of records. However, there was no documented evidence that MCCo has performed reviews to determine if documents are missing . Also, current procedures do not specifically provide instructions for replacing lost records. This deficiency is identified in Attachment A to this report.

6. Asessment of Audit Questions 34-38

A review of the storage and maintenance of calibrated equipment and tools noted that they are kept in an orderly manner and in locked cabinets. Observance of millwrights and fitters in the performance of calibrations indicated they were knowledgeable and competent in their jobs. One procedural deviation was noted with the calibration of torque wrenches and is identified in Attachment A of this report.

The following procedural deficiencies were identified regarding calibrations:

a. Acceptance criteria and calibration points are not stated for each type of tool or instrument.

b. Not all calibrated equipment is addressed in procedure PC-31.

The specific deficiencies are identified in Attachment A of this report.

For the majority of the calibrations witnessed, the procedure and actual performance of the calibrations was determined to be acceptable.

Upon interviewing personnel performing calibration and reviewing calibration histories it was concluded that, in instances where the actual test data was extremely close to the established test reading, the actual values may not have been recorded in all cases; rather, the generic test point values were recorded. In other words, a go, no-go check was performed. The interviews did, however, indicate that out-of-tolerance calibration readings were identified. The deficiency relating to recording of actual data is identified in Attachment A to this report.

Response

Morrison Construction Company is requested to respond in writing to the Findings and Observations listed in Attachment A to this report by July 23, 1982. The response must include the following:

1. Action taken to correct the deficiency
2. Action taken to prevent recurrence
3. Data for full compliance

Please address your response to:

Commonwealth Edison Company
G. F. Marcus
Director of Quality Assurance (Eng./Constr.)
P.O. Box 767
Chicago, IL 60690

With copies to:

Commonwealth Edison Company
W. J. Shewski
Manager of Quality Assurance
P.O. Box767
Chicago, IL 60690

and

Commonwealth Edison Company
K. J. Hansing
QA Supervisor - Byron Station
Byron, IL 61010

ATTACHMENT A

FINDING #1

Contrary to 10CFR50 Appendix B Criteria XVII the Quality Control final review and acceptance of calibration reports was not adequately implemented to assure complete and acceptable calibration records.

Discussion

An examination of completed and accepted calibration reports identified the following problems:

a) Wrench type was not indicated on torque wrench #10894 calibration reports dated March 16, April 3 and September 25, 1981.

b) Quality Control review of torque wrench #97366 calibrated on July 31, 1981 was not documented.

c) Error measurements were not indicated on the report for torque wrench #97366 calibrated on January 2, 1981. The error values, however, were attainable from information on the report and found acceptable.

d) Test data was not recorded for pressure gauge #PG-31 calibrated on June 8, 1982.

FINDING #2

Contrary to MCCo Quality Assurance Manual Section 10.10, nonconformance reports have not been generated for disposition of each calibrated tool or instrument found suspect.

Discussion

A review of MCCo's Deleted Instrument Log, Hold Tag Log and NCR's indicated the following torque wrenches as damaged or out of calibration but records of NCR's for each was not available:

a) Torque wrench #4292 - Hold Tag #1106 - Torque wrench was scrapped instead.

b) Torque wrenches #15758 and #10468 - hold tag #1223 - Torque wrenches needed to be repaired. (The Hold Tag Log showed item was repaired and the Hold Tag removed.)

c) Torque wrench #E40468 - hold Tag #1171 - Torque wrench was damaged while being used. (Hold Tag Log showed item was repaired and the Hold Tag removed.)

Other instances of hold tags being issued on calibrated torque wrenches or torque screwdriver without references to applicable NCR's are as follows:

Hold Tags #1131, 1132, 1133, 1134, 1108, 1129, 1092, 1103.

Furthermore, a review of procedure PC-31 Rev. 3 dated July, 1977 indicates that the requirement of Section 10.10 is not addressed and thus could be a contributing factor in the inconsistent implementation of NCR's for suspect calibrated tools and equipment.

FINDING #3

Contrary to 10CFR50 Appendix B Criteria XII, MCCo has not established and implemented adequate guidelines for receipt inspection review of calibration reports as to acceptability of calibrations performed by outside agencies.

Discussion

Five calibrations were accepted for the torque wrench tester where the tester was out of calibration. CECo NCR #599 has been issued for engineering disposition.

FINDING #4

Contrary to 10CFR50 Appendix B, Criteria XII and V, MCCo Procedure PC-31 has the following deficiencies:

a) Not all calibrated tools & equipment used by MCCo are covered by this procedure.

b) The procedure does not address acceptance criteria for all the calibrated tools and equipment and

c) The procedure does not always address the calibration method and specifically the points of calibration where applicable.

