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

ENCLOSURE 2

30-17034

September 25, 1984

Mr. James G. Keppler
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Byron Generating Station Units 1 and 2
Byproduct Material License No. 12-05650-18
NRC Docket Nos. 50-454 and 50-455

Reference (a): September 13, 1984 letter from T. R. Tramm to J. G. Davis.

Dear Mr. Keppler:

Commonwealth Edison hereby requests amendment of NRC Material License 12-05650-18 to permit wet storage of our californium startup sources in new fuel assemblies at Byron Station. Expedited consideration of this amendment is requested so that these neutron sources may be moved prior to issuance of the Part 50 operating license for Byron 1.

In reference (a), Commonwealth Edison requested amendment of the special nuclear materials license which authorizes onsite storage of the Byron 1 initial core. Amendment of that license is necessary to permit relocation of two new fuel assemblies into two failed fuel cannisters. The californium startup sources will be installed into these fuel assemblies and stored there until the two fuel assemblies are loaded into the Byron 1 core.

After further review it has become apparent that a corresponding amendment to the special nuclear materials license is also needed to permit wet storage of the californium sources in new fuel assemblies. The proposed revision to the authorized use section of License 12-05650-18 is contained in Attachment A to this letter. The basis for such a change is also provided, including a description of the wet storage area, the source transfer procedure, and the measures being taken to assure that radiation exposure to the personnel involved is ALARA. Additional details are provided in reference (a).

Expedited consideration of this amendment is requested so that the source relocation can be completed prior to commencement of the Byron 1 core loading sequence. As you are aware, the Byron 1 operating license is expected to be issued as soon as the ASLB and ASLAB decisions are issued, probably in mid-October. Issuance of the requested byproduct materials license amendment by October 5, 1984 should permit adequate time to complete the source relocation.

According to 10 CFR 170.11, it is not necessary to enclose a fee with this amendment request.

Please address further questions regarding this matter to this office.

Very truly yours,

David H Smith

For T. R. Tramm
Nuclear Licensing Admininstrator

lm

Attachment

9224N

ATTACHMENT A

Proposed Change to Byron Byproduct Materials
License 12-05650-18

Present Wording of License Condition 9.I:

For storage only.

Proposed Wording of License Condition 9.I:

For storage only in either a dry, protected storage area or in a new fuel assembly located in borated water in a failed fuel container.

Basis:

The primary neutron sources are currently stored on the 411' elevation in the Fuel Handling Building (see Drawing #3). They are individually stored inside 1 inch diameter stainless steel pipes for mechanical support and protection. The source material is located in a 1 1/2 inch long section of the 12 foot long rod approximately 33 inches up from the bottom. Surrounding the steel pipes, in the vicinity of the sources, are cylindrical shields made of paraffin and borated concrete. Surrounding the pipes and containers is a concrete block wall approximately 4 feet high (see Drawing #4). This area has been thoroughly surveyed and is appropriately posted and surveilled.

The proposed storage location for each source rod is attached to a Burnable Poison Rod Assembly located in a new fuel assembly (see Drawing #5). The fuel assemblies are stored in borated water shielding inside failed fuel containers (see Drawing #2). The failed fuel containers (see Drawing #1) are cylindrical vessels approximately 13 inches in diameter containing a internal fuel assembly support framework. The failed fuel containers are stored in pipes approximately 15 inches in diameter which make up the failed fuel container storage rack (see FSAR Figure 9.1-4 and Drawing #2). The failed fuel rack is located in the Spent Fuel Pool (see FSAR Figure 9.1.2).

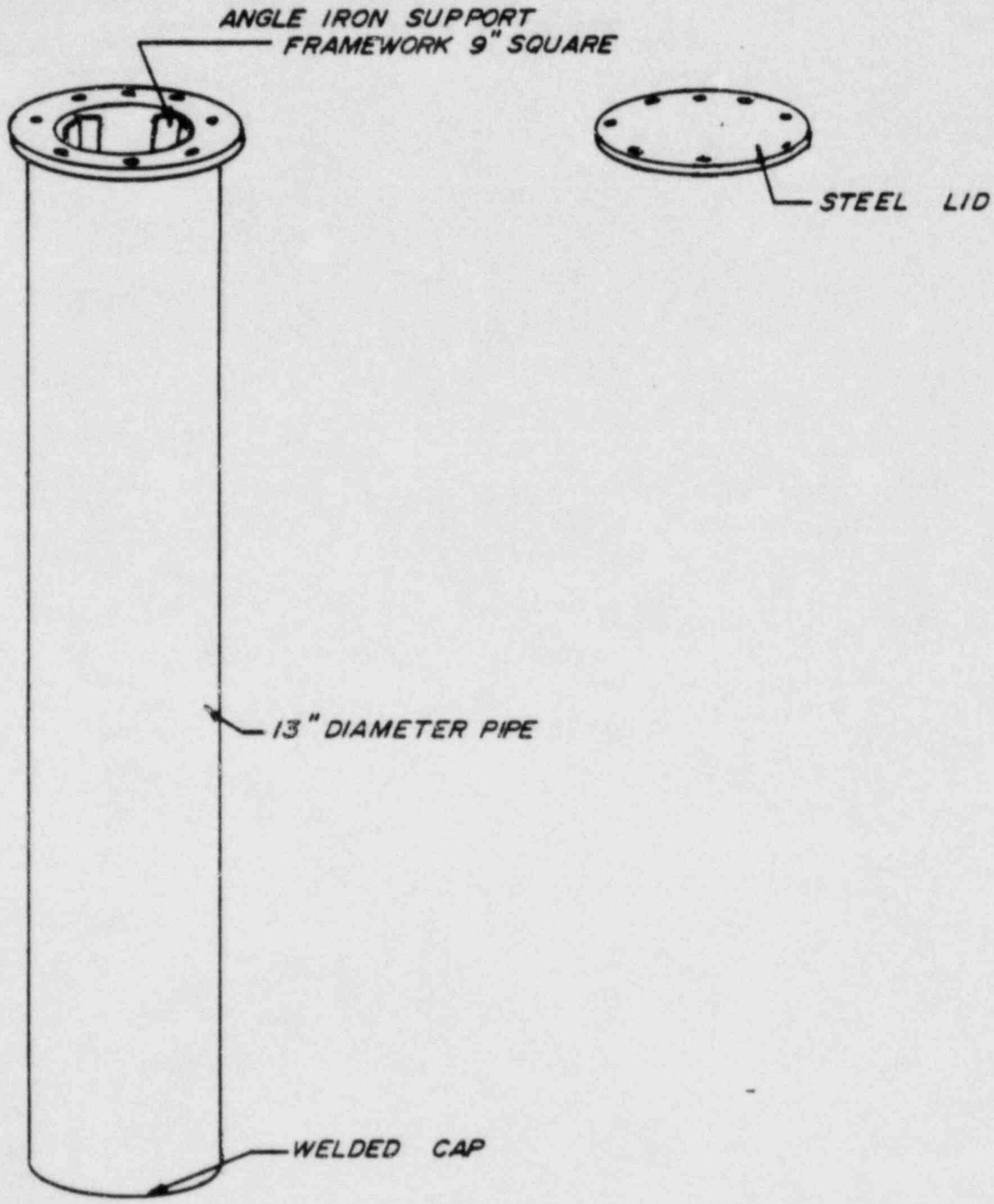
The installation and storage of this source in fuel will not affect the effective multiplication factor of the fuel storage array. There will be an increased neutron population in the immediate source area but this in no way affects criticality.

Transfer of the source rods will be controlled by a written procedure. The source rod will be withdrawn, by hand, from its current storage container into a support assembly and secured. Leak test swipes will be taken during this operation. The source rod and support assembly will be lifted by the fuel handling building overhead crane across the top of the new fuel vault and inserted into the center failed fuel container (see Drawing #3). In this water shielded condition, a person will approach the source and detach the source and support assembly from the crane. This person will then attach the end of the source rod itself to a line extending up to the spent fuel pit bridge crane. After this person withdraws from the area, another person on the bridge crane will manually extract the rod and insert it into the appropriate fuel assembly with the help of a guide funnel. Once inserted in the fuel assembly it will again be water shielded and may be safely approached. The burnable poison rod assembly (BPRA) will then be reinserted into the fuel assembly and the source rod secured to the BPRA with a nut on the threaded source rod end. A lock wire will then be tack welded to the nut to prevent the nut from loosening (see Drawing #5).

