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

PRC

DEC 13 1984

Docket Nos.: STN 50-454/50-455

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Asselstine
Commissioner Bernthal
Commissioner Roberts

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: BYRON QUALITY ASSURANCE RELATED DOCUMENTS
(BOARD NOTIFICATION 84-186)

In accordance with present NRC procedures for Board Notifications, the following documents related to Byron quality assurance are being provided.

1. Letter dated November 15, 1984 from J. F. Streeter (NRC) to Cordell Reed (Commonwealth Edison) enclosing Inspection Report Nos. 50-454/84-55 (DRP); 50-455/84-38 (DRP). Item 4 of this Inspection Report closes the issue of electrical conductor butt splices.

2. Letter dated December 5, 1984 from James G. Keppler (NRC) to Cordell Reed (Commonwealth Edison) enclosing Inspection Report No. 50-454/84-32 (DRP); 50-455/84-25 (DRP) and a Notice of Violation and Proposed Imposition of Civil Penalty for actions related to Systems Control Corporation.

3. Letter dated November 20, 1984 from T. R. Tramm (Commonwealth Edison) to R. C. DeYoung (NRC) providing additional information concerning the Integrated Design Inspection (IDI).

Frank J. Miraglia
for Darrell G. Eisenhut, Director
Division of Licensing

Enclosures: As stated

cc: SECY (2)
OPE
OGC
EDO
Parties to the Proceeding
Alan S. Rosenthal, ASLAB
Dr. Reginald L. Gotchy, ASLAB
Howard A. Wilber, ASLAB
Ivan W. Smith, ASLB
Dr. Dixon Callihan, ASLB
Dr. Richard F. Cole, ASLB
ACRS (10)

8412170374

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

L. Olshan
NRR

ENCLOSURE 1

NOV [illegible] 1984

Docket No. 50-454
Docket No. 50-455

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

Gentlemen:

This refers to the routine safety inspection conducted by Messrs. J. M. Hinds, Jr., K. A. Connaughton, P. G. Brochman of this office on August 1 - October 3, 1984, of activities at Byron Station, Units 1 and 2, authorized by NRC Construction Permits No. CPPR-130 and No. CPPR-131 and to the discussion of our findings with Mr. R. E. Querio and others of your staff at the conclusion of the inspection.

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.

During this inspection, certain of your activities appeared to be in noncompliance with NRC requirements, as specified in the enclosed Appendix. A written response is required.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure(s) will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1). If we do not hear from you in this regard within the specified periods noted above, a copy of this letter, the enclosure(s), and your response to this letter will be placed in the Public Document Room.

The responses directed by this letter (and the accompanying Notice) are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

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

Sincerely,

J. F. Streeter, Director
Byron Project Division

Enclosure: IE Inspection
Report No. 50-454/84-55(DRS)
and No. 50-455/84-38(DRS)

cc w/encl:
D. L. Farrar, Director
of Nuclear Licensing
V. I. Schlosser, Project Manager
Gunner Sorensen, Site Project
Superintendent
R. E. Querio, Station
Superintendent
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII Byron
Resident Inspector, RIII
Braidwood
Phyllis Dunton, Attorney
General's Office, Environmental
Control Division
D. W. Cassel, Jr., Esq.
Diane Chavez, DAARE/SAFE
W. Paton, ELD
L. Olshan, NRR LPM

Appendix

NOTICE OF VIOLATION

Commonwealth Edison Company

Docket No. 50-454
Docket No. 50-455

As a result of the inspection conducted on August 1 - October 3, 1984, and in accordance with the General Policy and Procedures for NRC Enforcement Actions, (10 CFR Part 2, Appendix C), the following violation was identified:

10 CFR 50, Appendix B, Criterion XI states, in part: "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents... Test results shall be documented and evaluated to assure that test requirements have been satisfied."

The Byron FSAR, Appendix A states, in part, that the licensee complies with the requirements of NRC Regulatory Guide 1.79, Revision 1, September 1975.

NRC Regulatory Guide 1.79, Revision 1, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors", Regulatory Position C.2.b, "Valves" requires verification of proper operation of system valves including response times. Regulatory Guide 1.79 further states that this requires visual verification as well as proper control room indication.

Contrary to the above:

a. For numerous Emergency Core Cooling System (ECCS) and other Engineered Safety Feature (ESF) valves, preoperational testing was not conducted to verify the accuracy of remote valve position indication used in ESF response time measurements to signify that valves completed stroking to the positions required to fulfill their safety function.

b. Valve stroke time data for certain ECCS and other ESF valves which suggested inaccuracies in remote valve position indication was not adequately evaluated to determine the acceptability of ESF response time measurements.

This is a Severity Level IV violation (Supplement II). (454/84-55-01; 455/84-38-01)

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Pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within thirty days of the date of this Notice a written statement or explanation in reply, including for each item of noncompliance: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Dated

J. F. Streeter, Director
Byron Project Division

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-454/84-55(DRP); 50-455/84-38(DRP)

Docket Nos. 50-454 and 50-455 License Nos. CPPR-130; CPPR-131

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

Facility Name: Byron Station, Units 1 and 2

Inspection At: Byron Station, Byron, IL; Corporate Office, Chicago, IL

Inspection Conducted: August 1 - October 3, 1984; Corporate Office on September 21, 1984

Inspectors: J. M. Hinds, Jr.
K. A. Connaughton
P. G. Brochman

Approved By: J. F. Streeter, Director
Byron Project Division

J. F. Streeter

11/13/84
Date

Inspection Summary

Inspection on August 1 - October 3, 1984 (Report Nos. 50-454/84-55(DRP); 50-455/84-38(DRP)

Areas Inspected: Routine, unannounced safety inspection of licensee action on previously identified items; SER items; 10 CFR 50.55(e) reports; 10 CFR 21 reports; IE Bulletins; operating procedures; emergency procedures; onsite and offsite review; preoperational test results; operational staffing, plant tours/housekeeping and other activities. The inspection consisted of 556 inspector-hours onsite by three NRC inspectors including 53 inspector-hours during off-shifts and 10 inspector-hours at corporate headquarters.

Results: One item of noncompliance was identified; failure to adequately measure time response of certain valves required to actuate on receipt of an ESF signal.

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DETAILS

1. Persons Contacted

Commonwealth Edison Company

*R. Querio, Station Superintendent
R. Tuetkin, Startup Coordinator
*R. Ward, Assistant Superintendent, Administrative & Support Services
*R. Pleniewicz, Assistant Superintendent, Operating
L. Sues, Assistant Superintendent, Maintenance
*F. Hornbeak, Unit 2 Testing Supervisor
M. Loehman, Project Construction Assistant Superintendent
T. Tulon, Operating Engineer
T. Higgins, Training Supervisor
*S. Dresser, Preoperational Test Coordinator
J. Hart, Personnel Administrator
T. Brechon, Technical Staff
S. Gackstetter, Technical Staff
A. Chernick, Technical Staff
A. Chomache, Independent Safety Engineering Group
W. Benjamin, Senior Participant, Offsite Review
T. Oracki, Operating Staff (SRO)
R. Poche, Technical Staff
D. St. Clair, Technical Staff Supervisor
C. Kilbride, Technical Staff
D. Meier, Engineering Assistant, Operating
J. Pausche, Technical Staff
A. Mills, Fuel Handling Foreman
D. Popkins, Shift Foreman
S. Barrett, Station Chemist
*D. Sible, QA Engineer
*R. Gruber, QA Engineer
*G. Stauffer, Technical Staff
*P. Anthony, Technical Staff

The inspectors also contacted and interviewed other licensee and contractor personnel during the course of this inspection.

*Denotes those present during the exit interview on October 3, 1984.

2. Licensee Actions on Previously Identified Items

(Closed) Open Item Nos. (454/83-14-02; 455/83-12-02); Evaluation of anomalies in radiological environmental monitoring program. Based on the licensee's response of August 3, 1984, this item was closed by letter dated August 6, 1984, from C. J. Paperiello to Commonwealth Edison Company, Cordell Reed.

(Closed) Open Item Nos. (454/83-46-02; 455/83-34-02); Additional four air samplers in field. Based on the licensee's response of August 3, 1984, this item was closed by letter dated August 6, 1984, from C. J. Paperiello to Commonwealth Edison Company, Cordell Reed.

(Closed) Unresolved Items Nos. (454/83-49-02; 455/83-35-02); Method of calculating engineered safety features response times. As discussed in Paragraph 7b of this report the licensee did not account for potentially nonconservative errors in ESF time response measurements introduced by the use of remote valve position indication for measurement endpoints. The licensee had not established the accuracy of remote indication (e.g., by evaluation of locally measured stroke times and visual verification of acceptable limit switch operation) for all applicable valves. This matter will be tracked as an item of noncompliance as discussed in Paragraph 7b.

(Closed) Open Item Nos. (454/84-15-02; 455/84-11-02); Byron Administrative Procedure (BAP) 300-2, "Shift Manning". The inspector reviewed BAP 300-2, Revision 7 dated August 20, 1984. The procedure has been revised to reflect the licensee's latest proposal to utilize experienced personnel on each operating shift where the regular shift crew does not possess established minimum experience levels. These experienced personnel are called "Shift Advisors". Based on discussions with licensee personnel, should the latest proposal be rejected or require modification as a result of NRC staff review, BAP 300-2 will be revised to reflect any alternative on-shift operating experience requirements imposed prior to issuance of the Byron Unit 1 operating license.

(Closed) Open Item Nos. (454/84-15-05; 455/84-11-05); Acceptability of preoperational test results for preoperational test 2.17.10, "Containment Spray". Completion of inspector review of test results for this preoperational test is discussed in Paragraph 7a of this report. No items of noncompliance or deviation were identified.

(Closed) Open Item Nos. (454/84-19-03; 455/84-14-01); Adequacy of implementing procedures for the Byron onsite review investigative function. The inspector reviewed revised Byron Administrative Procedure (BAP) 1210-1, "Conduct of Onsite Reviews and Investigations", Revision 3. This procedure was revised to include additional instructions for conducting reviews and investigations in accordance with the requirements of Byron Unit 1 Technical Specification 6.5.2. Items not included in previous revisions of this procedure and documented in NRC Inspection Report Nos. (454/84-19; 455/84-14) have been incorporated. The procedure details for each review item, methods of review, criteria to be applied, methods of documenting reviews and, where required, provisions for forwarding review results to the offsite review group.

(Closed) Open Items Nos. (454/84-42-01; 455/84-29-01); Licensee committed to revising the FSAR and tech specs organization to reflect required changes, etc. The licensee revised Figure 6.2-2, Unit Organization, Chapter 16 of the FSAR and submitted the change in the Farrar to Denton letter of October 3, 1984, to incorporate the revisions necessary to address the discrepancies identified by the inspector.

3. Byron Safety Evaluation Report (SER) Items

(Closed) SER Item Nos. (454/83-00-01; 455/83-00-01); Relocation of military training route SR 774. The inspector was provided with a copy of a letter dated March 16, 1983, from David H. Spindle, Major, U.S. Air Force Reserves, Airspace Manager for the 928 Tactical Airlift Group to the Air Force Representative of the Federal Aviation Administration. The letter stated that based upon a review of route usage it was determined that the route's existence was no longer justified. Furthermore the "bomb drop zone" associated with the training route (the sole reason for flying the route) had been eliminated by the state of Wisconsin several years earlier. Route SR 774 was removed from all publications, charts, and maps that show low level routes.

(Closed) SER Item No. (454/83-00-11); Tornado missile protection for diesel generator exhaust stacks. The inspector physically verified the installation of tornado missile protected relief devices on the Byron Unit 1 standby diesel generator exhaust lines. Should the vulnerable portion of the diesel generator exhaust stacks become constricted as a result of deformation caused by tornado missile impingement the rupture disks will relieve exhaust backpressure and ensure continued operation of the diesel generators. Earlier, installed rupture disks were found to have ruptured (failed safe) as a result of diesel generator operation though the exhaust stacks were not constricted. The currently installed rupture disks will be visually verified intact following routine surveillance tests of the diesel generators conducted in accordance with Byron Operating Surveillance Procedure (BOS) 4.8.1.1.2.

(Closed) SER Item Nos. (454/83-00-16; 455/83-00-16); Periodic sampling of instrument air quality. In accordance with the licensee's commitment provided in a letter dated January 2, 1982, the licensee has written Byron technical staff surveillance procedure (BVS) IA-1. "Instrument Air Sampling Requirements for Refueling Outages" Revision 0, dated September 27, 1984. The procedure requires determination of particulate content and moisture content, flushing of the instrument air risers and a check for water or oil. Air quality is required to meet ANSI N-45.2.1-1973 (Class C Cleanliness). The procedure can be performed in any operating mode and will be performed once per 18 months.

(Closed) SER Item Nos. (454/83-00-17; 455/83-00-17); Emergency operating procedures to require immediate starting of essential service water (ESW) makeup pumps upon loss of offsite power when outside air temperature is less than 40°F. In accordance with the licensee's commitment contained in a letter dated January 2, 1982, the licensee has included requirements to start the ESW makeup pumps under these specified circumstances. The inspector reviewed the following procedures:

Byron Annunciator Response (BAR) 1-20-C12 "RSH Bus 35-1 Volt Low" R-2, dated September 25, 1984.

Byron Annunciator Response (BAR) 2-20-C12 "RSH Bus 035-2 Volt Low" R-0, dated September 25, 1984.

Byron Emergency Procedure (BEP) ES-0.1 "Reactor Trip Recovery, Unit 1" R-2, dated September 25, 1984.

Byron Abnormal Operating Procedure (BOA) ELEC-4, "Loss of Offsite Power, Unit 1, For Modes 3 and 4" R-1 dated September 25, 1984.

The two BAR's are employed upon loss of power to the River Screenhouse (including heating and ventilation). If the outside air temperature is less than 40°F the ESW makeup pumps are required to be started. BEP ES-0.1 requires verification of offsite power availability. If offsite power is not available, starting of the ESW makeup pumps is required when the outside air temperature is less than 40°F (Step 9). BOA ELEC-4 requires a check of the outside air temperature. If the outside air temperature is less than 40°F starting the ESW makeup pumps is required (Step 6).

(Open) SER Item (454/83-00-19; 455/83-00-19); Second verification of AFWS valve position. In a letter dated January 2, 1982, the licensee committed to including an independent (second) verification of valve positions at the completion of a procedure. The inspector reviewed the following procedures for incorporation of independent verification:

BVS 7.1.2.1a-1, R-0, "Motor Driven Auxiliary Feedwater Pump Monthly Surveillance"

BVS 7.1.2.1.a-2, R-1, "Diesel Driven Auxiliary Feedwater Pump Monthly Surveillance"

BVS 7.1.2.1.b.1-1, R-1, "Auxiliary Feedwater Valve Emergency Activation Signal Verification Test"

BVS 7.1.2.1.b.2-1, R-1, "Auxiliary Feedwater Pump Emergency Actuation Signal Verification Test"

BVS 7.1.2.3.C-1, R-2, "Auxiliary Feedwater Diesel Prime Mover Inspection"

BVS 0.5-2,AF.3, R-1, "Auxiliary Feedwater Valves Indication Test"

BVS 0.5-3.AF.1, R-1, "ASME Surveillance Requirements for Auxiliary Feedwater Pumps"

BOS 7.1.2.2-1, R-1, "Train A Auxiliary Feedwater Flowpath Operability Surveillance Following Cold Shutdown"

BOS 7.1.2.2-2, R-0, "Train B Auxiliary Feedwater Flowpath Operability Surveillance Following Cold Shutdown"

The inspector identified items of concern and provided comment to the licensee for the following items:

For BVS 7.1.2.1.b.1-1 valves 1AF006A&B, 1AF017A&B, and 1AF018A&B are operated during this procedure, but do not have independent verification of position at the completion of the procedure.