Discussion

a) A review of calibration equipment indicates that some tools and equipment have not been included in procedure PC-31. These items are:

1) Deadweight tester
2) Torque wrench tester
3) Densitometer

b) The following items, although listed in MCCo procedure PC-31, did not have specific calibration acceptance criteria listed (ie. acceptance tolerances). The auditor was told that the manufacturers recommendations were used as the basis for acceptance.

1) Gauge blocks
2) Multi point recorder
3) Micrometers
4) Linear measuring devices
5) Optical level
6) Ammeter
7) Standard micrometer lengths
8) Standard verniers

In addition, the following list of items were found to have acceptance criteria discrepancies or lack clarity.

9) PC-31 states that dial indicators only need free movement of the plunger, but calibration Certification form PC-61 indicates tools should be within 0.001" of a standard.
10) PC-31 indicates pressure gauges are to be calibrated within + 5%. While witnessing pressure gauge calibration demonstration, MCCo personnel were noted to be divided as to whether this means 5% of full scale or each incremental calibration value.

c) While PC-31C was noted to be very specific about data recording for torque wrenches, the following inconsistencies were noted in other areas:

1) Micrometer (S/N QC-5) was tested on June 4, 1982 at six settings, whereas on March 4, 1982 it was tested at thirteen settings.
2) Test settings for pressure gauges are listed on Calibration Certification form PC-61, but are not indicated in procedure PC-31.

FINDING #5

Contrary to 10CFR50 Appendix B, Criteria XII and XVII, it was an accepted MCCo practice not to record the actual test data on calibration records when the data was extremely close to the standard value used.

Discussion

Through interviews with the personnel performing and witnessing the calibration activities and a review of the historical records, it became evident that MCCo did not always record the actual test data readings when they were within acceptable limits. The generic test point values were recorded. A go, no-go check was performed. The interviews did conclude, however, that out of tolerance calibration readings were being identified.

FINDING #6

Contrary to 10CFR50 Appendix B, Criteria XII and XVII, Morrison Construction Co. is not adequately controlling calibration documentation for measuring and test equipment.

Discussion

A review of all calibration records on file at Morrison was conducted to determine the extent of documentation deficiencies. The deficiencies involved (1) the frequent use of reproductions of calibration report forms already signed by the cognizant Quality Control Inspector and Witnesses to record calibration data, (2) the incompleteness of data, dates and signatures on calibration reports and (3) the use of reproduction of calibration data on consecutive reports.

FINDING #7

Contrary to 10CFR Appendix B, Criteria IX and XVII and Morrison procedure PC-41 Rev. 0, three (3) voltage/amperage (V/A) readings recorded on Morrison Quality Control surveillances were found to be out of procedure tolerances and there was no evidence of corrective action. In addition, there were instances of incomplete and/or incorrect information on the surveillance forms, such as lack of proper procedure revision and V/A ranges listed on the form which were not consistent with the procedure ranges.

Discussion

a) The following reports had the V/A readings out of procedure tolerances.

Report Date	Reading	Procedure Range	Observed	Procedure
4/24/81	Amps	60-250	55 amps	P1-3LS Rev. 7
6/1,2/78	Amps	60-250	50 amps	P1-11LS Rev. 7
12/2/77	Amps	60-110	125 amps	P1-20LS Rev. 4

The audit team subsequently reviewed the three above cases with a Sargent & Lundy Welding Engineer. This discussion included a review of the associated weld procedures, the materials being welded, and the weld inspections performed. Based on this review, it did not appear to this S&L Engineer that a quality concern exists.

b) In addition, inaccurate recording of surveillance information was identified. Out of 343 surveillance reports reviewed, 104 had incorrect or no procedure revision identified on the report form and 95 had V/A ranges listed which were not consistent with the ranges provided in the applicable procedure.

OBSERVATION #1

Quality Control final review requirements on quality records are not clearly defined in MCCo procedures or site instructions.

Discussion

The following record types receive a final review by MCCo Quality Control personnel but the meaning and requirements of this review are not delineated:

a) Calibration records
b) Installation documentation/travelers
c) NDE reports
d) Test reports

Additionally, the final Quality Control review of the Weld Data Record is performed, but documentation to indicate acceptance, other than entering selected information from it on the Status Log, is not provided.

OBSERVATION #2

MCCO has not performed reviews of records to determine if documentation is missing. Additionally, current procedures do not provide instructions for replacing lost records.