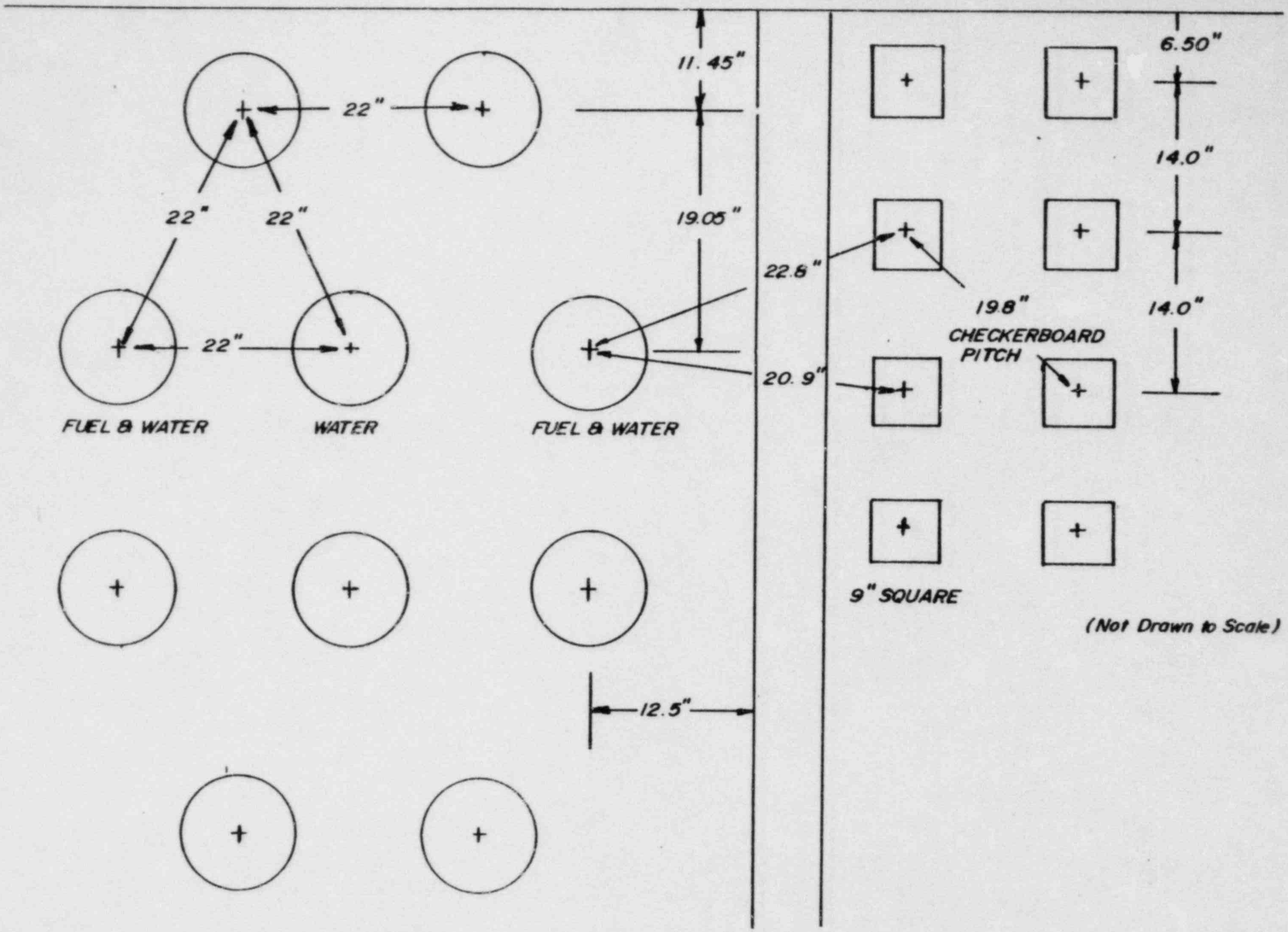
Radiological controls will be in effect before, during, and after the source installation operation to satisfy the requirements of Parts 20 and 50. To keep exposures low during the source installation operation, temporary shielding will be erected in work areas, the borated water shield in the work area will be used to reduce exposure rates by several orders of magnitude, personnel access time will be greatly reduced because of rehearsals performed using mockups, and special tools and the use of cranes will greatly increase source to personnel distances. No person will be within 20 feet of a unshielded source at any time. The total dose commitment for this job is estimated at less than 10 milli-man-rem and a total job time of less than two hours. Area surveys will be conducted after the operation is complete and areas will be posted as necessary. Dose rates to personnel will continually be monitored during the operation and appropriate actions levels have been set.

All work will be performed under supervision of an SRO licensed person with a health physicist present to provide health physics support. Either the health physicist or another radiation chemistry department person present shall be one of the persons named in the byproduct license. The access to the fuel storage areas is restricted and a security guard post is staffed around-the-clock. Access to the source area also will be governed by posted radiological controls.