For BVS 7.1.2.1.b.2-1 valves 1AF013A through 1AF013H should have individual independent verification rather than group verification of valve position.

This item will remain open pending licensee's resolution of the inspector's comments.

4. 10 CFR 50.55(e) Report Followup

(Closed) 50.55(e) Report No. (454/84-02-EE); Containment spray pump impeller misinstallation. The inspector observed disassembly of the 1A and 1B containment spray pumps and visually verified that a high capacity impeller had been erroneously installed in the 1A pump and that a low capacity impeller had been erroneously installed in the 1B pump. The inspector has monitored licensee corrective actions which are now complete. The licensee has reinstalled the correct impeller types in the 1A and 1B pumps. Impeller serial numbers have been recorded for both pumps. Additional labeling of the pumps has been provided to prevent improper substitution of pump subassemblies. As discussed in paragraph 7a of this report, the Unit 1 pumps have been retested following corrective actions. Pump performance was determined to be acceptable.

(Closed) 50.55(e) Report Nos. (454/84-03-EE; 455/84-03-EE); electrical conductor butt splices. The licensee has completed an engineering review of 82 butt splices reinspected. The results of the review indicated that in 73 of the cases, a butt splice failure (resulting in an open circuit at the splice) would not result in a loss of control that would defeat the safety function of the associated equipment. In the remaining 9 cases it was determined that the safety function of associated equipment would be impaired. The licensee reported these results in a letter dated August 28, 1984, which also included detailed descriptions of butt splice failure consequences and an evaluation of safety impact. These evaluations included considerations of redundancy, indication available to alert operators of resultant abnormal equipment status and manual actions required to restore the safety function of affected equipment. The inspector reviewed the licensee's evaluations for the 9 cases determined by the licensee to result in a loss of safety function. Based upon the reported effects of these postulated butt splice failures the inspector determined that the failure of any one of the butt splices, while impairing a safety function, would not, in and of itself, put the plant in an unanalyzed condition. In addition, manual operator actions could be taken to restore the safety function of affected equipment. All butt splices reviewed were found to have performed satisfactorily during preoperational testing.

5. 10 CFR Part 21 Report Followup

(Open) 10 CFR 21 Report Nos. (454/84-01-PP; 455/84-01-PP); Recommended replacement of viton elastoner seals on hydrogen recombiners - Rockwell International. The inspector verified that the licensee had been informed by Rockwell of the concern relating to possible seal failure resulting from exposure of the viton elastoner seals to a post LOCA environment. By letter dated May 28, 1984, Rockwell specified acceptable replacement seals and instructions for procurement. The licensee issued Purchase Order #501175 to procure the replacement seals. The licensee has also issued Construction Work Record VQ-0006 to require installation of the replacement seals.

(Closed) 10 CFR 21 Report Nos. (454/84-02-PP; 455/84-02-PP); Environmental qualification of Gould circuit breakers employed as supply breakers for hydrogen recombiners manufactured by Rockwell International. Information concerning failure of the subject circuit breakers during Environmental Qualification testing was previously reported to the licensee via NRC's Office of Inspection and Enforcement Information Notice No. 83-72 (Item 18). The licensee addressed the concern by relocating the power and control cabinets 0G004J and 0G006J which house the circuit breakers. The inspectors physically verified the locations of the panels. The inspectors then verified by review of FSAR Section 3.11 Tables 3.11-1 and 3.11-2 and Figure 3.11-1 that the Environental Zones in which the circuit breakers are located will be maintained at ambient temperatures of less than 100°F under normal, abnormal and accident conditions. This corrective action is consistent with that specified in the Part 21 report submitted by Rockwell International to the NRC by letter dated March 18, 1983.

6. Inspection and Enforcement Bulletin (IEB) Followups

(Closed) IEB 81-03 "Flow Blockage of Cooling Water to Safety System Components by Corbicula Sp. (Asiatic Clam) and Mytilus Sp. (Mussel)". As discussed in NRC Inspection Report Nos. (454/84-42; 455/84-29 (DRP)), this bulletin was reopened pending review of the data from the licensee's last sampling survey and the documents which implement the commitment to periodically conduct such surveys. The licensee provided the inspector with a letter dated September 25, 1984, from C. L. McConough to R. E. Querio which reported the results of the 1984 Byron Station Corbicula Survey. Corbicula were not found to inhabit the forebay or screenhouse located on the Rock River. The inspector reviewed Byron Administrative Procedure (BAP) 599-41, Revision 0, dated March 17, 1984, "Byron Station Microbiological Program Description". "Operational Limits", Section 3 of this program description, states that "A search for Asiatic clams will be initiated every spring or fall starting one year after continuous heat discharge".

7. Preoperational Test Results Evaluation

a. Preoperational Test 2.17.10, "Containment Spray"

As documented in NRC Inspection Report Nos. (454/84-19; 455/84-14), the licensee's Project Engineering Department (PED) had provided comments on the containment spray test results by letter dated April 12, 1984. The letter required additional testing re-evaluation and resubmittal of the test results package for review by PED.

The inspector reviewed the approved test package which consisted of the completed original preoperational test procedure and completed retest procedures R-35, R-58 and R-232. The inspector's review was performed to verify that testing activities were conducted in an acceptable sequence; that test changes were properly documented, reviewed and approved in accordance with the licensee's administrative procedures; all test deficiencies were appropriately resolved and that any required retesting was performed; test results were evaluated against acceptance criteria by engineering personnel and required results approvals were obtained and properly documented.

The inspector also reviewed outstanding Construction Work Records (CWR's), Action Item Records (AIR's) and Nuclear Work Requests issued to track corrective actions specified in the resolution of deficiencies. The inspector verified that these items have been appropriately scheduled for resolution as required to support system operability per the Byron Unit 1 Technical Specifications.

Inspector concerns included in the licensee's April 12, 1984 letter have been resolved. This report documents completion of inspector review of the subject test results. No items of noncompliance or deviation were identified.

b. Preoperational Test 2.26.10, "Engineered Safety Features"

During this reporting period the inspector began reviewing the results of the subject test which were approved by PED on June 29, 1984 and by the licensee's QA organization on August 14, 1984. The review during this reporting period was limited to review of corrective action taken concerning test deficiency No. 8967. This deficiency was written to address potential nonconservative errors in Engineered Safety Features (ESF) time response measurement for valves required to change position in response to an ESF actuation signal within a specified maximum allowable time interval. The potentially nonconservative error would result from premature limit switch actuation (i.e., limit switch actuation providing remote indication that the valve was in the required position prior to the valve actually reaching the required position). The subject preoperational test was employed to measure ESF time response from the ESF actuation system output relay operation to the completion of valve's stem motion. The time response measurement endpoint was taken from remote valve position indicating lights actuated by limit switches at the valve. Earlier in the preoperational test program the licensee had, for certain valves, measured valve stroke time both remotely using indicator lights) and locally by direct observation of stem travel. In some cases the data indicated that valve motion had continued after limit switch actuation and receipt of remote indication.

Deficiency No. 8967 documented the following recommended resolution: "Compare remote timing obtained in EF 2.26.10 test to values obtained in the individual tests for the equipment". Deficiency No. 8967 was subsequently closed and a duplicate deficiency No. 16328 written against preoperational test EF 2.26.12. Data gathered EF 2.26.10

test was later summed with time response data for instrumentation and logic in preoperational test EF 2.26.12, "EF Logic and Time Response" and then compared with acceptance criteria for overall time response.

The inspector reviewed data provided to PED by Byron letter 84-1081, dated August 26, 1984, to resolve Deficiency No. 16328. For valves which had previously been stroke timed both remotely and locally during preoperational testing these stroke times were provided along with the response time measurement taken in EF 2.26.10. Each of these three times were individually summed with the slowest logic response time to yield "total response times". For valves which were not locally stroke timed during preoperational testing, data was not provided. The data sheet was marked N/A in such cases.

The inspector determined the corrective actions taken to address Deficiency Nos. 8967 and 16328 to be inadequate in two respects: (1) where data was provided for both local and remote valve stroke times, the method of evaluation was inconclusive with respect to establishing acceptable time response (i.e., differences in remote and local stroke times were not added to the response time measurements taken in EF 2.26.10 where local stroke times were longer indicating possible premature remote indications) and (2) local stroke time data was not provided to support such an evaluation for numerous valves including certain ECCS values for which Regulatory Guide 1.79 explicitly requires "...verification of proper operation including response times. This requires visual verification as well as proper control room indication." The licensee is committed to Regulatory Guide 1.79 as discussed in Appendix A to the Byron FSAR. This is an item of noncompliance (454/84-55-01; 455/84-38-01).

8. Safety Committee Activities

a. Onsite Review Activities

The inspector reviewed the following Onsite Review Meeting Documentation Packages:

Onsite	Review No.	Date	Subject
OSR	84-20	7/13/84	Proposed Tech. Spec. Revision
OSR	84-21	7/14/84	Hot Ops. Sequencing Document
OSR	84-22	7/17/84	Proposed Tech. Specs.
OSR	84-23	7/18/84	GSEP Revision 4a
OSR	84-24	7/25/84	"C" RCP Problem - Hot Ops.
OSR	84-26	7/31/84	Proposed SNM License Amendment
OSR	84-27	8/2/84	INRYCO Tendon Surveillance Procedures
OSR	84-28	8/6/84	INRYCO Tendon Surveillance Procedures
OSR	84-32	9/4/84	INRYCO Tendon Surveillance Procedures

The inspector determined that reviews were being performed and documented in accordance with the effective revisions of Byron Administrative Procedure (BAP) 1210-1, "Conduct of Onsite Review and Investigative Function". The inspector also verified that onsite reviews were being conducted for new and revised station procedures written to satisfy the current proposed Technical Specifications. As discussed in paragraph 2 of this report, BAP 1210-1, Revision 3, has been written to include instructions for reviews and investigations of all item types described in the Byron Technical Specifications. Onsite reviews and investigations will be conducted in accordance with all the provisions of this procedure following issuance of the Byron Unit 1 Operating License. No items of noncompliance or deviation were identified.

b. Offsite Review

(1) Procedures

The inspector reviewed the licensee's Office of Nuclear Safety Manual, Revision dated February 3, 1984. This document contains implementing procedures for the Offsite Review Group including designated participants, descriptions of items to be reviewed, methods of documenting reviews, distribution of review results and reporting requirements. These procedures are common to the Offsite Review Groups for all of the licensee's nuclear stations. The inspector interviewed the recently designated Senior Participant for the Byron Offsite Review Group and provided the following comments:

- The procedures refer to "Reportable Occurrences requiring 24- hour notification" instead of Reportable Events as defined in 10 CFR 50.72.

- Reference to Byron Technical Specification 6.3.1a is referenced in paragraph IV.A.1.I.B of the manual instead of Technical Specification 6.5.1a as is currently applicable. (Typographical error) No items of noncompliance or deviation were identified.

(2) Activities

Based upon interviews with licensee personnel, the inspector determined that offsite review activities would not formally commence in accordance with Byron Unit 1 Technical Specifications until issuance of the operating license. The Offsite Review is not required to be functioning prior to that time. The inspector informed licensee personnel that, in view of the nature and number of requirements which will be placed upon the Offsite Review Group at the time of operating license issuance, resources should be directed towards assuring that the Offsite Review Group is familiar with the procedures and practices of the Onsite Review and Investigative Function and that effective lines of communication have been established prior to operating license issuance. Licensee personnel acknowledged the inspector's comments. No items of noncompliance or deviation were identified.

9. Operating Procedure Review (42450B)

a. General

The inspector's review included general plant operating, system operating, chemistry surveillance, operating surveillance, technical specification surveillance, and fuel handling procedures. Procedures were reviewed for compliance and consistency with proposed Technical Specifications, Regulatory Guide 1.33 and ANSI N18.7-1976/ANS-3.2.

b. The following General Plant Operating Procedures were selected for review:

BGP 100-2, R-3, "Plant Startup"
BGP 100-3, R-1, "Power Ascension 5% to 100%"
BGP 100-5, R-2, "Plant Shutdown and Cooldown"
BGP 100-A7, R-1, "Mode 2 to 1 Checklist"
BGP 100-A10, R-0, "Main Steam Piping Pressurization"
BGP 100-T2, R-1, "Startup Flow Chart"
BGP 100-T9, R-0, "Main Control Board Feedwater 1PM04J"
BGP 100-T11, R-2, "Main Control Board Safeguard Panel 1PM06J"
BGP 100-T13, R-0, "Main Control Board Nuclear Instrumentation Panel 1PM08J"
BGP 100-T16, R-0, "Main Control Board Containment Isolation Valve Panel 1PM11J

c. The following System Operating Procedures were selected for review:

BOP AB-1, R-2, "Preparing a 4.0 - 4.4 w/o Solution of Boric Acid"
BOP AB-4, R-2, "Filling Boric Acid Tank '1' from Boric Acid Tank '2'"
BOP AB-5, R-2, "Filling Boric Acid Tank '2' from Boric Acid Tank '1'"
BOP AB-9, R-2, "Boric Acid Recycle Monitor Tank Operations"
BOP AF-1, R-5, "Motor Driven Auxiliary Feedwater Pump A Startup"
BOP AF-4, R-3, "Auxiliary Feedwater Pump B (Diesel) Startup"
BOP AF-7, R-0, "Draining the Auxiliary Feedwater System"
BOP AR-1, R-2, "Interrogation of the AR/PR RM-11"
BOP BR-1, R-1, "Dilute Mode of the Boron Thermal Regeneration System"
BOP BR-2, R-1, "Borate Mode of the Boron Thermal Regeneration System"
BOP BR-8, R-1, "Startup of BR Chillers"
BOP CC-1, R-4, "Component Cooling Water System Startup"
BOP CC-6, R-1, "Electrical Line Up of Component Cooling Pump '0'"