Discussion

MCCo's systems of logging documentation appear adequate as a means of accounting for documentation but no evidence was available to indicate that MCCo had performed reviews for missing documentation. MCCo also has not established steps to be taken to replace documents discovered missing.

OBSERVATION #3

Millwrights were observed calibrating torque wrenches with some variation from procedure PC-31.

Discussion

While observing millwright and pipefitter demonstrations of the calibration of various tools, it was noted that "clicker type" torque wrenches were manually calibrated. A review of procedure PC-31 indicates that these wrenches are to be calibrated using the mechanical lead device which is built into the torque wrench tester rather than be pulled by hand.

The following comments are relevant to this observation:

1) The pipefitters and millwrights performing the various calibration functions appear to be competent.

2) Although use of the mechanical lead device would seem to make the recording of calibration data more accurate, the results obtained by manually calibrating the "clicker type" wrench were not substantially different.

3) The mechanical device on the torque wrench tester can not accommodate large (500 ft-lbs) torque wrenches.

OBSERVATION #4

MCCo Tool/Instruments Inventory listing is not current.

Discussion

Gauge blocks (S/N BE1-9) are not listed on MCCo Tool/Instruments Inventory List. Also, as a comment, the list should be updated as several hand written entries were noted.

OBSERVATION #5

A Quality Control Inspector certification was not signed by the MCCo certifying individual.

Discussion

A review of certifications for all Quality Control Inspectors identified the omission of the Certification signature for J. R. Ragan, Level II Limited. Other than the missing signature, the information on the certification form and back up documentation appeared to be in order.

LASALLE COUNTY UNIT I & II

REPORT OF: ACTION TAKEN BY PROJECT CONSTRUCTION AS A RESULT OF NRC AND C.E.Co./Q.A. CONCERNS RELATIVE TO MORRISON CONSTRUCTION COMPANY CALIBRATION RECORDS.

DATE: JULY 8, 1982

REPORTED BY: Daniel J Skoza
DAN SKOZA
PROJECT CONSTRUCTION ENGINEER

REVIEWED BY: Warren E Vahle
WARREN VAHLE
PROJECT CONSTRUCTION LEAD MECHANICAL ENGINEER

APPROVED BY: N J Maley
N.J. MALEY
MANAGER OF PROJECTS

SUMMARY

This report summarizes the confirmatory measures performed by Project Construction to verify the adequacy of the field work done by Morrison Construction Company which required the use of calibrated torque wrenches and hydro test pressure gauges. Our review substantiates our conclusion that bolts were properly tightened and pressure systems were properly measured. Exceptions that were found represent small percentages of the sample. It is not evident that these exceptions are traceable to calibration deficiencies. All exceptions discovered have been corrected or are otherwise addressed in this report. Therefore, we believe that the plant construction is adequate and satisfactory.

REPORT

I. TORQUE WRENCH CERTIFICATIONS:

Of the approximately 6,000 MCCO torque wrench certifications the following Minor and Major catorgories were identified.

A. Specific Concerns Deemed Minor

1. Torque wrench type misidentified or not stated.
2. Math errors on certification but wrench within tolerance.
3. Error calculations not performed but wrench within tolerance.
4. Missing QC signatures or report date.

B. Specific Concerns Deemed Major

1. Single xeroxed report - data and all signatures.
2. Group xeroxed report - data and all signatures (1-6 week period, 1-12 week period, 1-3 week period).
3. Missing tester or witness signature.
4. Incomplete or uncertain data at isolated data points.
5. Missing reports.

C. Torque Wrench Out-Of Tolerance Concerns

Certifications identified with out-of-tolerance readings at isolated data points and had been accepted by MCCO Q.C.

D. Generic Concerns

1. Numerous series of consecutive reports with identical original test data indicating actual test data was not recorded but a go/no go test was utilized and nominal values recorded when the data point was deemed within calibration tolerance.
2. Numerous cases of Q.C. acceptance stamp xeroxed onto calibration data sheet already signed. Only date of Q.C. review and page number was affixed to indicate Q.C. acceptance review.

The specific concerns deemed minor required no remedial field checks of the work done with the subject wrenches. In all cases the concern is a paper problem. Other information existed on the certification to correct the concern, or a calculation was easily performed to correct the concern, or the Q.C. review was performed on the calibration certification.

The specific findings deemed major and the out of tolerance wrenches required ascertaining the safety related work performed with the wrench in question and field checking the torque for acceptability. 67 hangers with approximately 392 CEA's and bolting on 2 valves were required to be rechecked. The results of the rechecks indicated 14 CEA bolts on 9 hangers were found lower than the specified test torque. These bolts will be retorqued or replaced per LS-CEA requirements. The first 4 of these hanger assemblies were recalculated and found that they would function without the subject CEA's. The remaining corrective action will be completed by July 12, 1982.