DRAWING 1

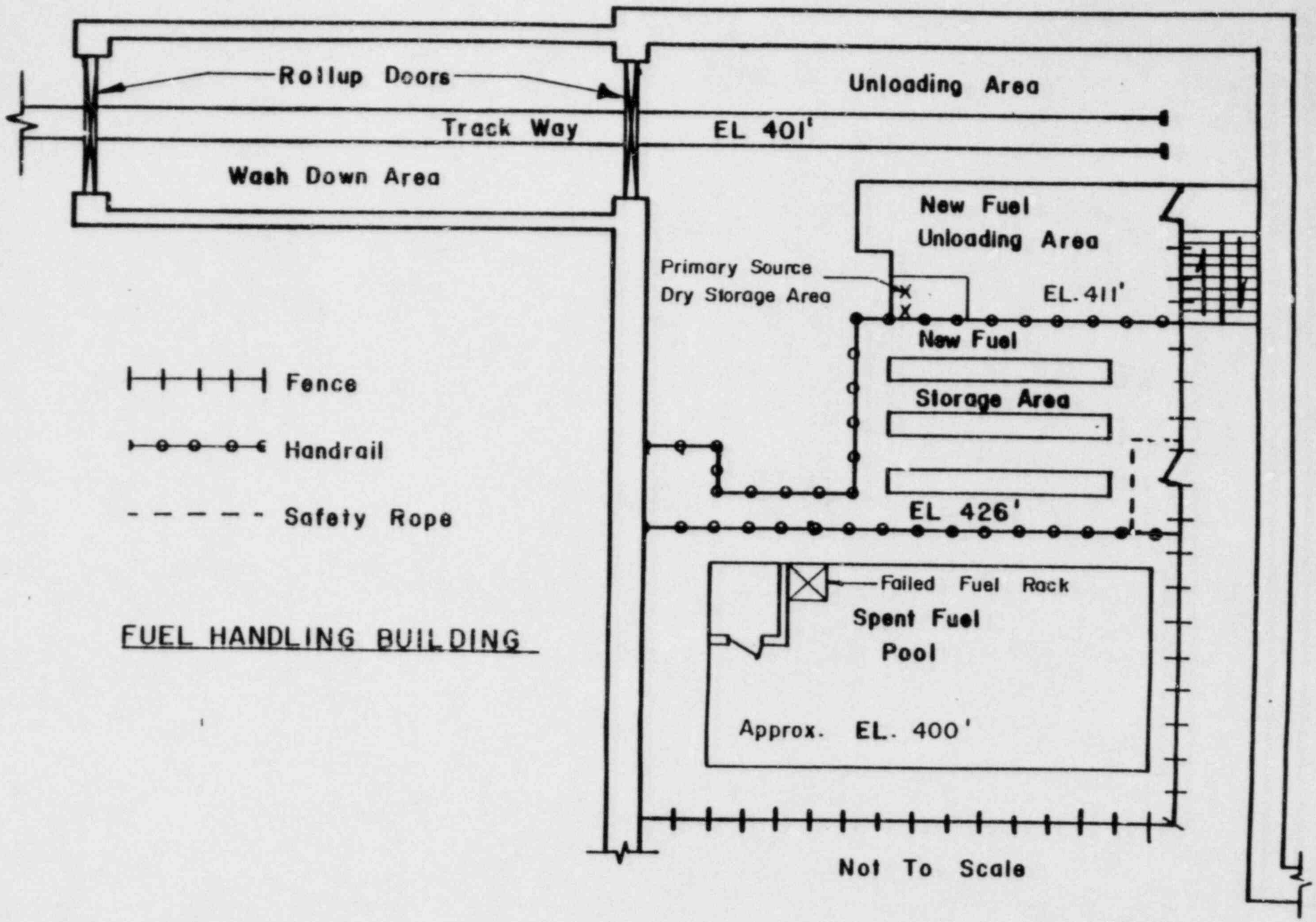


ANGLE IRON SUPPORT
FRAMEWORK 9" SQUARE
STEEL LID
13" DIAMETER PIPE
WELDED CAP



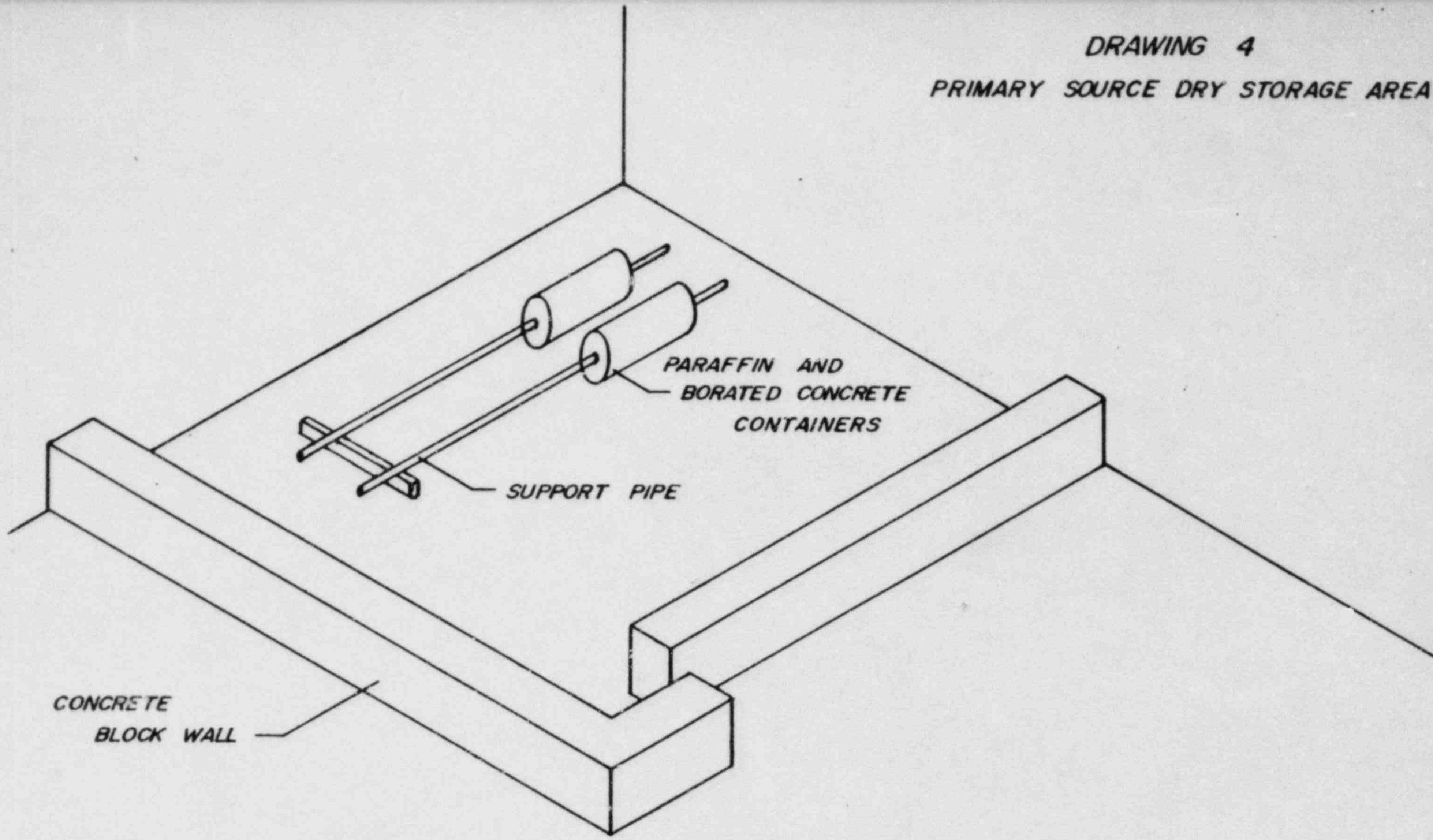
11. 45"
6.50"
22"
14.0"
22"
22"
19.05"
22.8"
19.8"
14.0"
CHECKERBOARD PITCH
22"
20. 9"
FUEL & WATER
WATER
FUEL & WATER
9" SQUARE
(Not Drawn to Scale)
12.5"