BOP CC-7, R-1, "Component Cooling System Filling and Venting"
BOP CD-2, R-4, "Condensate/Condensate Booster Pump Initial Startup"
BOP CD-3, R-0, "Starting an Additional Condensate/Condensate Booster Pump"
BOP CS-1, R-1, "Containment Spray System Recirculation to the RWST"
BOP CS-2, R-0, "Containment Spray System Shutdown after Automatic Initiation"
BOP CS-5, R-1, "Filling and Venting the Containment Spray System"
BOP CV-1, R-3, "Charging Pump Operation"
BOP CV-3, R-1, "Operation of the Reactor Makeup System to the Borate Mode"
BOP CV-5, R-1, "Operation of the Reactor Makeup System to the Dilute Mode"
BOP CV-6, R-1, "Operation of the Reactor Makeup System to the Alternate Dilute Mode"
BOP CV-8, R-2, "Operation of the Reactor Makeup System to the Manual Mode"
BOP CV-18, R-0, "Degassing the Reactor Coolant System and Pressurizer"
BOP CV-19, R-1, "Establishing CV Charging"
BOP DC-1, R-4, "125 VDC Battery Charger StartUp"
BOP DC-2, R-2, "125 VDC Battery Charger Shutdown"
BOP DC-5, R-3, "125 VDC Buss Cross-Tie"
BOP DG-1, R-5, "Diesel Generator Startup, Remote or Local"
BOP DG-3, R-5, "Diesel Generator Shutdown, Remote, Local or Emergency"
BOP DG-4, R-4, "Paralleling the Diesel Generator"
BOP DO-5, R-3, "Filling the Auxiliary Feedwater Pump Diesel Oil Day Tank"
BOP DO-12, R-1, "Draining a Diesel Generator Diesel Oil Day Tank"
BOP DO-16, R-1, "Draining a Diesel Generator Fuel Oil Storage Tank"
BOP FC-1, R-1, "Unit 1 and 2 Fuel Pool Cooling System Start-Up"
BOP FC-2, R-0, "Fuel Pool Cooling System Shutdown"
BOP FC-7, R-2, "Spent Fuel Pit Make-Up"
BOP FP-1, R-3, "Manual Startup and Shutdown of the Diesel Driven Fire Pump"
BOP FP-12, R-1, "Diesel Driven Fire Pump Auto Low Pressure Start"
BOP FW-1, R-1, "Startup of a Turbine Driven Main Feedwater Pump"
BOP GW-2, R-3, "Gas Decay Tank Release"
BOP GW-4, R-0, "Placing a Gas Decay Tank in Storage"
BOP IC-1, R-1, "Incore Moveable Detectors Flux Mapping Procedure"
BOP IP-1, R-4, "Energizing Instrument Distribution Busses From Their Inverter"
BOP IP-2, R-2, "Transfer of Instrument Bus from Inverter to Reserve Feed"
BOP IP-, R-0, "Restoring AC Input to an Instrument Bus"
BOP LM-1, R-0, "Loose Parts Monitoring System Low-Alarm Response"
BOP LM-2, R-0, "Loose Parts Monitoring System High Alarm Response"
BOP NR-3, R-1, "Excore System Verification"
BOP OG-10, R-2, "Startup of the 'A' Hydrogen Recombiner"
BOP OG-11, R-1, "Shutdown of the Hydrogen Recombiners"
BOP PS-3, R-0, "Steam Generator Blodown Sampling"

BOP PS-9, R-0, "Post LOCA Containment Hydrogen Monitoring System Operation"
BOP PW-3, R-0, "Shutdown of the Primary Water Make-Up System"
BOP RC-1, R-1, "Draining an Isolated RCS Loop"
BOP RC-2, R-1, "Startup of a Reactor Coolant Pump"
BOP RC-3, R-3, "Filling and Venting the Reactor Coolant System"
BOP RC-7, R-1, "Isolations on RCS Loop"
BOP RC-9, R-1, "Shutdown of a Reactor Coolant Pump"
BOP RD-3, R-0, "Control Rod Drive Motor-Generator Set Start-Up and Paralleling to Operating M-C Set"
BOP RE-4, R-1, "Draining the Refueling Cavity"
BOP RF-1, R-2, "Containment Filor Drain System Startup"
BOP RF-2, R-2, "Containment Floor Drain System Shutdown"
BOP RH-1, R-2, "RH System Startup"
BOP RH-7, R-2, "Filling and Venting the RH System from the RWST"
BOP RY-3, R-0, "Drawing a Pressurizer Steam Bubble"
BOP SD-2, R-1, "Shutdown of the Steam Generator Blowdown System"
BOP SI-3, R-2, "Lowering SI Accumulator Level with RCS Pressure Below 1000 psig"
BOP SI-4, R-3, "Increasing SI Accumulator Pressure"
BOP SI-7, R-2, "Lowering SI Accumulator Level by Equalizing the High Accumulator with the Low Accumulator (At All RCS Pressures)"
BOP SI-8, R-2, "Lowering SI Accumulator Level with RCS Pressure Greater than 1000 psig"
BOP SI-2, R-1, "Safety Injection System Startup"
BOP SI-14, R-0, "Initial Filling and Venting the Safety Injection System"
BOP SI-6, R-1, "Filling the Refueling Water Storage Tank"
BOP SI-17, R-0, "Placing the Refueling Water Storage Tank Heating Pump and Heater in Service"
BOP SI-18, R-0, "Removing the Refueling Water Storage Tank Heating Pump and Heater From Service"
BOP SI-20, R-2, "Filling and Venting an SI Accumulator"
BOP SX-2, R-4, "Essential Service Water Pump Shutdown"
BOP SX-8, R-0, "Drains the Essential Service Water System"
BOP VA-1, R-2, "Auxiliary Building HVAC System Operation"
BOP VA-3, R-2, "Auxiliary Building HVAC System Shutdown"
BOP VC-1, R-2, "Startup of Control Room HVAC System"
BOP VE-1, R-1, "Startup of Miscellaneous Electric Equipment Room Ventilation System"
BOP VI-4, R-0, "Radwaste and Remote Shutdown Control Room HVAC System Shutdown"
BOP VP-5, R-, "Reactor Containment Fan Cooler Startup"
BOP VP-6, R-1, "Reactor Containment Fan Cooler Shutdown"
BOP VQ-2, R-5, "Containment Purge System Shutdown"
BOP VQ-3, R-2, "Containment Mini Purge Startup"
BOP VX-6, R-0, "ESF Division 11 (21) HVAC Shutdown"
BOP WX-45, R-1, "Release Tank Recirculation and Discharge"

d. The following Surveillance Procedures were selected for review:

1) Chemistry Surveillance Procedures

BCS 1.2.5.a.1-1, R-0, "Unit-1 Borated Water Source at Shutdown - Weekly"
BCS 4.7-1, R-0, "Unit-1 Reactor Coolant System Chemistry - Once per 72 Hours"
BCS 5.1.1.b-1, R-0, "Unit-1 Accumulator Boron Concentration Monthly and Shift Engineer Request"
BCS 5.4-1, R-0, "Unit-1 Refueling Water Storage Tank Boron - Weekly"
BCS 7.1.4-2, R-0, "Unit-1 Secondary Coolant System Dose Equivalent Iodine - 131 Activity - Semi-Annual"
BCS 7.9.2-1, R-0, "Plant Systems Sealed Source Contamination - Semi-Annual/Startup/Repair"
BCS 9.1.2-1, R-0, "Unit-1 Refueling Operations Boron Concentration Once per 72 Hours
BCS 12.2-1, R-0, "Radiological Environmental Monitoring, Land Use Census - Annual

(2) Operating Surveillance Procedures

BOS 1.1.4.a-1, R-2, "RC System Minimum Temperature for Criticality Surveillance"
BOS 1.1.4.b-1, R-1, "RC System Minimum Temperature for Criticality Surveillance"
BOS 1.3.1.2-1, R-0, "Moveable Control Assemblies Monthly Surveillance"
BOS 1.3.5-1, R-0, "Shutdown Rod Insertion Limit During Approach to Criticality Surveillance"
BOS 1.3.6-1, R-0, "Control Rod Insertion Limit Surveillance"
BOS 3.1.1-2, R-0, "Calorimetric Calculation Surveillance"
BOS 3.1.1-20, R-0, "Train A Solid State Protection System Bi-Monthly Surveillance"
BOS 3.2.1-12, R-0, "Reactor Trip P-4 Contacts
BOS 4.3.2-1, R-1, "Pressurizer Heaters Quarterly Surveillance"
BOS 4.3.3-1, R-1, "Pressurizer Heaters 18 Month Surveillance"
BOS 4.6.2.1.d-1, R-1, "RCS H_2O Inventory Balance Surveillance"
BOS 4.9.1.1-1, R-1, "RCS Pressure/Temperature Limit Surveillance"
BOS 6.2.1.a-1, R-1, "Containment Spray System Valve Lineup Monthly Surveillance
BOS 7.1.2.2-1, R-1, "Auxiliary Feedwater Flowpath Operability Surveillance Following Cold Shutdown"
BOS 8.1.1.2.a-1, R-2, "1A Diesel Generator Operability Monthly Surveillance"
BOS 9.1.1-1, R-1, "RCS Refueling Reactivity Limit Surveillance"
BOS 9.6.1-1, R-0, "Refueling Machine Manipulator Crane Operability Surveillance"
BOS 9.6.2-1, R-0, "Refueling Machine Auxiliary Hoist and Load Indicator Operability Surveillance"
BOS 9.10-1, R-0, "Refueling Cavity H_2O Level Surveillance"
BOS FW-2, R-0, "Motor Driven Feedwater Pump Operability Monthly Surveillance"

BOS DC-13, R-0, "Unit 1 125 VDC ESF Batteries Monthly Surveillance
BOS IS-, R-2, "Security Diesel Generator Monthly Operability Surveillance"
BOS MS-4, R-0, "Steam Dump Valve Operability Quarterly Surveillance"
BOS NR-1, R-0, "Power History Hourly Surveillance"
BOS SX-1, R-0, "Ultimate Heat Sink Make-Up Operability Quarterly Surveillance"
BOS XCC-1, R-1, "Caution Card Annual Surveillance"
BOS XFT-1, R-0, "Freezing Temperature Equipment Protection"
BOS XLE-1, R-0, "Locked Equipment Annual Surveillance"

(3) Technical Specification Surveillance Procedures

BVS 4.4.2-1, R-0, "ASME Surveillance Requirements for PORV's"
BVS 4.5.0-1, R-0, "Eddy Current Testing of Steam Generator U-Tubes"
BVS 0.5-3.CC.1, R-0, "ASME Surveillance Requirements for Component Cooling Pumps"
BVS 0.5-3.AF.1, R-0, "ASME Surveillance Requirements for Auxiliary Feedwater Pumps"
BVS 0.5-3.AB.1, R-1, "Test of the Boric Acid Transfer Pumps and Associated Discharge Check Valves"
BVS 0.5-3.DO.1, R-0, "ASME Requirement for Test of the Diesel Oil Transfer System"
BVS 0.5-2.RC.3, R-0, "Reactor Vessel Heat Vent Valves Indication Test"
BVS 0.5-2.RY.3, R-0, "Pressurization System Valve Indication Test"
BVS 0.5-2.SI.1, R-0, "Safety Injection System Valve Stroke Test"
BVS 0.5-2.AF.3, R-1, "Auxiliary Feedwater Valve Indication Test"
BVS 6.2.3.b-1, R-0, "Containment Cooling Fan Automatic Actuation Test"
BVS 6.2.1.d-1, R-0, "Containment Spray System Nozzle Flow Test"
BVS 6.3.3-3, R-0, "Component Cooling Isolation Valve Stroke Test"
BVS 7.1.1-1, R-0, "Main Steam Safety Valves Operability Test"
BVS 7.1.2.1.b.1-1, R-1, "Auxiliary Feedwater Valve Emergency Actuation Signal Verification Test"
BVS 7.10.1.1.f.2-1, R-0, "Fire Protection Manual Valve Stroking"
BVS 7.10.2.d-1, R-0, "Diesel Fuel Oil Storage Tank Rooms Foam Spray Headers and Deluge Nozzles Air Flow Test"

e. The following Fuel Handling Procedures were selected for review:

BFP FH-3, R-2, "Movement From New Fuel Storage Vault to New Fuel Elevator"

BFP FH-4, R-1, "Transfer of Fuel From New Fuel Elevator to Spent Fuel Racks or Fuel Transfer System"
BFP FH-5, R-1, "Fuel Movement in Containment"
BFP FH-14, R-1, "Operation of Manipulator Crane"
BFP FH-20, R-2, "Operation of Fuel Handling Building Crane"

f. As a result of this review, the inspector identified items of concern and provided comments to the licensee for consideration. Because many of the procedures are the initial procedures and need to go through a trial period after the plant becomes operational, the inspector anticipated subsequent reviews by the licensee as the procedures are put to use.

Licensee's review, resolution and response to the identified items for BGP and BOS procedures has not been completed yet and the inspector will review this information and document this in a subsequent report.

No items of noncompliance or deviation were identified.

10. Emergency Procedure Review (42452B)

a. General

The inspector's review included abnormal operating procedures and control room annunciator response procedures. Procedures were reviewed for compliance and consistency with Proposed Technical Specifications, Regulatory Guide 1.33, ANSI N18.7-176/ ANS-3.2 and Byron Emergency/Abnormal/Critical Safety Function Procedure Writers Guide, BAP 1310-A3.

b. The following Abnormal Operating Procedures were selected for review:

1BOA ELEC-4, R-0, "Loss of Offsite Power, Unit 1 for Modes 30-4"
0BOA ENV-1, R-1, "Operation During Tornado or Sustained Wind Conditions, Unit 0,1,2"
1BOA INST-1, R-1, "Nuclear Instrumentation Malfunction, Unit 1"
1BOA PRI-1, R-1, "Excessive Primary Plant Leakage, Unit 1"
1BOA PRI-2, R-2, "Emergency Boration, Unit 1"
1BOA PRI-4, R-1, "High Reactor Coolant Activity, Unit 1"
1BOA PRI-5, R-2, "Control Room Inaccessibility, Unit 1"
0BOA REFUEL-1, R-1, "Fuel Handling Emergency, Unit 0,1,2"
1BOA REFUEL-2, R-1, "Reactor Cavity Level Loss, Unit 1"
1BOA ROD-2, R-1, "Failure of Rods to Move, Unit 1"
1BOA SEC-11, R-2, "High Temperature in AF Nozzle Piping, Unit 1"

c. The following Control Room Annunciator Response Procedures were selected for review:

BAR 1-1-A2, R-1, "CNMT Dain Leak Detect Flow High"
BAR 1-1-A4, R-1, "Feedwater Isolation"
BAR 1-1-C1, R-2, "Spent Fuel Pit Level High Low"
BAR 1-2-A4, R-1, "CC Pump Trip"
BAR 1-3-A4, R-2, "CS Actuation"
BAR 1-3-C4, R-2, "CNMT Press Ht-2"
BAR 1-4-E3, R-2, "Misc Cont Cab Doors Open"
BAR 1-5-A7, R-1, "CNMT Phase B Isolation"
BAR 1-5-D2, R-1, "Accum ID Press High Low"
BAR 1-6-B7, R-1, "RWST Level LO-2"
BAR 1-6-D2, R-2, "RH Pump 1B CC Flow Low"
BAR 1-7-C2, R-1, "RCP Lower BRNG Temp High"
BAR 1-7-E3, R-1, "RCP Therm Barr CC Wtr Temp High"
BAR 1-8-B5, R-1, "Ltdn HX Outlet Press High"
BAR 1-9-B2, R-1, "VCT Press High Low"
BAR 1-9-D3, R-1, "CHG Line Flow High Low"
BAR 1-10A1, R-1, "SR S/D Flux High"
BAR 1-10-C3, R-2, "Pwr Rng Flux Rate RX Trip Alert"
BAR 1-10-E3, R-1, "BDPS Flux Doubled"
BAR 1-11-B1, R-1, "Manual SI/RX Trip"
BAR 1-11-C3, R-1, "PZR Press Low RX Trip"
BAR 1-11-E1, R-1, "CNMT Press High SI/RX Trip"
BAR 1-12-B2, R-2, "PZR PORV Or SAF Vlv Open"
BAR 1-12-D6, R-2, "PZR SAF Rlf Dsch Temp High"
BAR 1-13-A4, R-1, "Loop 1A RTD Byp Flow Low"
BAR 1-13-D1, R-1, "ID Byp Vlv Open with Stop Valves Open"
BAR 1-14-B5, R-1, "Loop 1B DT Dev Low"
BAR 1-4-E4, R-1, "RX Head Vent Temp High"
BAR 1-15-C5, R-2, "S/G 1C Lvl LO-2 RX Trip Alert"
BAR 1-15-D12, R-1, "SIG ID FWIV Byp Temp Low"
BAR 1-15-E5, R-2, "Low TAVE With P-4 FW Isol"
BAR 1-16-B1, R-1, "FW Pump 1B Trip"
BAR 1-16-D1, R-1, "FW Pump Suct Hdr Press Low"
BAR 1-17-A13, R-1, "CW Pump Trip"
BAR 1-17-B12, R-1, "CNDSR Emergency Makeup Vlv Open"
BAR 1-17-D13, R-1, "Intake Bay Level High Low"
BAR 1-18-B1, R-1, "S/G 11B Level Hi-2 Turb Trip"
BAR 1-18-D4, R-1, "CNDSR Vacuum Low"
BAR 1-18-H1, R-1, "RX Trip Turb Trip"
BAR 1-18-E2, R-1, "CNDSR Vacuum Low Turb Trip"
BAR 1-19-A4, R-1, "Main Xfmr Diff Gen Trip"
BAR 1-19-C5, R-2, "UAT Overcurrent Gen Trip"
BAR 1-20-A1, R-2, "Loss of Off Site Power"
BAR 1-20-E9, R-2, "DG Fuel Oil Sto Tank Level Low"
BAR 1-21-B9, R-1, "DG 1A Overload"
BAR 1-21-D9, R-1, "DG 1A Diff Lockout/Overspeed"
BAR 1-22-B8, R-1, "Brkr 1424 Cross-Tie Diff Overcurrent"
BAR 1-22-D6, R-1, "125 VDC Bus 112 Ground"
BAR 0-33-A5, R-1, "CROM Exh Fan Trip"
BAR 0-33-A6, R-1, "MCR M/U Air Radiation High"

BAR 0-33-05, R-1, "RX Cav Fan 1B Trip DP Low"
BAR 0-33-E5, R-1, "Neutron Det Cav Temp High"
BAR 0-34-C1, R-1, "DG Room 1A Ionization High"
BAR 0-34-E3, R-1, "Unit 1 CSR Door Open"
BAR 0-37-B7, R-1, "SX Makeup PP Auto Start"
BAR 0-37-E6, R-3, "SX CLG TWR Fan Vibration High"
BAR 0-38-B7, R-1, "Fire Pump 0A Running"
BAR 0-38-E5, R-1, "Accelerograph Accel High"
BAR 0-39-E3, R-1, "Emer Breathing Air Press Low"

d. As a result of this review, the inspector identified items of concern and provided comments to the licensee for consideration.

   Licensee's review, resolution and response to the identified items for BCA procedures has not been completed yet and the inspector will review this information and document this in a subsequent report. No items of noncompliance or deviation were identified.

11. <u>Operational Staffing</u>

   a. The inspector determined, by review of applicable Byron Final Safety Analysis Report (FSAR) Sections and personnel records, that the staff positions have been filled with personnel possessing the ANSI 18.1 requisite education, experience, health and skills or qualifications commensurate with the level of responsibility for the following positions:

      Shift Engineers
      Shift Foremen
      Station Control Room Engineers
      Fuel Handling Foremen
      Master Mechanic
      Master Electrician
      Operating Engineer
      Quality Supervisor
      Health Physics Foremen
      Chemistry Foremen
      Mechanical Maintenance Foremen
      Electrical Maintenance Foremen
      Instrument Maintenance Foremen
      Head Nuclear Engineer
      Master Instrument Mechanic
      Radiation/Chemistry Supervisor
      Nuclear Station Operators

At least 2 or a 10% selective sample of:

Equipment Operators
Equipment Attendants
Radiation Control Technicians
Senior Nuclear Mechanics
Senior Nuclear Electricians
Control System Technicians
Fuel Handler "A"
Electrician "A"
Electrician "B"

No items of noncompliance or deviation were identified.

12. Plant Tours/Housekeeping

The inspectors conducted plant tours on August 5, 7, 8, 15, 23, September 5, 9, 23, 24, 26, 27, 30, 1984. The areas of the plant observed during the tours included Unit 1 and 2 containments, control room, fuel handling and storage areas, auxiliary building areas and the 1A diesel generator room. Areas were inspected for work in progress, state of cleanliness resulting from construction work, overall housekeeping, state of fire protection equipment and methods being employed, and the care and preservation of safety-related components and equipment. The inspectors were accompanied by licensee personnel on portions of the tours for the purpose of identifying areas where additional housekeeping efforts should be concentrated to bring the overall cleanliness of Unit 1 spaces up to par with the current state of construction. Inspector concerns were related to the licensee. No items of noncompliance or deviation were identified.

13. Meetings

A meeting was held August 8, 1984 between NRC and Commonwealth Edison staff. The status of open inspection issues and responsibilities for resolution were discussed. The purpose was to coordinate efforts in preparation for licensing of Unit 1.

CECO Attendees	NRC Attendees
R. Tuetken	J. Streeter
D. St. Clair	C. Ramsey
T. Tramm	D. Danielson
R. Klingler	K. Connaughton
R. Poche	W. Guldemond
D. Farrar	J. Hinds
M. Loehmann	R. Lerch
M. Snow	R. Love
	R. Hasse
	J. Belanger
	P. Brochman

14. Exit Interview

The inspectors met with licensee representatives denoted in paragraph 1 at the conclusion of the inspection on October 3, 1984. The inspectors summarized the purpose and the scope of the inspection and the findings noted in this report.

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

ENCLOSURE 2

DEC

Docket No. 50-454
Docket No. 50-455
License No. CPPR-130
License No. CPPR-131
EA 84-81

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 Messrs. D. W. Hayes and K. A. Connaughton during the period of April 26 through July 17, 1984 of activities at Byron Station, Units 1 and 2, authorized by NRC Licenses No. CPPR-130 and CPPR-131. The purpose of the inspection was to determine the effectiveness of Commonwealth Edison Company's (CECo) corrective actions relating to deficiencies in equipment supplied by Systems Control Corporation (SCC). The results of this inspection were discussed on June 6, 1984 during an Enforcement Conference held in the NRC Region III office between you and other members of CECo and myself and other members of the NRC Region III and Headquarters staffs.

In its January 26, 1981 response to the Notice of Violation transmitted on December 30, 1980 with NRC Inspection Report Nos. 50-454/80-04; 50-455/80-04, CECo stated that, "For Systems Control Corporation, source inspection has been conducted for all safety-related equipment shipped since February 1980 and source inspection will be conducted on all future shipments involving Systems Control. These inspections have been conducted by the Pittsburgh Testing Laboratory (PTL) under the direction of the Byron Quality Assurance Department. The inspections cover welding, equipment identification, sealing of instrumentation lines and other specification requirements."

Inspection findings indicate that the above statement was false in that NRC inspectors identified numerous safety-related components which were shipped from SCC during the period February 1980 through January 26, 1981 and which

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had not been source inspected for welding by PTL or any other non-SCC organization. The statement was material in that the Region III staff relied upon the independent source inspections performed by PTL to assure the adequacy of CECo's corrective actions regarding safety-related equipment supplied by SCC. If the NRC had known that independent source inspections for welding were not being performed, they would have been required.

Two other statements in your January 26, 1981 letter were not as clear as they should have been. The first statement was made in conjunction with the material false statement discussed above and indicated that all future shipments of SCC equipment would be subject to source inspection. We understood that to mean that <u>all</u> items in <u>all</u> future shipments would be source inspected. CECo has since stated that the intent was to only do a sampling inspection of all shipments. However, considering the context of the statement, we believe that there was a basis for the NRC's interpretation. The second statement was that "...since January 1978 Commonwealth Edison has not made any purchases from Systems Control." While we did not identify any cases where new purchase orders had been issued to SCC after January 1978, we did identify several cases where additional purchases were made by changing existing purchase orders. Although citations are not included for these statements, it is our view that the statements were misleading and indicate a need for you to improve the clarity of future submittals to the NRC to remove ambiguities.

To emphasize the importance of the need to ensure accurate submittals to the NRC, I have been authorized, after consultation with the Deputy Director, Office of Inspection and Enforcement, to issue the attached Notice of Violation and Proposed Imposition of Civil Penalty in the amount of Forty Thousand Dollars for the violation set forth in the Notice. The violation has been categorized as a Severity Level III violation. Although the General Policy and Procedure for NRC Enforcement Actions, 10 CFR Part 2, Appendix C, as revised, 49 FR 8583 (March 8, 1984) states that the base civil penalty for a Severity Level III violation is $50,000, since this violation occurred prior to this revision and at that time the base civil penalty was $40,000, the staff has determined that a $40,000 civil penalty is appropriate.

You are required to respond to this letter and should follow the instructions in the Notice when preparing your response. Your reply to this letter and the results of future inspections will be considered in determining whether further enforcement action is warranted.

In accordance with 10 CFR 2.790, "Rules of Practice," a copy of this letter and the enclosure will be placed in the NRC Public Document Room.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Sincerely,

James G. Keppler

James G. Keppler
Regional Administrator

Enclosures:
1. Notice of Violation and Proposed Imposition of Civil Penalty
2. Inspection Report No. 50-454/84-32(DRP); No. 50-455/84-25(DRP)

cc w/encls:
D. L. Farrar, Director of Nuclear Licensing
V. I. Schlosser, Project Manager
Gunner Sorensen, Site Project Superintendent
R. E. Querio, Station Superintendent

Distribution
PDR
NSIC
LPDR
SECY
CA
JTaylor, IE
JAAxelrad, IE
GRKlingler, IE
JLieberman, ELD
VStello, DED/ROGR
FIngram, PA
JKeppler, RIII
Enforcement Coordinators,
 RI, RII, RIII, RIV, RV
SConnelly, OIA
BHayes, OI
HDenton, NRR
RStark, NRR
JCrooks, AEOD
EJordan, IE
NGrace, IE
IE:ES File
IE:EA File
EDO Rdg File
DCS
Resident Inspector, RIII Byron
D. W. Cassel, Jr.
Diane Chavez, DAARE/SAFE
W. Paton, ELD
L. Olshan, NRR LPM

Phyllis Dunton, Attorney
 General's Office, Environmental
 Control Division

IE:ES	RIII	ELD	IE:D	IE:DD	EDO
GKlingler	JKeppler	JLieberman	JAxelrad	JTaylor	WDircks
10/26/84	10/26/84	10/29/84	10/30/84	10/30/84	10/6/84

RIII	RIII	RIII	RIII
Streeter/jp	Schultz	Davis	Keppler
11/28/84	11-30-84	12/3	12/3/84

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Commonwealth Edison Company
Byron Station Units 1 & 2

Docket Nos. 50-454; 50-455
Licenses No. CPPR-130; CPPR-131
EA 84-81

A special NRC inspection was conducted at Byron Station Units 1 and 2 during the period April 26 through July 17, 1984 to determine the effectiveness of Commonwealth Edison Company's (CECo) corrective actions in connection with deficiencies in equipment supplied by Systems Control Corporation. During this inspection it was determined that CECo had not taken all the corrective actions described in its January 26, 1981 letter in response to a Notice of Violation issued December 30, 1980. The violation described below represents a lapse in management oversight and control of the accuracy of CECo submittals to the NRC. To emphasize the importance of this matter, the NRC proposes to impose a civil penalty in the amount of Forty Thousand Dollars. In accordance with the General Policy and Procedure for NRC Enforcement Actions, 10 CFR Part 2, Appendix C, as revised, 49 FR 8583 (March 8, 1984), and pursuant to Section 234 of the Atomic Energy Act of 1954, as amended ("Act"), 42 U.S.C. 2282, PL 96-295, and 10 CFR 2.205, the particular violation and the associated civil penalty are set forth below:

In its January 26, 1981 letter in response to the Notice of Violation issued December 30, 1980 for Noncompliance Items No. 454/80-04-01; 455/80-04-01, the licensee stated, in part, "For Systems Control Corporation, source inspection has been conducted for all safety-related equipment shipped since February 1980 and source inspection will be conducted on all future shipments involving Systems Control. These inspections have been conducted by Pittsburgh Testing Laboratory (PTL) under the direction of the Byron Quality Assurance Department. The inspections cover welding, equipment identification, sealing of instrumentation lines and other specification requirements."

Contrary to the above statement, all safety-related equipment shipped since February 1980 had not been source inspected for welding by PTL or any other non-SCC organization as of January 26, 1981. Equipment not source inspected for welding during the period February 1980 through January 26, 1981 included two main control boards, four dc fuse panels, a cable pan hanger and numerous welded items covered by Material Receiving Reports 8453, 8773, 8907, 8964 and 9283.

The above statement constituted a material false statement within the meaning of Section 186 of the Atomic Energy Act of 1954, as amended. The statement was false in that NRC inspectors identified numerous safety-related components which were shipped from SCC during the period February 1980 through January 26, 1981 and which had not been source inspected for welding by PTL or any other non-SCC organization. The statement was material in that the Region III staff relied upon the independent source inspections performed by PTL to assure the adequacy of CECo's corrective actions regarding safety-related equipment supplied by SCC. If the NRC had known that independent source inspections for welding were not being performed, they would have been required.

This is a Severity Level III violation (Supplement VII).
(Civil Penalty - $40,000)

Pursuant to the provisions of 10 CFR 2.201, Commonwealth Edison Company is hereby required to submit to the Deputy Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 799 Roosevelt Road, Glen Ellyn, IL 60137, within thirty days of the date of this Notice, a written statement or explanation in reply, including for the alleged violation: (1) admission or denial of the alleged violation; (2) the reasons for the violation, if admitted; (3) the corrective steps that have been taken and the results achieved; (4) the corrective steps that will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, Commonwealth Edison Company may pay the civil penalty in the amount of $40,000 or may protest imposition of the civil penalty in whole or in part, by a written answer. Should Commonwealth Edison Company fail to answer within the time specified, the Deputy Director, Office of Inspection and Enforcement will issue an order imposing the civil penalty proposed above. Should Commonwealth Edison Company elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, such answer may: (1) deny the violations listed in this Notice, in whole or in part; (2) demonstrate extenuating circumstances; (3) show error in this Notice; and (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty, in whole or in part, such answer may request remission or mitigation of the penalty. In requesting mitigation of the proposed penalty, the five factors contained in Section V(B) of 10 CFR Part 2, Appendix C should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanations by specific reference (e.g., giving page and paragraph numbers) to avoid repetition. The attention of Commonwealth Edison Company is directed to the other provisions of 10 CFR 2.205, regarding the procedures for imposing a civil penalty.