The basis for accepting the installed work given the generic findings are the following:

A. MCCO general calibration procedure PC-31 only requires a 6 month frequency for torque wrenches utilized on all work except CEA's. CEA work required weekly or monthly calibrations of torque wrenches. Calibration certifications with actual data exist to meet the 6 month frequency intervals.

B. It has been shown that torque wrench accuracy is not important for CEA work. The primary reason for specifying a torque value in CEA work is to assure an auditable method of ascertaining that the CEA is set in the concrete. Once the anchor has been shown to be set, torque tightness of the CEA becomes irrelevant. This being the case, the go/no go calibration test utilized should be sufficient to ascertain that the torque wrench is accurate enough to set the CEA. An independant check performed by ConAm Inspection Agency with their own calibrated torque wrenches on a minimum of one anchor per hanger assemblies but at least one out of each ten expansion anchors installed in that assembly showed no excessive rejection of CEA's throughout the work history. The rejection has been extremely low; on the order of 1% of the bolts tested.

C. Evidence exists that torque wrenches were discarded or returned to the manufacturer when calibration data showed the wrench out-of-calibration.

D. Evidence exists that the torque wrench calibration certification did receive a Q.C. review even though the Q.C. stamp and signature were xeroxed. The date of review and page number were original writings.

In summary, we have concluded that the torque wrenches were sufficiently accurate to assure the installed work meets its functional requirements.

II. TORQUE WRENCH TESTER CALIBRATION:

During the NRC review a concern was stated that the tester utilized to calibrate the torque wrenches was out-of-tolerance on specific scale readings. Commonwealth Edison Company Nonconformance Report 599 documented the calibration discrepancies. Preliminary reviews have indicated the following:

A. Variables such as the tester condition and operator variance probably caused the erroneous readings in the ranges identified.

B. The torque tester or torque wrench calibration tolerances are not the primary concern in torque operations. Tolerances can be relaxed in most situations with assurance the installed bolt condition is acceptable.

On the above basis, no checks of the installed work has been initiated.

III. HYDRO PRESSURE GAUGE CALIBRATION CERTIFICATIONS:

Approximately 1000 MCCO Hydro Test Pressure Gauge Calibration Certifications for 100 gauges were reviewed by C.E.Co. Q.A. auditors. The known concerns are the following:

A. Specific Concerns

1. Partial or no calibration data listed.
2. No witness and/or Q.C. signatures.
3. Date of calibration in error.
4. Out-of-tolerance readings accepted by Q.C.
5. Calibration missing.
6. Tester and Q.C. signature not original.

B. Generic Concerns

1. Actual test data was not recorded but go/no go test utilized and nominal values recorded when the data point was deemed within calibration tolerance.
2. Witness and Q.C. signatures xeroxed onto calibration data sheet. No objective evidence exists on calibration data sheet indicating tests were witnessed and Q.C. reviewed.

The specific concerns can be resolved without remedial field checks. The questionable gauges were not utilized at all, or not utilized in the ranges with the problem data.

The Generic concerns require several different explanations and actions to resolve them:

1. Variations in industry instrument standards and dial reading techniques which include paralax effects tend to vary the readings taken during instrument calibrations. In some cases, time was not taken to log the actual readings if it was apparent that the values were within the specified tolerances. In these cases, the nominal settings were apparently logged. Reviews of the calibration files for each instrument and successive calibrations of the same instrument have shown only several instances of instruments out of tolerance and only within a small part of their range. These reviews substantiate that the readings taken were of sufficient accuracy to use as a basis for the certifications. Additionally, records indicate 35 hydro test pressure gauges out of 107 were discarded during the course of work when calibrations could not be completed within specifications.

2. The establishment of validity of the xeroxed witness and Q.C. signature can be done through the Hydro Test Pressure Gauge Usage and Calibration Log and by virtue of the fact that gauge calibration is a two(2) man operation. For each gauge, a log is kept identifying the dates the instrument was calibrated and the test number corresponding to that date. The Q.C. inspector that witnessed the test then signs the pressure gauge log indicating the test was completed. This was done with original signatures on a 100% basis. The testing apparatus requires two people to operate, one to pump the machine and read the pressure gauge and another to assure the dead weights are free floating. This has been done historically by a Pipefitter and Q.C. inspector.

On the above basis, it was not necessary to establish a program to recheck previous pressure tests.

IV. The containment valve structural bolting review has been completed. All bolts on the safety related valves have been torqued to within the manufactures' recommended values.