DRAWING 2

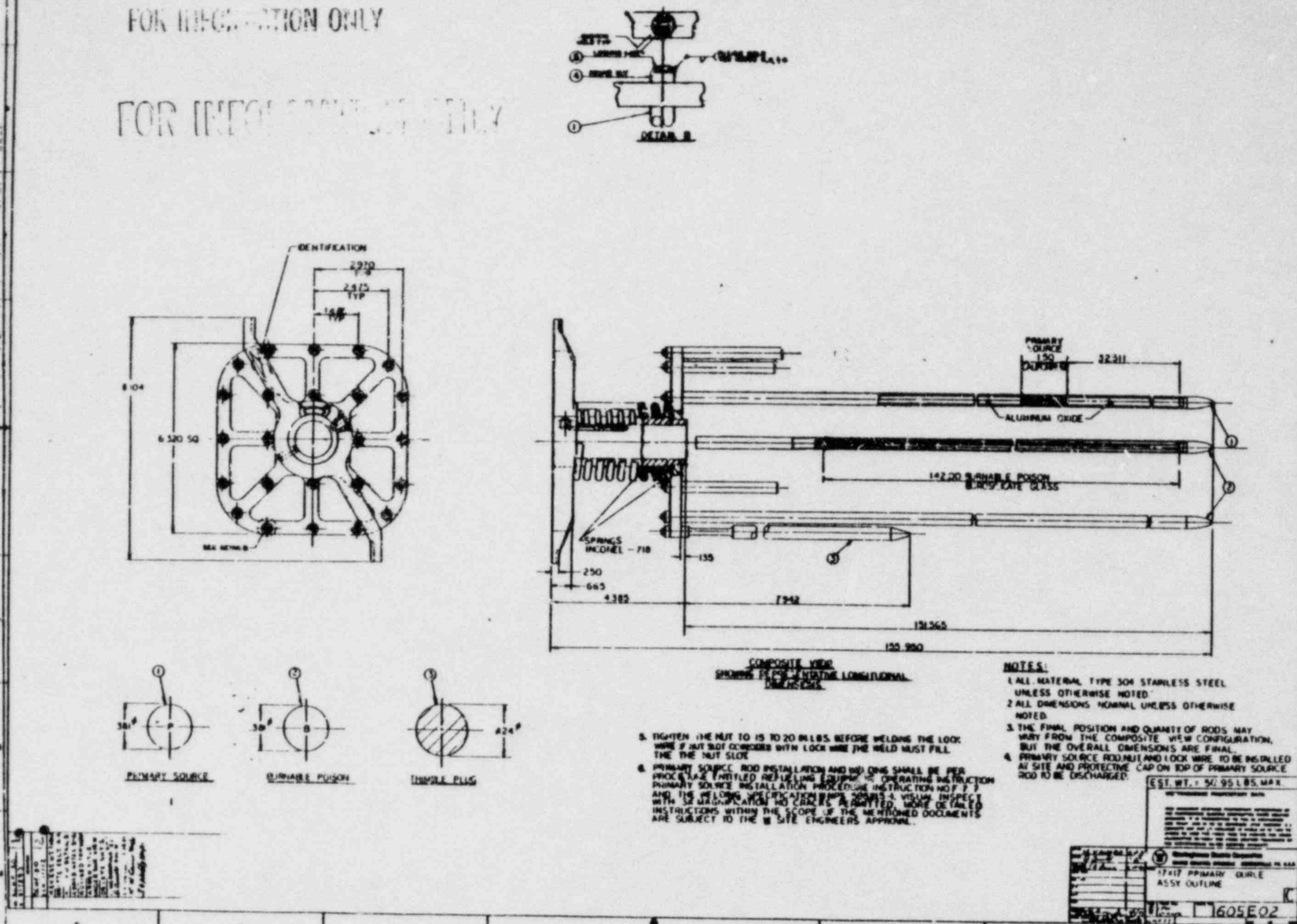

Rollup Doors
Unloading Area
Track Way
EL 401'
Wash Down Area
New Fuel
Unloading Area
Primary Source
Dry Storage Area
EL. 411'
New Fuel
Storage Area
Fence
Handrail
Safety Rope
EL 426'
Failed Fuel Rack
Spent Fuel
Pool
FUEL HANDLING BUILDING
Approx. EL. 400'
Not To Scale

DRAWING 4

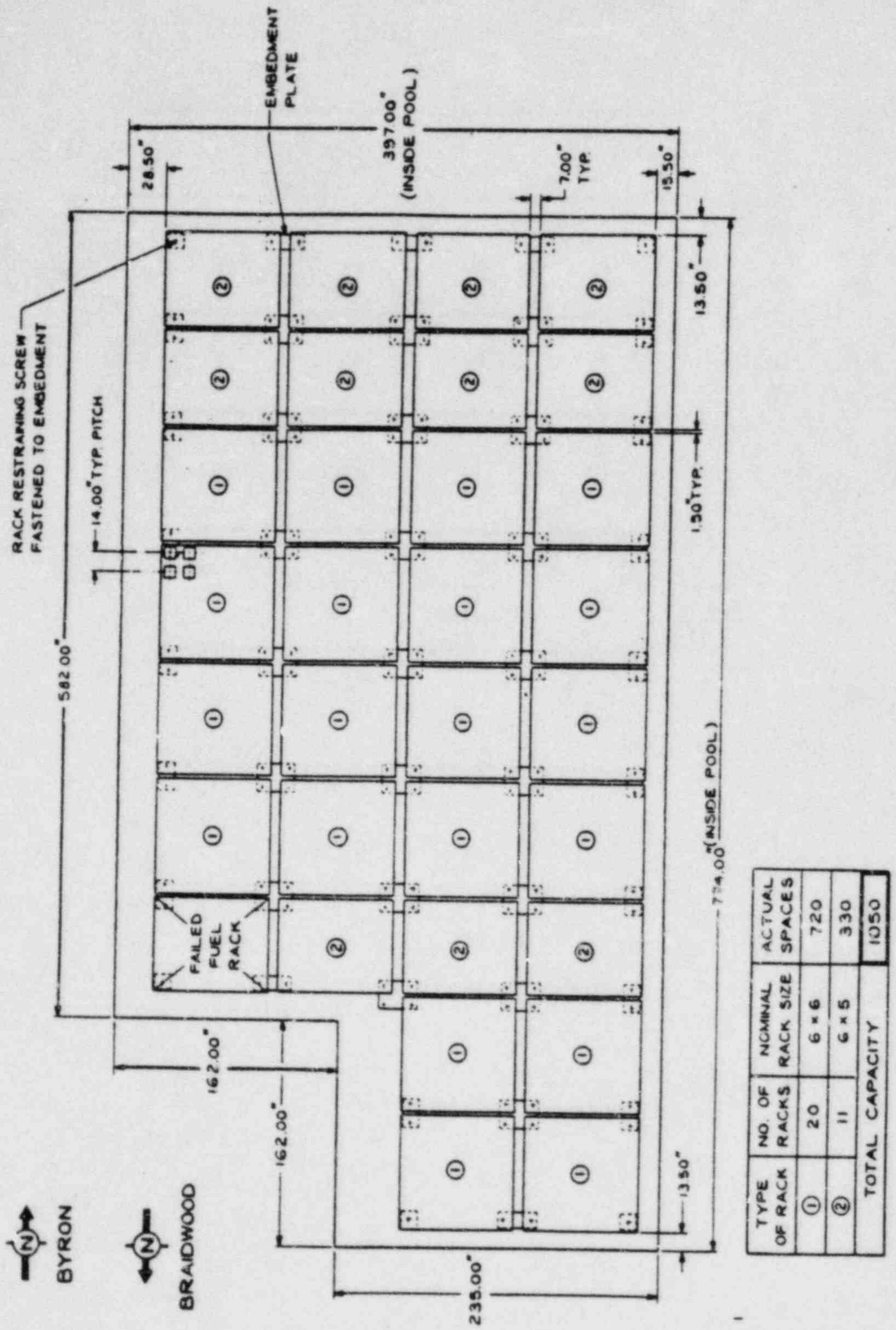
PRIMARY SOURCE DRY STORAGE AREA



DRAWING 5



FOR INFORMATION ONLY
DETAIL B
IDENTIFICATION
2.970 TYP
2.475 TYP
8.04
6.320 SQ
PRIMARY SOURCE
1.50
32.311
ALUMINUM OXIDE
142.20 BURNABLE POISON BOROSILICATE GLASS
SPRINGS INCONEL - 718
.135
.250
.665
4.385
7.942
151.565
155.950
COMPOSITE VIEW
SHOWING REPRESENTATIVE LONGITUDINAL DIMENSIONS
PRIMARY SOURCE
BURNABLE POISON
THIMBLE PLUG
.424
NOTES:
1. ALL MATERIAL TYPE 304 STAINLESS STEEL UNLESS OTHERWISE NOTED.
2. ALL DIMENSIONS NOMINAL UNLESS OTHERWISE NOTED.
3. THE FINAL POSITION AND QUANTITY OF RODS MAY VARY FROM THE COMPOSITE VIEW CONFIGURATION, BUT THE OVERALL DIMENSIONS ARE FINAL.
4. PRIMARY SOURCE ROD, NUT AND LOCK WIRE TO BE INSTALLED AT SITE AND PROTECTIVE CAP ON TOP OF PRIMARY SOURCE ROD TO BE DISCHARGED.
5. TIGHTEN THE NUT TO 15 TO 20 IN.LBS. BEFORE WELDING THE LOCK WIRE. IF NUT SLOT COINCIDES WITH LOCK WIRE THE WELD MUST FILL THE NUT SLOT.
6. PRIMARY SOURCE ROD INSTALLATION AND WELDING SHALL BE PER PROCEDURE ENTITLED REFUELING EQUIPMENT OPERATING INSTRUCTION PRIMARY SOURCE INSTALLATION PROCEDURE INSTRUCTION AND THE WELDING SPECIFICATION. VISUAL INSPECT WITH MAGNIFICATION. NO CRACKS PERMITTED. MORE DETAILED INSTRUCTIONS WITHIN THE SCOPE OF THE MENTIONED DOCUMENTS ARE SUBJECT TO THE SITE ENGINEERS APPROVAL.
EST. WT. = 50.95 LBS. MAX.
17x17 PRIMARY DOUBLE ASSY OUTLINE
1605E02