Upon failure to pay any civil penalty due, which has been subsequently determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Keppler

James G. Keppler
Regional Administrator

Dated at Glen Ellyn, Illinois
this 4th day of December 1984

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-454/84-32(DRP); 50-455/84-25(DPP)

Docket Nos. 50-454; 50-455 Licenses No. CPPR-130; CPPR-131

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

Facility Name: Byron Station, Units 1 and 2

Inspection At: Byron Station, Byron, IL

Inspection Conducted: April 26 through July 17, 1984

Inspectors: J. F. Streeter for D. W. Hayes
J. F. Streeter for K. A. Connaughton

Approved By: J. F. Streeter
J. F. Streeter, Director
Byron Project Division

7/30/84
Date

Inspection Summary

Inspection on April 26 through July 17, 1984 (Report No. 50-454/84-32(DRP); 50-455/84-25(DRP))

Areas Inspected: Special unannounced safety inspection of corrective actions taken in response to Noncompliance Items No. 454/80-04-01; 455/80-04-01; SCC problems and licensee corrective actions.

Results: Of the two areas inspected, three items of noncompliance were identified (failure to document nonconforming conditions and control nonconforming items (2 examples) - Paragraphs 3.d(2)(c) and 4.i; failure to include SCC on the Approved Bidders List as a supplier of equipment and purchase of SCC equipment - Paragraph 4.c; and failure to take timely and effective corrective actions to ensure SCC weld problems were corrected - Paragraph 4.i). The inspection consisted of 221 inspector-hours on site by two NRC inspectors including 32 inspector-hours during off-shifts.

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DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

**B. Thomas, Executive Vice President
**C. Reed, Vice President, Nuclear Operations
°**L. DelGeorge, Assistant Vice President, Licensing and Engineering
°**T. Maiman, Manager of Projects
**W. Shewski, Manager of Quality Assurance
**B. Shelton, Project Engineering Manager
**D. Farrar, Director of Nuclear Licensing
°**T. Tramm, Nuclear Licensing Administrator
**J. Westermeier, Project Engineer
**A. Zecto, Purchasing Agent
#V. Schlosser, Byron Project Manager
**#G. Marcus, Director of Quality Assurance
#G. Sorensen, Project Construction Superintendent
**#K. Hansing, Quality Assurance Superintendent
°**#R. Tuetkin, Byron Startup Coordinator
#R. Klingler, Project Quality Control Supervisor
#M. Lohman, Assistant Project Construction Superintendent
#J. Binder, Project Electrical Supervisor
#J. Bergner, Quality Assurance Staff

Hatfield Electric Company

T. Hill, Quality Control Supervisor
J. Spangler, Lead Weld Inspector

Sargent and Lundy Engineers

°K. Kostal, Assistant Manager, Structural Department

# Denotes those present at the exit meeting of May 14, 1984.

** Denotes those present at the Enforcement Conference conducted on June 6, 1984.

° Denotes those present at the technical meeting on July 17, 1984.

2. Background

Systems Control Corporation (SCC) was a supplier of both safety-related and nonsafety-related cable pans and fittings, cable pan supports (hangers), local instrument panels, main control board sections, and vertical panels. SCC began shipping safety-related equipment to the Byron site in January 1977.

On various occasions from early 1977 through March 1984, both the licensee and the NRC identified deficiencies in SCC's quality assurance program and its implementation. These quality assurance program deficiencies included repeated instances of nonconformance in the areas of weld quality, dimensional accuracy, protective coatings, and general workmanship. The purpose of this special inspection was to determine if corrective actions were of sufficient scope and depth to ensure that installed equipment supplied by SCC was of acceptable quality.

3. Licensee Action on Previous Inspection Findings - Noncompliance Items No. 454/80-04-01; 455/80-04-01)

a. General

The licensee's January 26, 1981, response to Noncompliance Items No. 454/80-04-01; 455/80-04-01 described corrective actions taken and commitments to take additional corrective actions to prevent recurrence of problems related to equipment supplied by SCC. Certain of these actions were selected for verification by the inspectors. The actions selected for verification and the NRC findings relative to each are discussed in Paragraphs 3.b through 3.e below.

b. (1) Item From Licensee Response

"Corrective action has been completed for the Local Instrument Panels. Nonconformance Reports F-474 and F-484 covering this were closed on 10/21/80."

(2) NRC Findings

Based on a review of Pittsburgh Testing Laboratory (PTL) visual weld inspection records, Midway Industrial Contractors, Inc. daily coating work inspection records, and material receiving reports the inspectors verified that corrective actions specified in the NCRs were satisfactorily accomplished.

c. (1) Item From Licensee Reponse

"For the Main Control Boards, engineering analysis to determine disposition has been initiated under NCR F-544 dated 8/8/80."

(2) NRC Findings

CECo NCR F-544 for the Unit 1 Main Control Boards and Panels was closed by the licensee based upon the completion of inspection and weld mapping by S&L and Westinghouse, completion of required modification, and analysis by Westinghouse. Region III review of those corrective actions is in progress. This is an open item (454/84-32-01; 455/84-25-01) pending completion of the Region III review.

d. (1) Item From Licensee Response

"For Systems Control Corporation, source inspection has been conducted for all safety-related equipment shipped since February 1980 and source inspection will be conducted on all future shipments involving Systems Control. These inspections have been conducted by Pittsburgh Testing Laboratory under the direction of the Byron Quality Assurance Department. The inspections cover welding...."

(2) NRC Findings

The inspectors reviewed the following documents:

- All PTL visual welding inspection reports pertaining to equipment supplied by SCC.
- All material receiving reports (MRRs) for main control boards and vertical panels, cable pan hangers, and cable pans and fittings supplied by SCC since February 1980.
- Packing lists which identified shipping dates and all items shipped. (All items other than cable pans and fittings were individually identified whereas cable pans and fittings were only identified by type and quantity.)

(a) Between February 1980 and January 26, 1981, the following safety-related items were shipped from SCC without a source inspection by PTL for weld quality:

- Main Control Boards	2PM04J 2PM11J
- D.C. Fuse Panels	1DC10J 1DC11J 2DC10J 2DC11J
- Cable Pan Hanger	22-HV4-2-3285C
- Cable Pans and Fittings (all welded items in shipments)	MRR 8453 MRR 8773 MRR 8907 MRR 8964 MRR 9283

Failure to perform source inspections for weld quality for the above safety-related items is contrary to a statement in the licensee's January 26, 1981, response. This matter was a subject of the Enforcement Conference conducted in Region III on June 6, 1984 (see Paragraph 8 of this report). This matter is under NRC review for possible enforcement action. This is an unresolved item (454/84-32-02A; 455/84-25-02A).

(b) Not all items shipped from SCC after January 26, 1981, were inspected for weld quality; however, each shipment was subject to a sampling inspection. This was not inconsistent with the licensee's response since the response indicated that "...source inspection will be conducted on all future shipments..." but did not specifically indicate that all items in each shipment would be source inspected.

Region III understood the licensee's statement to mean that all items in all future shipments would be source inspected by an independent party. Considering the full sentence which included the statement, it appears there was a clear basis for the Region III understanding. Region III may or may not have accepted a sampling approach. Although the sampling approach was not contrary to regulatory requirements, there is evidence that it was not entirely effective in identifying nonconforming conditions on SCC equipment shipped after January 26, 1981.

(c) PTL Visual Weld Inspection Report No. 3592 dated February 17, 1981, was annotated to indicate that the 41 items inspected and rejected were from SCC Shipment No. 195. Included with the report was a packing list which indicated the shipping date was January 30, 1981, and that the shipment was covered by MRR 9778. The licensee's MRR log indicated that the MRR was voided; however, MRR 9778 could not be retrieved by the licensee. Therefore, the disposition of the 41 items found rejectable by PTL as well as numerous other items indicated on the packing list was not known. Failure to maintain adequate records of nonconforming items is an item of noncompliance (454/84-32-03; 455/84-25-03).

e. (1) Item From Licensee's Response

"...since January 1978 Commonwealth Edison has not made any purchases from Systems Control. ...Systems Control has been barred from procurement activity involving safety-related purchases for an indefinite period."

(2) NRC Findings

(a) Based upon a review of licensee Purchase Orders 20038, 219596, 201534 and their respective change orders, the inspectors determined that between January 1978 and

January 26, 1981, the licensee had ordered additional safety-related items from SCC by using change orders to add new items to existing purchase orders.

(b) Safety-related items were added to purchase orders for cable pan, hangers and main control boards and panels after January 26, 1981, via change orders.

The NRC findings in this area were subjects of the Enforcement Conference conducted in Region III on June 6, 1984 (see Paragraph 8 of this report). This matter is under NRC review for possible enforcement action. This is an unresolved item (454/84-32-02B; 455/84-25-02B).

4. Review of SCC Problems and Licensee Corrective Actions

a. General

The inspectors reviewed the engineering specifications defining the scope of SCC work and procurement documents in order to determine what equipment was supplied to Byron by SCC and to establish the time frames during which the various types of equipment were supplied. The inspectors also reviewed documentation of licensee QA/QC activities relative to SCC including initial and periodic reviews of the SCC QA/QC program, audits of SCC QA/QC program implementation, and inspections of SCC equipment. This review was conducted to determine if the licensee had obtained appropriate corrective actions for the specific problems identified and to determine if there were any trends indicating corrective actions were not adequate to ensure SCC supplied equipment was of acceptable quality.

b. Review and Approval of SCC's QA/QC Program

Based upon discussions with licensee personnel and document review, the inspectors determined that the licensee's QA organization and engineering department had conducted initial reviews of the SCC QA/QC program and resolved any identified deficiencies prior to the awarding of bids. The licensee conducted similar reviews of changes to the program. These reviews did not include reviews of detailed implementing procedures. The licensee's QA program review procedures were subsequently upgraded to require more detailed reviews. Each licensee review was conducted in accordance with the latest procedures.

c. Inclusion of SCC on the Licensee's Approved Bidders List (ABL)

The licensee's QA and engineering organizations determined SCC's QA program to be acceptable for the scope of work defined by LaSalle engineering specification J-2560 (cable pans and hangers) early in 1975. Based on that determination, SCC was then added to the licensee's ABL on July 16, 1975, as a supplier of safety-related cable pans and hangers. The licensee's QA and engineering organizations determined SCC's QA program to be adequate for the scope of work defined by engineering specification F/L 2788 (main control boards and vertical panels) in April 1977 and for the scope of work defined by engineering specification F/L 2809 (local instrument panels) in November 1977; however, as a result of an apparent administrative error SCC was not added to the licensee's ABL as a supplier of equipment encompassed by the latter two specifications. While the licensee performed all actions prerequisite to including SCC on the ABL as a supplier of all equipment types ultimately purchased, the ABL was not updated as required prior to the awarding of bids. This condition apparently went undetected by the licensee until the time of this inspection.

In January 1984 the licensee removed SCC from the ABL as a supplier of safety-related cable pans and hangers. Licensee QA personnel indicated it was intended that SCC be removed from the ABL for all equipment types. Since SCC had only been included on the ABL as a supplier of cable pans and hangers, the removal of SCC as a supplier of that equipment constituted total removal of SCC from the ABL. However, on May 10, 1984, the licensee issued Change Order No. AM to Purchase Order 207534 which added eight safety-related combination indicator light/control switches to that SCC Purchase Order.

Failure to include SCC on the ABL for all equipment types purchased, and purchase of safety-related items from SCC after removal from the ABL is an item of noncompliance (454/84-32-04; 455/84-25-04).

d. <u>Licensee QA and Station Nuclear Engineering Department Audits and Surveillances</u>

The licensee conducted numerous audits of SCC's QA program implementation (including inspections of SCC equipment) over the time period in which SCC supplied equipment to Byron. The inspectors reviewed documentation of all audits and surveillances conducted by the licensee of SCC since SCC began supplying equipment to Byron. Based upon this review, the inspectors determined that the licensee sought and obtained some measure of corrective action for all identified deficiencies. Corrective actions in all cases except those involving weld quality problems appeared to be appropriately implemented such that affected equipment was verified to be of acceptable quality and repetition of the problems was minimized. However, for certain identified weld quality problems corrective actions were untimely and ineffective. Discussions of these weld quality problems are summarized in Paragraphs 4.e through 4.i.

e. <u>Welding Problems - General</u>

The engineering specifications governing equipment supplied to Byron by SCC require that welds conform to the American Welding Society

AWS D1.1 Code. Nonconforming welds as well as missing welds have repeatedly been identified on all equipment types supplied by SCC.

For the main control boards, vertical panels, and local instrument panels, the licensee performed 100% reinspection of welds by personnel other than SCC QC personnel to ascertain weld quality. Deficient welds were either repaired or subject to engineering evaluations to ensure the equipment was acceptable. As discussed in Paragraph 3 of this report, NRC reviews of engineering evaluations of the main control boards and vertical panels are not yet complete and the results of the NRC reviews will be documented in a future NRC inspection report.

f. Cable Pans

For straight cable pans, the only welds made by SCC were cable pan stiffener attachment welds. The licensee reported during the technical meeting on July 17, 1984, that an analysis had been performed which demonstrated that the cable pan stiffeners were not required. The results of NRC reviews of the licensee's analysis will be documented in a future NRC inspection report. This matter remains open pending receipt of the licensee's analysis and NRC review (454/84-32-05; 455/84-25-05).

g. Cable Pan Fittings

In mid 1977 the licensee conducted a review of weld deficiencies identified on a sample of cable pan fittings which were inspected to assess the adequacy of all fitting welds. The licensee concluded that the identified deficiencies did not violate design requirements. In response to questions raised by the inspectors during this inspection, the licensee stated during the meeting on July 17, 1984, that, based upon a recent evaluation of fitting welds for structural significance, the only fitting welds required to meet design bases were the outboard vertical form welds on 90° fittings, and only then in the instance where outboard stiffener attachment welds were missing. The licensee stated that a 100% inspection of 90° fittings would be performed to determine if the outboard vertical form welds are present. If any of these welds are found to be absent, the outboard stiffener welds will be examined to determine if they are adequate. If necessary, the licensee will make repairs to the fittings. The licensee stated that the evaluation of fitting welds for structural significance and the results of inspections of 90° fittings would be provided for NRC review. This matter remains open pending receipt of the licensee's analysis and NRC review (454/84-32-06; 455/84-25-06).

h. Ladder-Type Cable Pans and Fittings

Prior to this inspection the licensee had not evaluated the adequacy of welds on ladder-type cable pans and fittings. As a result of questions raised by the inspectors the licensee performed an evaluation of weld quality on a sample of these items. A sample of 16 ladder-type pans and 10 fittings (containing over 300 welded connections) were inspected for weld quality. The inspection results were evaluated and the welded connection with the largest reduction in strength due to discrepancies was identified. This worst case condition was then assumed to exist on all welded connections and evaluated against design requirements. The licensee concluded that even with this assumption, the ladder pans and fittings were acceptable as-is. This matter remains open pending NRC review of the licensee's evaluation (454/84-32-07; 455/84-25-07).

i. Cable Pan Hangers

Nonconforming cable pan hangers supplied since May 1977 were identified on numerous occasions by the electrical installation contractor (HECO) and licensee QA/QC personnel and documented in nonconformance reports. Licensee personnel stated that these nonconformance reports dealt with very small numbers of items and were not indicative of a generic problem. Therefore, the disposition of these nonconformance reports only involved repair and reinspection of the identified items. However, in August 1982 as a result of some identified welding deficiencies the licensee directed HECO to inspect all hangers stored in the laydown area to verify the welds met requirements. The inspections were performed pursuant to HECo QA/QC Memorandum No. 345 and identified a number of hangers (approximately 30) as deficient. The hangers were repaired, reinspected, and found acceptable. The deficient hangers were not documented by NCR as required which may have accounted for the licensee not recognizing hanger weld quality as a persistent generic problem. Failure to document the nonconforming conditions as required is an example of a noncompliance (454/84-32-03; 455/84-25-03).