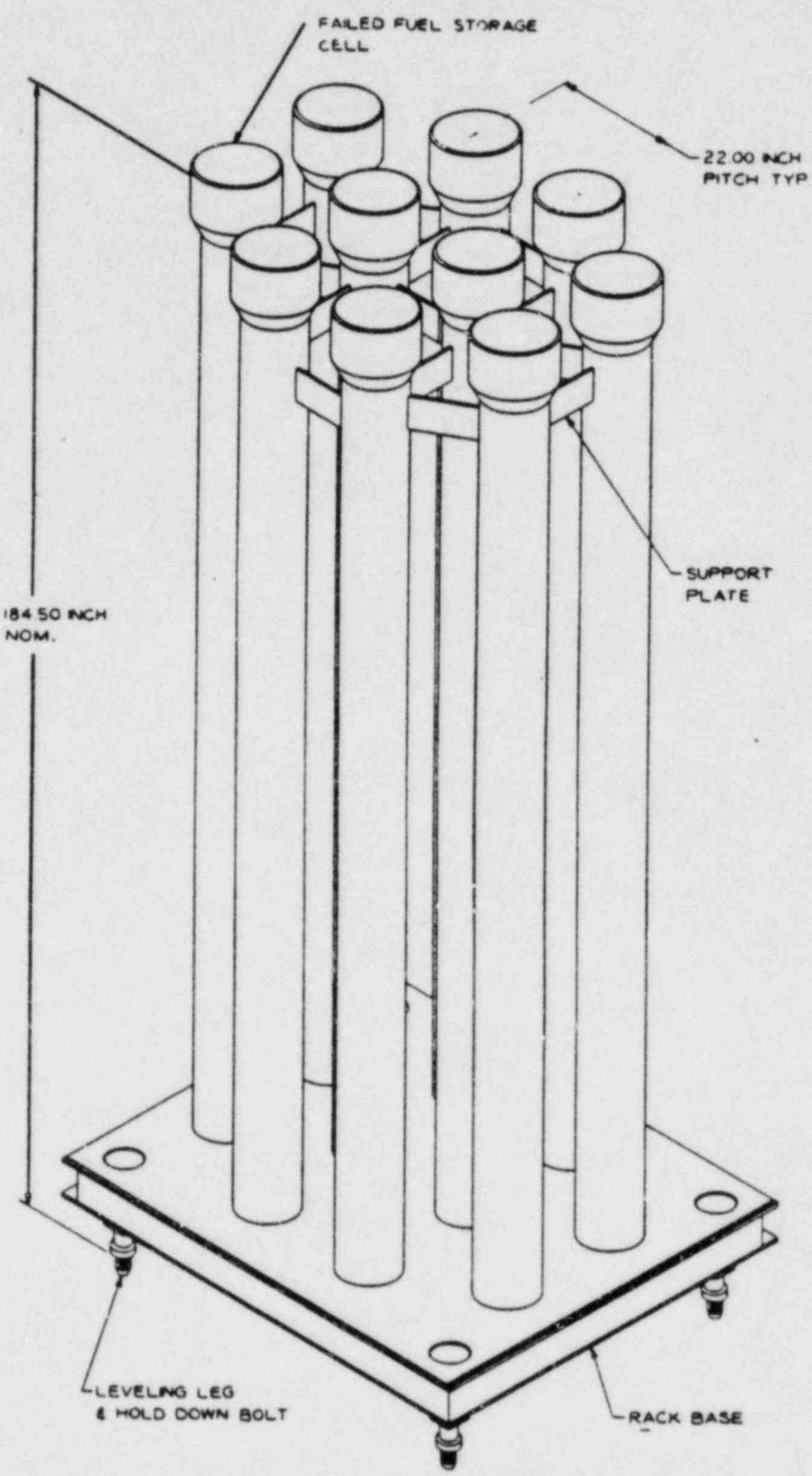
TYPE OF RACK	NO. OF RACKS	NOMINAL RACK SIZE	ACTUAL SPACES
①	20	6 × 6	720
②	11	6 × 5	330
TOTAL CAPACITY			1050

BYRON/BRAIDWOOD STATIONS

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.1-2

SPENT FUEL STORAGE RACK ARRANGEMENT



BYRON/BRAIDWOOD STATIONS

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.1-4

FAILED FUEL CONTAINER RACK

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

ENCLOSURE 3

OCT 01 1984

Commonwealth Edison Company
ATTN: T. R. Tramm
Nuclear Licensing Administrator
P. O. Box 767
Chicago, IL 60690

Gentlemen:

Enclosed is Amendment No. 04 to your NRC License No. 12-05650-18 in accordance with your request.

Please review the enclosed document carefully and be sure that you understand all conditions. You must conduct your program involving radioactive materials in accordance with the conditions of your NRC license, representations made in your license application, and NRC regulations. In particular, note that you must:

1. Operate in accordance with NRC regulations 10 CFR Part 19, "Notices, Instruction and Reports to Workers; Inspection," 10 CFR Part 20, "Standards for Protection Against Radiation," and other applicable regulations.

2. Possess radioactive material only in the quantity and form indicated in your license.

3. Use radioactive material only for the purpose(s) indicated in your license.

4. Notify NRC in writing of any change in mailing address.

5. Request and obtain appropriate amendment if you plan to change ownership of your organization, change locations of radioactive material, or make any other changes in your facility or program which are contrary to your license conditions or representations made in your license application and any supplemental correspondence with NRC. Any amendment request should be accompanied by the appropriate fee specified in 10 CFR Part 170.

6. Submit a complete renewal application with proper fee or termination request at least 30 days before the expiration date on your license. You will receive a reminder notice approximately 90 days before the expiration date. Possession of radioactive material after your license expires is a violation of NRC regulations.

7. Request termination of your license if you plan to permanently discontinue activities involving radioactive material prior to your expiration date.

~~8410310364~~

You will be periodically inspected by NRC. Failure to conduct your program in accordance with NRC regulations, license conditions and representations in your license application will result in enforcement action against you in accordance with the General Policy and Procedures for NRC Enforcement Actions, 10 CFR Part 2, Appendix C.

If you have any questions or require clarification of any of the above stated information, contact us at (312) 790-5625.

Sincerely,

B. J. Holt

Materials Licensing Section

Enclosure: Amendment No. 04

NRC Form 374A
(8-82)

U.S. NUCLEAR REGULATORY COMMISSION

MATERIALS LICENSE
SUPPLEMENTARY SHEET

License number 12-05650-18

Docket or Reference number

Amendment No. 04

Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

In accordance with letter dated September 25, 1984, License Number 12-05650-18 is amended as follows:

Subitem I of Item 9., (Authorized Use) is amended to read:

Authorized Use:

I. For storage only in either a dry, secured storage area or in a new fuel assembly located in borated water inside a failed fuel container.

Condition 16. is amended to read:

16. Except as specifically provided otherwise by this license, the licensee shall possess and use licensed material described in Items 6, 7, and 8 of this license in accordance with statements, representations, and procedures contained in application dated August 30, 1979; and letters dated October 27, 1981, January 4, 1983, January 11, 1983 and September 25, 1984. The Nuclear Regulatory Commission's regulations shall govern the licensee's statements in applications or letters, unless the statements are more restrictive than the regulations.

For the U.S. Nuclear Regulatory Commission

Date OCT 01 1984

By B. J. Holt
Materials Licensing Section, Region III

~~8410310348~~

Paton
~~MWBB 94C~~

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

ENCLOSURE 4

October 2, 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 is to inform you of our receipt of the final Federal Emergency Management Agency's (FEMA) approval of the State of Illinois' Plan for Radiological Accidents (IPRA) as well as associated local plans contained in Volume VI of IPRA related to the Byron Nuclear Generating Station. These plans have been reviewed and approved by FEMA Region V and the FEMA Headquarters. A copy of the letter dated September 12, 1984, from Mr. Samuel W. Speck, Associate Director, State and Local Programs and Support, FEMA to Mr. William J. Dircks, Executive Director for Operations, U. S. Nuclear Regulatory Commission is enclosed for your information.

Although our review of your alert and notification system indicates that it is designed to meet the requirements, and that it is operable, please note that this FEMA approval contains a condition that the adequacy of the public alert and notification system must be verified by FEMA according to the FEMA/Nuclear Regulatory Commission joint criteria as listed in NUREG-0654/FEMA-REP-1 (Revision 1) Appendix 3.