In August 1983 approximately 60 hangers with one or more weld deficiency were identified by the on site electrical installation contractor's QC inspection personnel. These deficiencies were reported to the licensee on August 29, 1983. As a result, the licensee issued Nonconformance Reports (NCR) Nos. F-850 in September 1983 and F-885 in February 1984 to address the generic implications of the deficiencies. To resolve these NCRs, the licensee selected a random sample of 80 hangers, subjected them to weld inspections, and evaluated them for structural adequacy. Based upon these evaluations, the licensee concluded that all hangers were acceptable. The evaluations did not apply the worst observed reduction in hanger connection strength caused by discrepant and/or missing welds to the most highly stressed connections in the plant. The licensee therefore did not satisfactorily demonstrate that all hangers in the plant were acceptable.

The licensee's corrective actions for cable pan hanger weld discrepancies are considered ineffective. The licensee was aware of numerous instances of nonconforming welds on cable pan hangers as well as other items which evidenced long standing deficiencies in SCC weld quality control practices. Corrective actions to address cable pan hangers supplied between May 1977 and February 1981 were untimely and ineffective. Failure to take timely and effective corrective actions to ensure the adequacy of cable pan hangers supplied by SCC is an item of noncompliance (454/84-32-08; 455/84-25-08). This is a repeat of Noncompliance Items No. 454/80-04-01; 455/80-04-01.

5. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Paragraphs 3.c(2), 4.f, 4.g, and 4.h.

6. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Unresolved items disclosed during the inspection are discussed in Paragraph 3.d(2)(a) and 3.e(2).

7. Exit Interview

The inspectors met with licensee representatives denoted in Paragraph 1 on May 14, 1984. The inspectors summarized the purpose and the scope of the inspection and findings. In a subsequent technical meeting on July 17, 1984, the licensee described evaluations completed and planned relating to SCC weld discrepancies and agreed to provide Region III with information including supporting analyses to enable Region III to assess the effectiveness of the licensee's corrective actions for equipment supplied by SCC.

8. Enforcement Conference

On June 6, 1984, an Enforcement Conference was held between members of the licensee's staff and the Region III staff. The Enforcement Conference was held to discuss the circumstances which led to the inclusion of certain statements in the licensee's response to Noncompliance Items No. 454/80-04-01; 455/80-04-01 which appeared to be false. The following statements contained in the licensee's response letter were discussed:

- "For Systems Control Corporation source inspection has been conducted for all safety-related equipment shipped since February 1980 and source inspection will be conducted on all future shipments involving Systems Control."

- "...since January 1978 Commonwealth Edison has not made any purchases from Systems Control."

Regarding the first statement, the licensee acknowledged that not all safety-related equipment shipped between February, 1980, and January 26, 1981, had been subject to source inspection. The licensee stated that source inspections had been conducted on at least a sample of each shipment after January 1981. The licensee representatives stated that it had always been the intent of CECo to only do a sampling inspection of each shipment.

Regarding the second statement, the licensee acknowledged that the statement was not as precise as it could have been but that the intent was not to allow Systems Control Corporation to bid on any additional engineering specifications. The statement was imprecise in that by amendment to existing specifications and by changes to existing Purchase Orders the licensee had purchased items in addition to those specified as of January 1978.

NRC representatives indicated that they would consider the information presented by the licensee when deciding if enforcement action is warranted.

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

ENCLOSURE 3

November 20, 1984

Mr. R. C. DeYoung, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Independent Design Inspection
NRC Inspection Report Nos. 50-454/83-32

References (a): October 1, 1984 letter from T. R. Tramm to R. C. DeYoung

(b): October 19, 1984 letter from J. Nelson Grace to Cordell Reed

(c): October 19, 1984 letter T. R. Tramm to R. C. DeYoung

Dear Mr. DeYoung:

This letter provides additional information regarding the actions taken in response to the NRC's Integrated Design Inspection (IDI) at Byron. The information presented here supplements that provided in reference (a) to address the NRC comments in reference (b).

Attachment A to this letter addresses the Staff comments regarding improvements in the A-E's documentation of the use of engineering judgements and the review of FSAR changes. It explains how the S&L QA program complies with Regulatory Guide 1.64 and ANSI N45.2.11 with respect to the documentation of design activities, especially the documentation of the design basis and engineering judgements. It also describes the manner in which S&L's engineers are trained in these procedures, particularly the procedures relating to engineering judgement and design change control.

Attachment D to reference (c) partially addressed the NRC comment regarding the process of FSAR review and revision to reflect design changes. Additional actions have been taken. Since, and as a result of, the Byron IDI and IDR, project personnel are more aware of the necessity to update the FSAR to reflect the design when design changes are made. The numerous correspondence and discussions relative to the subject has reinforced their on the job training relative to updating the FSAR. The subject has also been discussed and emphasized at meeting where both project and support personnel have attended.

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A Project Instruction under development for post fuel load design changes will reinforce the need to update the FSAR when design changes are made, by specifically requiring that the design change be evaluated for potential FSAR revision.

The NRC also requested additional details regarding the auxiliary building flooding analyses which were summarized in Attachment B to reference (c). Attachment B to this letter contains the documentation of this analysis and identified the disposition of items identified during the reviews.

Please direct further questions regarding these matters to this office.

One signed original and fifteen copies of this letter and the Attachments are provided for NRC review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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ATTACHMENT A

I. <u>COMPLIANCE OF THE SARGENT & LUNDY QUALITY ASSURANCE PROGRAM AND PROCEDURES WITH ANSI N45.2.11 AND REGULATORY GUIDE 1.64 WITH RESPECT TO DOCUMENTING DESIGN ACTIVITIES</u>

The Sargent & Lundy Quality Assurance Program which is described in the Topical Report SL-TR-1A, Rev. 6, has been approved by the Nuclear Regulatory Commission as meeting the criteria of Appendix B to 10CFR Part 50. The Topical Report states that S&L is committed to meeting and implementing the applicable provisions of Regulatory Guide 1.64, Revision 2, June 1976, Quality Assurance Requirements for the Design of Nuclear Power Plants (ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants) except as the provisions may be modified by a commitment in an applicable SAR.

The following is a list of pertinent sections of the S&L QA Program Topical Report that provides examples of how the pertinent sections of ANSI N45.2.11 are addressed:

<u>Pertinent Sections of ANSI N45.2.11</u>

Section 3 - Design Input Requirements
Section 4 - Design Process
Section 5 - Interface Control
Section 6 - Design Verification
Section 7 - Document Control
Section 8 - Design Change Control

<u>S&L QA Program (Topical Report SL-TR-1A, Rev. 6)</u>

Section 01, page 01-3, lines 09 through 19
Section 02, page 02-2, lines 13 through 22
Section 02, page 02-4, lines 01 through 16
Section 03, page 03-1 lines 01 through 34
pages 03-02; 03-03; 03-04; 03-05; 03-06; 03-07
Section 04, page 04-1, 04-2, 04-3 and 04-4.

We are also attaching for your information Table 0.2.04-1 "List Of General Quality Assurance Procedure" from the Topical Report. The following GQ procedures address the following pertinent sections of ANSI N45.2.11:

GQ Procedure	Pertinent Section of ANSI N45.2.11					
	3	4	5	6	7	8
GQ 3.01	x	x	x		x	x
GQ 3.02	x		x		x	x
GQ 3.03	x	x	x		x	x
GQ 3.04	x	x	x	x	x	x
GQ 3.05	x	x	x		x	x
GQ 3.06	x				x	
GQ 3.07	x	x	x	x	x	x
GQ 3.08	x	x		x	x	x
GQ 3.09	x	x		x	x	x
GQ 3 10	x	x	x	x	x	x
GQ 3.11	x	x		x	x	x
GQ 3.12		x			x	
GQ 3.13	x	x	x	x	x	x
GQ 3.14		x	x			
GQ 3.15	x	x	x	x	x	x
GQ 3.16	x	x	x	x	x	x
GQ 3.17	x	x	x	x	x	x
GQ 3.18	x	x	x	x	x	x
GQ 16.03		x	x	x	x	

The S&L Quality Assurance Program provides for control of S&L design and procurement activities which affect the quality of safety-related nuclear power plant structures, systems and components. It is S&L's policy that designs be in accordance with appliable quality assurance requirements and that design activities be procedurally controlled and documented. This includes training of personnel in quality-related S&L activities.

In addition to the QA procedures, the design of structures, systems, and components is planned and controlled by S&L Department Standards, Divisional Procedures and Project Instructions. Design processes are prescribed, accomplished and documented in accordance with procedures which establish the responsibilities and interfaces of design disciplines. Design procedures for control of changes, additions or deletions in design information require documentation and approval. The appropriate engineer is charged with the responsibility for defining other design documents affected by the change,and for resolving and coordinating changes from other disciplines whose design is affected.

Sargent & Lundy uses a system of planned and periodic audits of activities, records and facilities to verify compliance with, and to assess the effectiveness of, the various aspects of the S&L Quality Assurance Program and the implementing procedures. As part of the auditing process, samples of pertinent design documents requiring independent reviews are taken.

II. DOCUMENTATION OF DESIGN BASIS AND ENGINEERING JUDGMENT

Sargent & Lundy has addressed the need for documentation of design work as follows:

QA Procedure GQ-3.08, Design Calculations, contains requirements for the proper documentation of design basis including assumptions, formulae and steps used in the analysis. QA Procedure GQ-3.17, Design Information Transmittal has been issued to formalize the transmittal of design information between project team members in various design disciplines. The procedure covers any design input which is not already addressed in existing standards or procedures. It requires documentation of the basis for design information including identification of design input which is preliminary.

A nonconformance review program was initiated three years ago to identify trends. The Trend Review Report is issued by Quality Assurance Division every three months, identifying the nonconformances cited during the previous 12 months. This report addresses trends and recommends corrective action. In the last four Trend Review Reports, the subject, "documenting engineering judgment" was addressed. The corrective actions recommended in these reports were implemented.

The use of enginering judgment is specifically addressed by the three engineering disciplines through their respective departmental standards on calculations, SAS-22, MAS-22, and ESI-253. These standards provide the requirements for documenting engineering judgment and permit the use of engineering judgment under the following conditions.

Engineering judgment may be used when it is evident that the design meets the appropriate criteria by a substantial margin. Engineering judgment may be used in repetitive calculations for similar designs by referencing previously reviewed and approved calculations for the same project. Documentation shall include a discussion of differences between the similar designs. The referenced calculations shall be referenced by calculation number, and revision. Engineering judgment may be used when making revisions to approved calculations if it is evident that the revision does not affect the final calculation, and if it is evident from the previously prepared calculations that design limits are below the allowables. The impact of the revision on the final design is required to be documented.

The reviewer of a calculation may use engineering judgment when making comments if it is evident that his comments do not affect the end result of the calculation. The basis for engineering judgment is required to be documented to permit verification of the logic and adequacy of the judgment.

III. Training

Three Sargent & Lundy QA Procedures address the required training of engineers engaged in the design of nuclear facilities. These are discussed below:

1. Procedure GQ-2.04 describes the training system conducted by the QA Division in the Quality Assurance Procedures. This training covers QA Procedures GQ-3.08 on calculations & GQ-3.17 on Design Information Transmittal, discussed in Section II above.

2. Procedure GQ-2.05 addresses the training to be given by each department in its standards and procedures. A discussion of this training as it affects engineeirng judgment and review of design changes is given below.

3. Procedure GQ-2.07 addresses training in Project Instructions. This training is also given by the Engineering Departments and is discussed below.

Department Training

Training in Department Standards is performed to training procedures in each engineering department.

A. Training in the Documentation of Engineering Judgment

In the Structural Engineering Division (SED), a memo has been issued to all individuals currently performing quality-related activities. This memo contained the more detailed requirements for the use of engineering judgment. Each individual receiving this memo was instructed to document by signature that he has become familiar with and understands the revision to standard SAS-22 addressing engineering judgment. The memos are retained on file to document this training. Revisions to this standard is also discussed in the SED Supervisors Technical Meetings and summarized in the meeting notes which have broad distribution within SED.

The supervisory staff on all nuclear projects have held meetings with project personnel to review the detailed requirements governing the use of engineering judgment. Attendance sheets were signed to document attendance at these meetings.

In the Mechanical Department, a generic training program was established by Mechanical Department Standard MAS-8 issued in February 1984. This standard requires Mechanical Divisions performing quality-related activities (other than those governed by project unique procedures and instructions) to have a documented divisional training program. One aspect of the divisional program is to prepare an outline identifying the standards and procedures governing each individual's work and the need for training in these standards.

Each Mechanical Division performing quality-related work has prepared an outline as required, and MAS-22 (addressing engineering judgment) or its equivalent has been identified on that outline as one of the standards that individuals who prepare, review and approve safety-related calculations must be trained in. The actual training is now in progress and will be ongoing as new personnel are added to the list. The training consists of the Supervisor or his Designee directing the individual to read MAS-22 or its equivalent, and to discuss any questions that he may have with the Supervisor. The Supervisor also observes the work of the individual to determine that he has attained adequated knowledge of the procedure or standard. When this has been accomplished, the Supervisor documents the individual's proficiency and sends this documentation to the Divisional Training Coordinator for record purposes.

In the Electrical Department, documented training has been conducted dealing with the use of engineering judgment when preparing calculations. ESI-253 has been circulated through the Electrical Analytical Division and the Electrical Project Engineering Division to the responsible engineers. All employees responsible for performing Junction Box Calculations have received a revised copy of EDSI-77 which incorporates the concept of engineering judgment. Engineering judgment has also been the subject of intra and interdivisional meetings to emphasize its use.