~~8410100424~~

In Supplement 4 of the Safety Evaluation Report (NUREG-0876), the staff has already concluded that subject to the license conditions specified in that supplement, the state of onsite and offsite emergency preparedness related to the Byron Nuclear Generating Station provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. We based that finding regarding offsite preparedness on an interim finding received from FEMA. This final finding issued pursuant to 44 CFR 350 confirms that interim assessment.

If you have any questions regarding this letter, please contact Mr. M. Phillips of my staff on (312) 790-5530.

Sincerely,

L. R. Greger, Chief
Emergency Preparedness and
Radiological Protection Branch

Enclosure: As stated

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

Federal Emergency Management Agency
Washington, D.C. 20472

C2
9/17

SEP 12 1984

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Dircks:

In accordance with the Federal Emergency Management Agency (FEMA) rule, 44 CFR 350, the State of Illinois submitted its State and associated local plans for radiological emergencies related to the Byron Nuclear Power Plant to the Regional Director of FEMA Region V for FEMA review and approval. The Regional Director forwarded his evaluation of the Illinois State and local plans to me on June 18, 1984, in accordance with section 350.11 of the rule. His submission included an evaluation of the full participation exercise conducted on November 15, 1983, and a report of the public meeting held on December 8, 1983, which explained the site-specific aspects of the State and local plans.

Based on an overall evaluation, I find and determine that, subject to the condition stated below, the State and local plans and preparedness for the Byron Nuclear Power Plant are adequate to protect the health and safety of the public in that there is reasonable assurance that the appropriate protective measures can be taken offsite in the event of a radiological emergency. However, while there is a public alert and notification (A&N) system in place and operational, this approval is conditional on FEMA's verification of the A&N system in accordance with the criteria of appendix 3 of NUREG-0654/FEMA-REP-1, Rev. 1.

Sincerely,

Samuel W. Speck

Samuel W. Speck
Associate Director
State and Local Programs
and Support

~~8409280588~~

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

ENCLOSURE 5

OCT 4 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 special safety inspection conducted by Messrs. K. D. Ward, J. W. Muffett and J. M. Jacobson of this office on June 27-28, July 5, July 18-20, August 2-3, August 6-10, and September 7, 1984, of activities at Sargent and Lundy Engineers in Chicago and 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. Tuetken 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.

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

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 and the enclosed inspection report will be placed in the Public Document Room.

8410260187

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

Sincerely,

R. L. Spessard

R. L. Spessard, Director
Division of Reactor Safety

Enclosure: Inspection Report
No. 50-454/84-31(DRS); and
No. 50-455/84-24(DRS)

cc w/encl:
D. L. Farrar, Director
of Nuclear Licensing
V. I. Schlosser, Project Manager
Gunnar 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

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-454/84-31(DRS);
50-455/84-24(DRS)

Docket No. 50-454; 50-455 License No. CPPR-130; CPPR-131

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

Facility Name: Byron Station, Units 1 and 2

Inspection At: Byron Site, Byron, Il
Sargent & Lundy Engineers, Chicago, IL

Inspction Conducted: June 27-28, July 5, July 16-20, August 2-3, 6-10, and September 7, 1984.

Inpsectors: J. W. Muffett

Date

K. D. Ward

Date

J. M. Jacobson

Date

Approved By: D. H. Danielson, Chief
Materials & Processes Section

10/2/84
Date

Inspection Summary

Inspection on June 27-28, July 5, July 16-20, August 2-3, August 6-10, and Septemer 7, 1984 (Report No. 50-454/84-31(DRS); 50-455/84-24(DRS))
Areas Inspected: Special unannounced inspection of previous inspection findings; a 10 CFR 50.55(e) item; visual examination of various welds on cable trays, hangers and control boards; an allegation; and a review of detailed engineering evaluation of weld discrepancies on various components. The inspection involved a total of 88 inspector-hours by three NRC inspectors on site and 80 inspector-hours at the Regional Office.
Results: No items of noncompliance or deviations were identified.

~~8410260194~~

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

G. Sorenson, Construction Superintendent
*R. Tuetken, Startup Coordinator
R. Klinger, QC Supervisor
**T. Tramm, Licensing Administrator

Hatfield Electric Company (HECo)

J. Spangler, Lead Welding Inspector (PTL)
D. McCarty, Quality Control Engineer

Sargent & Lundy Engineers

R. W. Hooks, Assistant Head, Structural Engineer Division
K. T. Kostal, Partner

The inspectors also contacted and interviewed other licensee and contractor employees.

*Denotes those attending the final onsite exit interview on July 5, 1984.

**Denotes those attending the final exit concerning analysis on September 7, 1984.

2. Exit Interview

The inspectors met with applicant representatives denoted in Paragraph 1 at the conclusion of the inspection. The inspectors summarized the scope and findings of the inspection noted in this report.

3. Functional or Program Areas Inspected

The details of this inspection are documented in Sections I and II.

SECTION I

Prepared by: K. D. Ward
J. M. Jacobson

1. Licensee Action on 10 CFR 50.55(e) Item

(Closed) 50.55(e) 82-08 (455/82-08-EE): Inspection records do not exist for a significant quantity of high strength bolted connections in the auxiliary building. Also, establish that records do exist for Unit 1 and 2 Containment building connections. The inspector reviewed the final response dated January 14, 1983 and the statistical sampling plan.

During a review of structural steel bolting inspection records for the auxiliary building, fuel handling building, and the river screenhouse, it was determined that inspection records were not available for some of the high strength bolted connections. Specification requirements dictated testing a minimum of 10%, but not less than 2 of the bolts in each connection. Records are not available for inspection of 55.3% of the high strength bolted connections in the auxiliary building/fuel handling building and 49% of those connections in the river screenhouse. The lack of records was caused by a failure to establish an accountability system to indicate the status of inspection completed on the part of the contractor. Adequate records exist for inspection of the bolted structural connections in the containment buildings.

A statistical sampling plan was established to reinspect the high strength bolted connections. This reinspection was performed by the site independent testing contractor in accordance with an approved reinspection procedure.

Only one of 125 reinspected connections did not meet the inspection criteria. Per the sampling plan, reinspection of additional connections was not required.

The one connection which did not meet the inspection criteria was a ten bolt beam connection. One bolt was satisfactory, seven bolts were torqued to 96% of the required inspection torque and two bolts were in place, but were not torqued. This connection was reviewed against the original design loads and it was found that the connection was adequate to support the loads, in the condition that the connection was found at the time of the inspection.

Based on the results of the statistical sampling plan by CECo, it was concluded that the high strength bolted connections have been properly installed.

2. Visual Examination of Systems Control Corp. Welds

The NRC inspectors visually examined the following hanger welds comparing weld maps made by Sargent & Lundy (S&L) and verifying that all defects were correctly identified. It was found that all defects were identified and that the S&L inspectors were very conservative in making the maps and examining the welds.

Weld #	Hanger #	Traveler #	Drawing #	Item #	Random #	No. of Welds
85	14H7	51408	0-3022	109	570	6
86	14H7	51408	0-3022	109	570	6
87	14H7	51408	0-3022	109	570	6
88	14H7	51408	0-3022	109	570	6
89	14H7	51408	0-3022	109	570	6
6	H036	51377	0-3072	14	2099	1
7	H036	51377	0-3072	14	2099	1
8	H036	51377	0-3072	14	2099	6
9	H036	51377	0-3072	14	2099	6
10	H036	51377	0-3072	14	2099	2
11	H036	51377	0-3072	14	2099	6
17	H077	51450	2-3061	21	4429	2
18	H077	51450	2-3061	21	4429	2
19	H077	51450	2-3061	21	4429	6
20	H077	51450	2-3061	21	4429	6
21	H077	51450	2-3061	21	4429	6
4	H051	51376	1-3061	10	3202	8
5	H051	51376	1-3061	10	3202	8
31	H096	51432	0-3063	43	1794	8
32	H096	51432	0-3063	43	1794	8
33	H096	51432	0-3063	43	1794	5
34	H096	51432	0-3063	43	1794	5
35	H096	51432	0-3063	43	1794	4
36	H096	51432	0-3063	43	1794	4
81	H140	51378	0-3062	104	1646	14
82	H140	51378	0-3062	104	1646	1
83	H140	51378	0-3062	104	1646	4
84	H140	51378	0-3062	104	1646	4

The NRC inspectors also visually examined approximately 100 of the following welds which had minor porosity, undercut, surface irregularities, etc. It was determined that all the welds met the intent of the Code. They were shop welds (Systems Control) and field welds (Hatfield), pans welded to unistrut, channel to unistrut, etc.

Hanger #	Drawing #	Hanger #	Drawing #	Hanger #	Drawing #
H097	0-3063	H087	0-3063	H67	2-3061
H098	0-3063	H073	0-3063	H36	2-3061
H100	0-3063	H149	0-30162	H60	2-3061
H102	0-3063	H142	0-30162	H44	1-3061
H104	0-3063	H159	0-30162	H152	1-3061
H084	0-3063	H148	0-30162	H49	1-3061
H085	0-3063	H66	2-3061		

The NRC inspectors also visually examined the welds securing the main control boards in Unit 2 to the floor and found the welds to be acceptable. The welding was not completed and may be completed in the near future. The NRC inspectors also discussed the mounting of the Systems Control control boards with S&L and Hatfield personnel. S&L's

latest drawing, "Electrical Installation, Electrical Equipment - Mounting Details," Drawing No. 6E-0-3391AL, approved 4/3/84, was also reviewed. Hatfield welders were performing the welding.

The NRC inspector visually examined the inside welds of the following items welded by SCC and found them to be acceptable.

- Containment Isolation Panel #2PM11J
- Main Control Board - Generator and Auxiliary Power #2PM01J
- DC Fuse Panel #2DC10J
- Local Instrument Rack #2PL66J
- Local Instrument Rack #2PL75J
- Local Instrument Rack #2PL76J

The NRC inspector reviewed S&L Specification F-2815 "Cable Pans and Hangers", and selected various hanger and cable pan fitting details for inspection of weld quality. Approximately 300 welds were inspected, including welds in the following reas: elev. 439 (location 18-26 at L-Q), elev. 426 (cable spreading rooms), elev. 426 (location 12-16 at Q-V and 19-25 at Q-V) and elev. 414 and 426 (location R 18, inside containment). Weld quality in general appeared acceptable.

No items of noncompliance or deviations were identified.

3. <u>Cable Tray Hanger Connections and 90° Cable Tray Fittings</u>

The NRC inspector reviewed CECo's procedure, "Inspection of Cable Tray Hanger Connections and 90° Cable Tray Fittings". Hatfield visual welding inspection procedures, and training procedure, and several weld inspector qualifications were reviewed and found to be acceptable.

Systems Control Company (SCC) provided cable tray hanger assemblies at Byron. Hatfield installed the components supplied to the site by SCC. In order to address the general concern for weld quality covered in NCRs 850 and 895, a random sample of 80 hangers from the population of 5,717 Systems Control hangers at Byron was identified by Sargent and Lundy for weld inspection. The sample was selected from the population of hangers using a list of random numbers. This selection process ensured that the sample was unbiased and representative of all hangers in the plant. The sample captured all commonly used connection types, including 44 connections that, based on the original design, were deemed to be highly stressed.

The inspections of the selected hangers were performed by Hatfield with verification through field inspections by CECo's third party inspectors (Sargent & Lundy Level III inspectors on loan to Commonwealth Edison). The 80 hangers included 358 Systems Control shop-welded connections. Of the 358 connections inspected from the sample 80 hangers, 252 connections had no discrepancies, and 106 were found to have some form of discrepancies such as underlength, undersize, overlap, undercut, craters, and two connections with missing portions of welds. None of the welds had cracks.

Inspections of cable tray fittings were performed in 1977 pursuant to Commonwealth Edison's Byron NCR 105. NCR 105 was issued in response to the fact that Systems Control did not have approved welder qualifications and procedures. As part of the overall response to the nonconformance, 99 fittings out of approximately 1,200 which were at the Byron site at that time, were inspected by Industrial Contract Services for the purpose of determining SCC weld quality. Both stiffener welds and side channel welds were inspected with no discrepancies found in the stiffener welds. Four fittings were found to have side channel weld discrepancies. These discrepancies included lack of fusion, porosity, and a missing weld attaching a corner bent plate to the cable tray side channel. None of these discrepancies had design significance.

In June 1984 Sargent & Lundy performed an engineering evaluation in order to confirm that the fitting welds are not required to meet structural load-carrying requirements due to the presence of alternate load paths able to carry the cable loading. The evaluation confirmed that the fitting welds are not required to enable fittings to meet load requirements due to the existence of redundant load paths.

However, the evaluation determined that in one configuration, involving the outside fitting weld of a 90 degree fitting, only one load-bearing redundancy exists, the fitting stiffener. The fitting weld therefore is required if the stiffener weld in that corner of the fitting is missing. The condition of a missing stiffener weld at the outside corner of a 90 degree fitting has not been found in any inspection. In order to assure that this condition does not exist, all 90 degree fittings will be inspected and repaired as required.

Approximately 962 90°tray fittings and approximately 3,000 hanger connections were visually examined by CECo's Level IIs, contracted by Daniels. The unacceptable welds found by the Level IIs were reinspected by an S&L Level III who was involved in the reinspection program.

The NRC inspector observed the reinspection of the following Systems Control welds and basically agreed with the interpretation.

<u>90° Tray Fittings Welds</u>	<u>Drawing Number</u>
11516M P2E	6E-1-3061 Rev. V
11516L P2E	6E-1-3061 Rev. V
11491T P2B	6E-1-3061 Rev. V
11610J C2E	6E-1-3061 Rev. V
11612J K2B	6E-1-3061 Rev. V
11647J C2E	6E-1-3061 Rev. V
11659S K2B	6E-1-3061 Rev. V
11588F P1B	6E-1-3061 Rev. V
11588E P1B	6E-1-3061 Rev. V
11683S K2B	6E-1-3061 Rev. V
21693F P1B	6E-1-3061 Rev. V
21693E P1B	6E-1-3061 Rev. V
2P2B (EL. 421'4")	6E-0-3032 Rev. T

2C2B (EL. 420')	6E-0-3032 Rev. T
2P1B (EL. 411'10")	6E-0-3033 Rev. Y
1K1B (EL. 418'11")	6E-0-3031 Rev. AA
1P1E (EL. 420'3")	6E-0-3031 Rev. AA
1C1E (EL. 418'11")	6E-0-3031 Rev. AA

Cable Tray	Connection Welds	Drawing Number
H005/DV8	2 welds	6E-0-3062H
H006/DV8	2 welds	6E-0-3062H
H007/DV8	2 welds	6E-0-3062H
H008/DV8	2 welds	6E-0-3062H
H009/DV8	2 welds	6E-0-3062H
H011/DV8	2 welds	6E-0-3062H
H017/DV8	2 welds	6E-0-3062H
H019/DV8	2 welds	6E-0-3062H
H021/DV8	2 welds	6E-0-3062H
H024/DV8	2 welds	6E-0-3062H
H041/DV8	2 welds	6E-0-3062H
H109/DV8	2 welds	6E-0-3062H
H064/DV8	4 welds	6E-0-3062H
H044/DV8	4 welds	6E-0-3062H
H045/DV8	2 welds	6E-0-3062H
H046/DV8	2 welds	6E-0-3062H
H043/DV8	2 welds	6E-0-3062H
H051/DV8	2 welds	6E-0-3062H
12H5/DV8	2 welds	6E-0-3031 Rev. 8A

No items of noncompliance or deviations were identified.

4. Cable Tray Hanger Connection - Walkdown Training

The NRC inspector reviewed CECo's "Instruction for Walkdown Cable Tray Hanger Connection Welds" and attended the class for the training in accordance with the instruction.