B. Training in the Review and Control of Design Changes

In addition to the Quality Assurance procedures governing changes to drawings, specifications, calculations, and Engineering Change Notices, the processing of design changes is the subject of a number of project instructions. Personnel performing design verification are trained in applicable standards and procedures. Training in design verification activities is covered by both generic and project unique training programs.

An individual is required to be retrained whenever his responsibilities change or are substantially affected by a revision to applicable standards and procedures. Records are maintained to document the successful completion of all required training. Training in Project Instructions is generally conducted in the project team meetings which are attended by representatives from cognizant divisions and departments. In these meetings, the project instruction is introduced and discussed. This is documented in the meeting notes. The attendees at this meeting are usually the lead personnel from the various divisions. It is the responsibility of these lead personnel to carry the message back to the personnel under their supervision and to instruct those personnel in the requirements of the project instruction.

SARGENT & LUNDY

TABLE 02204-1

LIST OF GENERAL QUALITY ASSURANCE PROCEDURES

SL-TR-1A

Revision 6

Rev.	PROCEDURE NUMBER	PROCEDURE TITLE	10 CFR 50 APPENDIX B REFERENCE CRITERIA			
			PRIMARY CRITERION		SECONDARY CRITERION	
6	[illegible]	[illegible] Organization Manual	I	Organization	II	QA Program
	[illegible]	Project Organization [illegible]	I	Organization	II	QA Program
6	[illegible]	[illegible] Position Descriptions	I	Organization	XVII	QA Records
	[illegible]	Employee Experience Records and Qualification Statements	I	Organization	XVII	QA Records
	[illegible]	S&L Quality Assurance Manual	II	QA Program	I	Organization
	[illegible]	Distribution and Control of the S&L Quality Assurance Manual	II	QA Program	I	Organization
	[illegible]	Review of Client Quality Assurance Program Requirements	II	QA Program	I	Organization
6	[illegible]	Indoctrination and Training in the S&L Quality Assurance Manual	II	QA Program	I	Organization
	[illegible]	Technical Training	II	QA Program	—	—
	[illegible]	Management Review of the S&L Quality Assurance Program and Procedures	II	QA Program	XVIII	Audits
6	[illegible]	Project Indoctrination Training	II	QA Program	V	Instructions, Procedures and Drawings
	[illegible]	Control of Regulations, Codes and Standards	III	Design Control	IV	Procurement Document Control
	[illegible]	Project Scope of Work	III	Design Control	VI	Document Control
	[illegible]	[illegible] Criteria	III	Design Control	II	QA Program
	[illegible]	Design Criteria	III	Design Control	V	Instructions, Procedures and Drawings
	[illegible]	[illegible] Analysis Reports (PSAR's and FSAR's)	III	Design Control	V	Instructions, Procedures and Drawings
					VI	Document Control
	[illegible]	[illegible] Standards	III	Design Control	VI	Document Control
	[illegible]	[illegible]	III	Design Control	V	Instructions, Procedures, and Drawings
					VI	Document Control
	[illegible]	Design Calculations	III	Design Control	VI	Document Control
	[illegible]	[illegible] Design Documents	III	Design Control	VII	Control of Purchased Material, Equipment and Services
	[illegible]	[illegible] Design Reviews	III	Design Control	VI	Document Control
	[illegible]	Engineering Reports	III	Design Control	VI	Document Control
	[illegible]	Project Status Reports	III	Design Control	VI	Document Control
	[illegible]	[illegible]	III	Design Control	VI	Document Control
	[illegible]	Nuclear Feedback	III	Design Control	XVI	Corrective Action
	[illegible]	[illegible]	III	Design Control	VI	Document Control
6	[illegible]	System Functional Descriptions	III	Design Control	VI	Document Control
	[illegible]	Procurement Specifications	IV	Procurement Document Control	III	Design Control
6	[illegible]	Proposal Evaluation [illegible]	IV	Procurement Document Control	VII	Control of Purchased Material, Equipment and Services
	[illegible]	[illegible]	V	Instructions, Procedures, and Drawings	II	QA Program
	[illegible]	Project Distribution Lists and Project File Indexes	VI	Document Control	—	—
	[illegible]	[illegible] Test Procedures	XI	Test Control	V	Instructions, Procedures, and Drawings
	[illegible]	Corrective Action Reports	XVI	Corrective Action	I	Organization
	[illegible]	[illegible]	XVI	Corrective Action	I	Organization
6	[illegible]	Design Errors and Deficiencies	III	Design Control	XVI	Corrective Action
	[illegible]	[illegible] Quality Assurance Records	XVII	Quality Assurance Records	V	Instructions, Procedures, and Drawings
6	[illegible]	Quality Assurance [illegible] System	XVII	Quality Assurance Records	V	Instructions, Procedures, and Drawings
	[illegible]	Internal Audits	XVIII	Audits	I	Organization
	[illegible]	Qualifications of Auditors	XVIII	Audits	I	Organization
	[illegible]	External Audits of [illegible]	XVIII	Audits	IV	Procurement Document Control

ATTACHMENT B

Calculation Sheets Which Document The
Reviews of Auxiliary Building Flooding

9455N

SARGENT & LUNDY ENGINEERS CHICAGO	Calc. for [illegible] Capability After Aux Bldg Flooding	Calc. No. HELB-[illegible]
	X Safety-Related / Non-Safety-Related	Rev. 0 Date [illegible]
		Page 1 of 1[illegible]

Client CECo	Prepared by [illegible]	Date 10-[illegible]
Project Byron / Braidwood 1 & 2	Reviewed by BT Appelton	Date 10-1[illegible]
Proj. No. 4311/4312-00, 4391/4392-00 Equip. No.	Approved by [illegible]	Date 10-1[illegible]

PURPOSE: Confirm that an adequate design approach relative to High Energy Line Break (HELB) and Moderate Energy Line Break (MELB) Auxiliary Building flooding effects and single failure for safe shutdown mechanical and electrical systems has been accomplished and meets the objectives of Standard Review Plan (SRP) Sections 3.4 and 3.6. The scope of this assessment is limited to the safe shutdown systems in the Auxiliary Building since flooding effects in other areas and effects on structures have been assessed in other Sargent & Lundy Calculations. Cross references to these calculations will be made as necessary to verify safe shutdown capability.

METHOD: Reference 1 is a calculation which indicates maximum flood levels for every area within the Auxiliary Building. Based on this calculation, design drawing reviews or field walkdowns were performed (References 2, 3 and 4) which determined all safety related components existing below these flood levels that would be adversely affected by flooding. More specifically, all safety related instrument and electrical components below flood level which could be affected by flooding were identified. Mechanical items

SARGENT & LUNDY ENGINEERS CHICAGO	Calcs. For Reconfirmation of Safe Shutdown Capability After Aux Bldg Flooding	Calc. No. HELB-?
	☒ Safety-Related ☐ Non-Safety-Related	Rev. 0 Date 10-?
		Page 2 of 16

Client CECo	Prepared by M. Breitlaff	Date 10-9-
Project Byron / Braidwood 1 & 2	Reviewed by	Date
Proj. No. 4391/4312-00 4483/4484-00 Equip. No.	Approved by	Date

such as, but not limited to, piping, valve bodies, supports, tanks, heat exchangers and pump casings were not identified since flooding effects would not prevent these components from performing their safe shutdown functions.

The safety related items located below flood level which could be affected by flooding are listed on pages 4-8 of this calculation. The items are listed in groups that correspond to Auxiliary Building flood zones which are defined in Reference 1. The component listing identifies the safe shutdown function of each component and indicates whether that component is required for safe shutdown following any HELB or MELB initiating event in the auxiliary building. Components which would never be required for safe shutdown after one of the events listed above require no further assessment.

Components which may be required for safe shutdown following a HELB or MELB and which are located below flood level require further assessment. Each Auxiliary Building flood zone which contains component(s) described above is assessed individually to ensure that safe shutdown is attainable following the initiating event and single failure. These assessments are on pages 9-14

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CHICAGO

Calcs. For Confirmation of Safe Shutdown Capability After Aux. Bldg. Flooding

[X] Safety-Related | [ ] Non-Safety-Related

Calc. No. HELB-
Rev. 0 Date
Page 3 of 14

Client CECo
Project Byron/Braidwood 1&2
Proj. No. 4391/4392-00 4391/4392-00 Equip. No.

Prepared by M.S. Lentoff Date
Reviewed by Date
Approved by Date

of this calculation. These design justifications demonstrate that safe shutdown is attainable mainly for the following reasons.

(a) Redundant and/or diverse means of achieving the safe shutdown function of the disabled equipment exist.

(b) The safe shutdown equipment disabled by flooding is not required for safe shutdown following the specific HELB or MELB event which causes the flooding.

Based on the information given on pages 4-8 only the Auxiliary Building flood zones listed below contain equipment that may be required for safe shutdown following a HELB or MELB in the A.B. and which is located below flood level. These are the only zones which the assessments on pages 9-14 are required for.

G1-1A G1-1B
S2-8A S2-8B
S2-13A S2-13B
S3-10A S3-10B
S3-13A S3-13B
G4-1

Form GO-3.05.1 Rev. 2

FLOOD ZONE No	AFFECTED EQUIPMENT	SAFETY RELATED EQUIPMENT DESCRIPTION	REQUIRED FOR AUX. BLDG. HELB OR MELB SSD (YES or NO)
G1-1A	1VA01SA	ESW Pump 1A cubicle cooler	Yes
	2VA01SA	ESW Pump 2A cubicle cooler	Yes
G1-1B	1VA01SB	ESW Pump 1B cubicle cooler	Yes
	2VA01SB	ESW Pump 2B cubicle cooler	Yes
S1-3A	1SX001A	ESW Pump 1A inlet isolation valve for pump maintenance	No
S1-3B	2SX001A	ESW Pump 2A inlet isolation valve for pump maintenance	No
S1-3C	1SX001B	ESW Pump 1B inlet isolation valve for pump maintenance	No
S1-3D	2SX001B	ESW Pump 2B inlet isolation valve for pump maintenance	No

Note: Valves 1/2SX001A/B are not required for isolation following a MELB between the valves and pumps since the upstream valves at the ESW cooling tower basin could be utilized.

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Client CEG

Project Braidwood 1&2

Proj. No. [illegible]

Equip No.

Calcs. For [illegible]

Safety-Related ☒ Non-Safety-Related ☐

Prepared by [illegible] Date 10-7-

Reviewed by [illegible] Date

Approved by Date

Calc. No. HELB-2 Date 10-7-

Rev. 0

Form GQ-3.08.1 Rev. 2

SARGENT & LUNDY
ENGINEERS
CHICAGO

Calcs. For [illegible] of Safe Shutdown Capability After Fire Aux Bldg Elev [illegible]

X Safety-Related | Non-Safety-Related

Client CECo
Project Byron / Braidwood 1&2
Proj. No. [illegible]
Equip. No.

Prepared by [illegible] Date [illegible]
Reviewed by [illegible] Date
Approved by Date

Calc. No. [illegible] Rev. 0 Date [illegible] Page 5 of 14

S2-8A	1TS-VA003	Temp switch which supports RH pump 1A cubicle cooler	Yes
	1VA03J	Control panel which supports RH pump 1A cubicle cooler	Yes
	1VA02SA	RH pump 1A cubicle cooler	Yes
	1RH01PA-M	RH pump 1A motor	Yes
S2-8B	2TS-VA003	Temp switch which supports RH pump 2A cubicle cooler	Yes
	2VA03J	Control panel which supports RH pump 2A cubicle cooler	Yes
	2VA02SA	RH pump 2A cubicle cooler	Yes
	2RH01PA-M	RH pump 2A motor	Yes
S2-13A	1TS-VA004	Temp. switch which supports RH pump 1B cubicle cooler	Yes
	1VA04J	Control panel which supports RH pump 1B cubicle cooler	Yes
	1VA02SB	RH pump 1B cubicle cooler	Yes
	1RH01PB-M	RH pump 1B motor	Yes
S2-13B	2TS-VA004	Temp switch which supports RH pump 2B cubicle cooler	Yes
	2VA04J	Control panel which supports RH pump 2B cubicle cooler	Yes
	2VA02SB	RH pump 2B cubicle cooler	Yes
	2RH01PB-M	RH pump 2B motor	Yes

Form GQ 3.08.1 Rev. 2

S2-9A	1VA03SA	CS pump 1A cubicle cooler	No
	1CS010A	CS eductor valve 1A	No
S2-9B	2VA03SA	CS pump 2A cubicle cooler	No
	2CS010A	CS eductor valve 2A	No
S2-12A	1VA03SB	CS pump 1B cubicle cooler	No
	1CS010B	CS eductor valve 1B	No
	1VA06J	CS pump 1B cubicle cooler panel	No
S2-12B	2VA03SB	CS pump 2B cubicle cooler	No
	2CS010B	CS eductor valve 2B	No
	2VA06J	CS pump 2B cubicle cooler panel	No

Note: Containment spray system is only required following a containment LOCA and would not be required following a HELB or MELB in the Auxiliary Building.

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Client CECo
Project Braidwood 1&2
Proj. No. [illegible]
Equip. No.

Calcs. For [illegible]
✓ Safety-Related
Non-Safety-Related

Prepared by [illegible] Date 10-[illegible]
Reviewed by [illegible] Date
Approved by Date

Calc. No. [illegible]
Rev. 0 Date [illegible]

Form GQ-3.08.1 Rev. 2

S3-10A	1TS-VA010	Temp. switch for CV pump 1A cooler	Yes
	1VA10J	Control panel for CV pump 1A cooler	Yes
	1VA06SA	CV pump 1A cubicle cooler	Yes
	1CV01PA-M	CV pump 1A motor	Yes
S3-10B	2TS-VA010	Temp switch for CV pump 2A cooler	Yes
	2VA10J	Control panel for CV pump 2A cooler	Yes
	2VA06SA	CV pump 2A cubicle cooler	Yes
	2CV01PA-M	CV pump 2A motor	Yes
S3-13A	1TS-VA011	Temp switch for CV pump 1B cooler	Yes
	1VA11J	Control panel for CV pump 1B cooler	Yes
	1VA06SB	CV pump 1B cubicle cooler	Yes
	1CV01PB-M	CV pump 1B motor	Yes
S3-13B	2TS-VA011	Temp switch for CV pump 2B cooler	Yes
	2VA11J	Control panel for CV pump 2B cooler	Yes
	2VA06SB	CV pump 2B cubicle cooler	Yes
	2CV01PB-M	CV pump 2B motor	Yes

Note: Some additional misc. items which are listed in Reference 4 are also located in the zones above, however, these items are used to support their respective charging pumps. Most importantly, the worst case flood in any of the 4 CV pump cubicles could result in the loss of the charging pump in ...

SARGENT & LUNDY ENGINEERS CHICAGO

Client CECo

Project Byron/Braidwood 1&2

Proj. No. [illegible]

Equip. No.

Calcs. For [illegible]

Safety-Related ✓ Non-Safety-Related

Prepared by [illegible] Date 10-?-

Reviewed by [illegible] Date

Approved by Date

Form GQ-3.08.1 Rev. 2

S3-9A	1VA05S	Positive displacement charging pump cubicle cooler	No
	1VA09J	PDP cubicle cooler panel	No
S3-9B	2VA05S	Positive displacement charging pump cubicle cooler	No
	2VA09J	PDP cubicle cooler panel	No
S3-8A	1RH610	RH pump 1A mini-flow line valve	No
S3-8B	2RH610	RH pump 2A mini-flow line valve	No
S3-11A	1RH611	RH pump 1B mini-flow line valve	No
S3-11B	2RH611	RH pump 2B mini-flow line valve	No

NOTE: The RH pump mini-flow valves are normally open and are only required to close in a high RHR flow condition following a LOCA. Since flooding events in the A.B. only require normal flow, these valves are not required to operate following a HELB or MELB event inside the A.B.

G4-1	OCC01E	CCW pump 0 switchgear to switch power source for pump 0	Yes
S10-2A	1JB1653A	Junction box for valve 1VQ005C	No

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Calcs For [illegible] of Safe Shutdown [illegible] Elevation

X Safety-Related

Non-Safety-Related

Client CECo

Project Byron / Braidwood 1 & 2

Proj. No. [illegible]

Equip No.

Prepared by [illegible]

Reviewed by [illegible]

Approved by

Calc No. [illegible]

Date [illegible]

Rev 2

Date [illegible]

Page 8 of 1[illegible]

Date

Date

ENGINEERS CHICAGO

Capability Aftr. for. Eld's Flooding		Rev. 0 / Date 10-9
X Safety-Related	Non-Safety-Related	Page 7 of 16

Client CECo	Prepared by M.S. Lustloff / Date 10-7
Project Byron / Braidwood IE2	Reviewed by / Date
Proj. No. 4311/4312-00 / 4 / 4 -00 Equip No.	Approved by / Date

The following discussion explains in more detail the methodology that will be used in justifying safe shutdown equipment disabled by A.B. flooding events.

Only the zones with affected safe shutdown equipment required for HELB/MELB A.B. events will be assessed. All affected equipment within a flooded zone is assumed to be disabled simultaneously. Based on the building configuration, flooding events are not necessarilly limited to one flood zone. In cases where a single HELB or MELB causes flooding in multiple zones containing safe shutdown equipment, all the affected equipment in those multiple zones is assumed to be disabled simultaneously.

Reference 1 uses the worst case HELB/MELB to determine the maximum possible flood level in each area. The worst case break from a flood standpoint is not necessarilly the worst case break from a safe shutdown standpoint. Based on the existing piping configuration, each zone with affected safe shutdown equipment will be assessed to verify that the worst case flood break is conservative from a safe shutdown standpoint.

Form GQ 308.1 Rev. 2

SARGENT & LUNDY ENGINEERS CHICAGO	Calcs. For Reformation of Safe Shutdown Capability After Aux. Bldg. Flooding	Calc No. HELC-?
	Rev. 0	Date 10-?
	☒ Safety-Related ☐ Non-Safety-Related	Page 10 of 16

Client CECo	Prepared by [illegible] Date 10-?
Project Byron 1 Braidwood 1&2	Reviewed by Date
Proj. No. 4311/4312-00 4511/4521-00 Equip. No.	Approved by Date

Flood Zones G1-1A and G1-1B

The worst case flood in either of the above zones is due to an essential service water line break. The resulting flood level could disable the cubicle coolers for pumps 1A and 2A in Zone G1-1A. Similarly, the flood in Zone G1-1B could disable the cubicle coolers for pumps 1B and 2B. The separation between the two zones is a water tight separation and the flood does not affect any other zones. The essential service water system is a dual purpose moderate energy system and therefore, per APCSB 3-1 (Reference 5) a single failure in the ESW system need not be postulated. Based on the above, two redundant pumps will be available for safe shutdown in the unflooded cubicle.

Based on examination of piping composite drawings for Zones G1-1A and G1-1B, no HEL's exist in these areas and the only other MEL's in these areas are fire protection lines. Reference 6 indicates that these fire protection lines do not exceed the moderate energy stress limits and therefore cracks or breaks need not be postulated.

Form GQ 3.08.1 Rev. 2

Flood Zone G4-1

The worst case flood in this zone is due to a non-essential service water line break. The resulting flood level could disable the common component cooling water (CCW) pump to Units 1 and 2 (OCC01P). This pipe break also results in flooding of other zones, however, only the safe shutdown equipment in zones S2-8A, S2-8B, S2-13A and S2-13B could be disabled by the flooding. Flooding in all other safe shutdown equipment areas would not be high enough to disable any safe shutdown equipment.

Following the failure of one CCW pump due to flooding and the single failure of another, at least one CCW pump will remain functional per unit as required for safe shutdown. Concurrent flooding of the zones listed above could disable both RHR trains which are used to bring the plant from a hot standby to a cold shutd condition. The Byron / Braidwood licensing basis is to attain safe shutdown following any accident. For non-LOCA breaks, safe shutdown is defined as hot standby (T_{AVG} greater than or equal to 350°F, zero percent rated thermal power and k_{eff} less than 0.99). Since the licensing basis is hot shutdown it is not necessary to demonstrate capability to reach cold shutdown conditions (reactor coolant temperature less than or equal to 200°F, zero rated thermal power and keff less than or equal to 0.99) using only safety related equipment. However, alternate means for achieving cold shutdown without repair or replacement of equipment are available. These means are described on page 18 of Reference 7.

No other HELB's or MELB's in Zone G4-1 are more limiting than the non-essential service water break discussed above.

Flood Zones S2-8A, S2-8B, S2-13A, S2-13B

The worst case floods in these zones is caused by a sixteen inch RH line break in each of the respective zones. A break in any one of these zones causes concurrent flooding of both RH pump cubicles per Unit. The flooding does not affect safe shutdown equipment in any other flood zones. Based on the discussion given on pages 11 and 12 of this assessment, loss of the redundant RHR trains is acceptable for achieving hot standby conditions.

- No other HELB's or MELB's in these flood zones are more limiting than the breaks discussed above.

Flood Zones S3-10A, S3-10B, S3-13A, S3-13B

The worst case floods in these zones are caused by CV centrifugal charging pump discharge line breaks within these zones. This flooding plus a single failure could cause a loss of CV system charging capability. Following this event, the plant would be maintained in a hot standby condition until charging is restored. Capability to maintain hot standby conditions and to proceed to cold shutdown is available. This capability is described in detail on pages 46 and 47 of Reference 7.

No other HELB's or MELB's in these zones are more limiting than the breaks discussed above.

Client CECo		Prepared by M.J. Gentloff	Date 10-7-
Project Byron/Braidwood 1 & 2		Reviewed by	Date
Proj. No. 4391/4392-00 4691/4654-00	Equip No.	Approved by	Date

CONCLUSION: This assessment verified the existing plant mechanical and electrical systems design approach for HELB and MELB flooding effects in the Auxiliary Building. This assessment verified the design by showing that:

1. Safe shutdown equipment is located outside all flood zones of influence.
2. Safe shutdown equipment is not affected adversely by flooding.
3. Redundant and/or diverse means of achieving safe shutdown exist after certain equipment is damaged by flooding.
4. The safe shutdown equipment disabled by flooding is not required for safe shutdown following the specific HELB or MELB event which causes the flooding.

The worst case flooding events in one instance caused complete loss of RHR system capability and in another instance (after considering single failure) caused complete loss of CV system charging capability. Safe shutdown capability was demonstrated for each of these instances. Based on these results, the existing system designs are adequate to withstand Auxiliary Building floods.

A detailed review of the original calculation was performed.

BT Appelson

SARGENT & LUNDY ENGINEERS CHICAGO	Calcs. For Confirmation of Safe Shutdown Capability After Aux Bldg Flooding	Calc No. HELB-8
	X Safety-Related / Non-Safety-Related	Rev. 0 Date 10-9-
		Page 16 of 16

Client CECo		Prepared by	Date 10-9-
Project Byron / Braidwood 1 & 2		Reviewed by	Date
Proj. No. 4311/4312-00 4683/4684-00	Equip. No.	Approved by	Date

REFERENCES:

1. Sargent & Lundy Calculation 3C8-1281-001 Rev. 2 entitled "Auxiliary Building Flood Level Calculations".

2. Sargent & Lundy Design Information Transmittal No. DIT-BB-CID-0006-A dated 9-25-84.

3. Sargent & Lundy Document No. HDI-02-BB Rev. 1 entitled "Impact of Auxiliary Building Flood Levels on HVAC".

4. Sargent & Lundy Design Information Transmittal No DIT-BY-EPED-0012 dated 10-9-84

5. Branch Technical Position APCSB 3-1 entitled "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment". (Section B.3.b.(3)).

6. Sargent & Lundy Interoffice Memorandum dated March 19, 1984 entitled "Moderate Energy Piping".

7. Sargent & Lundy August 1984 Report entitled "Confirmation of Design Adequacy for Jet Impingement Effects".

Form GQ-3.08.1 Rev. 2

☒ SAFETY-RELATED ☐ NON-SAFETY-RELATED

DIT-BY-EPED-0012-1*

Page 1 of 1

CLIENT CECo

STATION Byron UNIT(S) 1 & 2

PROJECT NO(S). 4391/2

To M. S. Leutloff - 2

J. D. Regan
Responsible individual (Please Print)

EPED
Division

JDRegan
Responsible individual's signature

10-12-84
Issue date

STATUS OF INFORMATION (this information is approved for use. Design information, approved for use, that contains assumptions or is preliminary or requires further verification (review) shall be so identified)

This information is approved for use. No further verification is required.

IDENTIFICATION OF THE SPECIFIC DESIGN INFORMATION TRANSMITTED AND PURPOSE OF ISSUE (list any supporting documents attached to DIT by its title and revision and/or issue date)

Attachment "A" identifies all safety-related electrical equipment which is located below the flood level within the flood zone identified. The flood levels used to determine the affected equipment are those identified in the Auxiliary Building Flood Level Calculation (Calc. No. 3CB-1281-001), Rev. 1, dated 3-1-84. Only safety-related electrical equipment is listed.

*This DIT supersedes DIT-BY-EPED-0012, dated 7-19-84. The revision of the above calculation has been corrected.

BASIS FOR INFORMATION

Calc. no. N/A Rev. and/or date Report no. N/A Rev. and/or date

Other I.O.M. from T. Kepes to F. G. Gogliotti, dated 4-23-84, 5-10-84 and 6-11-84.

DISTRIBUTION

CC: D. L. Leone/W. C. Cleff - 22 (1/1)
K. J. Green - 22 (1/1)

Form GO-3.17.1 Rev. 0 (5-1-84)

SARGENT & LUNDY ENGINEERS

DESIGN INFORMATION TRANSMITTAL

☒ SAFETY-RELATED	☐ NON-SAFETY-RELATED	DIT- BB-CID-0006-A

CLIENT Commonwealth Edison Company

STATION Byron/Braidwood UNIT(S) 1 & 2

PROJECT NO(S). 4391/4392/4683/4684

To J. Gruneman - [illegible]

R. P. Orkfritz — Responsible individual (Please Print)

CID — Division

[signature] — Responsible individual's signature

9/25/84 — Issue date

STATUS OF INFORMATION (this information is approved for use. Design information approved for use that contains assumptions or is preliminary or requires further verification (review) shall be so identified)

This information is approved for use.

IDENTIFICATION OF THE SPECIFIC DESIGN INFORMATION TRANSMITTED AND PURPOSE OF ISS (list any supporting documents attached to DIT by its title and revision and/or issue date)

Refer to DIT-BB-CID-0006.

Revision made to page 5 which corrected instrument panel numbers on M-828-13, zone S3-13B.

BASIS FOR INFORMATION

Calc no. N/A Rev. and/or date ______ Report no. N/A Rev. and/or date ______

Other NONE

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W. C. Claff

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CLIENT Commonwealth Edison Company

STATION Byron/Braidwood UNIT(S) 1&2

PROJECT NO(S). 4391/4392/4683/4684

Page 1 of 12

To J. Grundeman

R. Orkfritz	C&ID	[signature]	8/8/84
Responsible individual (Please Print)	Division	Responsible individual's signature	Issue date

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This is approved for use.

IDENTIFICATION OF THE SPECIFIC DESIGN INFORMATION TRANSMITTED AND PURPOSE OF ISS (list any supporting documents attached to DIT by its title and revision and/or issue date)

C&ID has reviewed the auxiliary building flood levels versus the mechanical instrument location drawings with the results attached.

- There are no problems with safety-related instruments found to auxiliary flooding.

BASIS FOR INFORMATION

Calc. no ______ ______ Rev. and/or date Report no. ______ ______ Rev. and/or date

Other ______

DISTRIBUTION

W. C. Cleff

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INTEROFFICE MEMORANDUM

L9H7

From: D. H. Flens - 31 (X3901)

Date: August 1, 1984

Project No. 4391/2;[illegible]

Dept./Div.: Mechanical/HVAC

Spec. No.

File No.

Page No. 1

Client: CECo

Stn. Byron/Braidwood

Unit 1 & 2

Subject: HVAC Auxiliary Building Flood Level Impact Report

To: K. J. Green (1/1) - 22

cc: D. L. Leone (1/1) - 22
W. C. Cleff (1/0) - 22
J. Grundman (1/0) - 22
G. C. Jones (1/0) - 22
D. C. Soni (1/0) - 20
A. M. Bizarra (1/1) - 20
W. B. Paschal/S. N. Planjery (1/0) - 31
E. G. Hibbard (1/0) - 31

Attached is a copy of HVAC Calculation HDI-02-BB, Revision 1, dated 07-27-84, entitled "Impact of Auxiliary Building Flood Levels on HVAC," which was prepared per your request.

A list of our assumptions used in performing the evaluation is provided on page 2 of this report. A summary of our report is also provided on page 13.

If you have any questions, please contact me.

DHF:alj
Attachment

SL-F402 - 12/82

DISTRIBUTION LIST FOR BOARD NOTIFICATION

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Docket No. 50-454,455

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Rev. 8/27/84

DOCUMENT ISSUE SUMMARY SHEET

Client: CECO
Project: BYRON/BRAIDWOOD
Project No.: 4391/2 & 4683/4-00
Equipment No.: ___

- [x] Nuclear Safety-Related
- [ ] Nuclear Non-Safety-Related
- [ ] Fossil

Doc. No.: HDI-02-B5
Revision: 1
Responsible Division: HVAC
File Index No.: 8.1
Page 1 of 13

Title: IMPACT OF AUX. BLDG. FLOOD LEVELS ON HVAC.

Revision Number	Date	Revised Pages	Preparer	Method of Review	Reviewer/ Date	Approver/ Date
0	7/17/84	ALL	E. G. Hibbard 7/12/84	Detailed Review	[illegible] 7-16-84	[illegible] 7/17/84
1	7/27/84	Technical revisions pgs 2-7 & 13. All pgs updated dwg. revision letters.	E. G. Hibbard 7/27/84	Detailed Review	[illegible] 7-31-84	[illegible]