Approximately 100 walkdown personnel (S&L Designers and Engineers) and 7 certified AWS weld inspectors (Daniels personnel) received formal classroom training and practical test using actual mockups which the NRC inspector observed. The practical test consisted of 25 weld details with acceptable welds and welds missing. Records of this training and testing for walkdown personnel are maintained by the S&L overall field coordinator. Records of this training and testing for weld inspectors are maintained by the CECo QC Supervisor.

All accessible Systems Control shop cable tray hanger connections in safety related areas as issued by Sargent & Lundy and directed by CECo were walked down. Any walkdown findings or missing welds were inspected or mapped by certified AWS weld inspectors.

Fireproofing or blockwalls were not a cause for classifying DV-8 or DV-8A connections inaccessible. Where this condition existed, the fireproofing or blockwall section was removed to establish accessability after review of the condition by CECo.

12H4/DV8	2 welds	6E-0-3031 Rev. 8A
17H1/DV8	2 welds	6E-0-3031 Rev. 8A
12H2/DV8	2 welds	6E-0-3031 Rev. 8A
12H2/DV8	2 welds	6E-0-3031 Rev. 8A
13H20/DV8	2 welds	6E-0-3032H Rev. T
13H15/DV8	2 welds	6E-1-3032H Rev. T

"T" Fitting

1852N P13 (EL. 411'7")	6E-1-3042 Rev. S

No items of noncompliance or deviations were identified.

5. Allegation

Excessive heat input and violation of maximum interpass temperature for automatic welding of 30" primary coolant piping causing ferrite depletion.

NRC Findings

Ferrite is the magnetic phase found in many grades of otherwise non-magnetic austenitic stainless steel weld metals. Ferrite is desirable in weld metal to the extent that it helps prevent cracking and micro fissuring. The cracking of concern here is generally longitudinal centerline cracking or crater cracking, both of which occur during the final stages of solidification. Regarding fissuring, the consensus of experts is that it occurs in welds during the reheating process when an additional bead is deposited next to or over an existing bead. Except in very severe cases, the great bulk of fissures are microscopic in size. In a very notch tough material such as austenitic stainless steel, it would require very unusual service conditions to adversely affect the service life of the structure. From a practical viewpoint, millions of pounds of multipass fully austenitic weld metal have been used in production weldments with virtually no failures attributable to fissures (The Welding Journal, July 1974). It is generally recognized that a weld metal ferrite content of as little as 3FN is sufficient to prevent cracking or fissuring. Weld metal ferrite content is determined primarily by three factors in descending order of importance: weld electrode chemistry, nitrogen pick up during welding and heat input or cooling rate. The ASME B&PV Code, Section III, Subsection NB, requires that welding electrode and filler metal be capable of depositing weld metal with a minimum ferrite of 5FN. The alleger contends that the heat input of the welds was too high and that the welds do not contain adequate ferrite.

Beginning with the welding electrode chemistry, the inspector reviewed 23 Certified Material Test Reports and found all to meet or exceed ASME Code requirements, 7 out of the 23 were for use with automatic welders. These 7 CMTRs represented the automatic welding of approximately 65 welds.

The inspector then reviewed Hunter Corp.'s (the welding organization) Quality Control Surveillance Reports dating 1/74 through 7/80. Ferrite determinations were made with a Severn gage on most of the welds. Thirty welds were picked at random, and were reviewed for ferrite determinations. All welds were reported to have adequate ferrite content.

Eleven welds in the plant were selected by the inspector to physically measure ferrite range with a Severn gage. Of these 11 welds, 6 were chosen to verify the Quality Control Surveillance effort. All welds were found to contain adequate ferrite and the results agreed with those reported by the surveillance documentation.

This allegation could not be substantiated and is considered closed.

SECTION II

Prepared By: J. W. Muffett

1. Review of Engineering Analysis of Various System Control Corporation (SCC) Supplied Equipment and Components

Certain SCC supplied equipment was identified as having discrepant welds per AWS D1.1. The details concerning the history of these problems are contained in Inspection Report 50-454/84-32(DRP).

The equipment addressed by the detailed engineering analysis are:

- Main Control Boards
- DC Fuse Panels
- Local Instrument Racks
- Solid Bottom Cable Trays
- Solid Bottom Cable Tray Fittings
- Ladder Trays and Fittings
- Cable Tray Hangers

These analyses address either specific discrepancies identified in inspections or whether types of welds which were found to be discrepant were required for structural adequacy.

a. Main Control Boards-Open Item 454/84-32-01; 455/84-25-01 (Closed)

Westinghouse reports WCAP-10390, "Service Qualification of the Byron/Braidwood Main control Board", and WCAP-10412, "Seismic Qualification of the Byron/Braidwood Main Control Room Control Panels and Remote Shutdown Panels", were reviewed. These reports demonstrate the structural adequacy of these components in their "as-built" condition. This closes open item 454/84-32-01; 455/84-25-01.

b. DC Fuse Panels (1DC10J, 1DC11J, 2DC10J, 2DC11J)

The Sargent & Lundy document "Seismic Qualification of DC Fuse Panels" was reviewed along with the weld maps of the DC fuse panels. Also, the Wyle seismic test report of DC fuse panel 1DC10J was reviewed. During the course of the Sargent & Lundy inspections it was discovered that panel 2DC10J was discrepant enough so that the results of the test of panel 1DC10J did not apply. Therefore a detailed engineering analysis of panel 2DC10J was performed. This analysis was also reviewed. All stresses in the members and in the welds are within Code allowables. The highest stress in a weld is only 38% of the Code allowable. These analyses demonstrate that all the DC fuse panels are adequate to perform their design functions.

No items of noncompliance or deviations were identified.

c. Local Instrument Racks

A number of Sargent and Lundy documents and analyses concerning the local instrument racks have been reviewed: "Evaluation of 17 Local Instrument Panels Inspected by Sargent and Lundy", "Determination of Total Weld Length, Area, and Discrepancies for SCC Panels 1PL54J, 1PL71J, 1PL78JA, and 1PL60JA", "Seismic Qualification of Local Instrument Panels", and Wyle Laboratories "Seismic Qualification Test Report of a Local Instrument Rack."

These analyses use two methods to demonstrate the adequacy of these panels. The first is comparison of the panels with a panel which was subjected to a qualification test (the Wyle lab test). The second is a detailed engineering evaluation. Both of these methods demonstrate the adequacy of the panels. The most highly stressed weld was stressed to 10% of the Code allowable.

No items of noncompliance or deviations were identified.

d. <u>Solid Bottom Cable Trays-Open Item 454/84-32-05; 455/84-25-05 (Closed)</u>

The Sargent & Lundy calculation 98.20.1-3, "Effect of Missing Stiffeners on Cable Tray Design" was reviewed. This calculation demonstrates that the stiffener is not required for the cable trays to perform their design function. This effectively addresses the question of the effect of missing or discrepant welds on the cable tray stiffeners. Therefore the structural adequacy of the solid bottom cable trays has been shown. This closes open item 454/84-32-05; 455/84-25-05.

e. <u>Solid Bottom Cable Tray Fittings-Open Item 454/84-32-06; 455/84-25-06 (Closed)</u>

The Sargent & Lundy calculation "Cable Tray Fittings" (12.2.139) was reviewed. This analysis of cable tray tees, crosses, and elbows shows that with one qualification, fitting welds are not required to carry design loads. The qualification pertains to 90° fittings. On the outside of those fittings only two load paths exist; the fitting weld and the fitting stiffener weld. Therefore, if either weld is missing or otherwise incapable of carrying the requisite load (i.e. cracked) the other weld must be capable of carrying the design load. To provide assurance that there is no 90° fitting with two inoperative load paths, all 90° fittings have been inspected for missing fitting welds. No fittings were discovered which were incapable of carrying their design loads. This closes open item 454/84-83-06; 455/84-25-06.

f. <u>Ladder Trays and Ladder Tray Fittings-Open Item 454/84-32-07; 455/84-25-07 (Closed)</u>

The Sargent & Lundy calculation "Ladder Type Cable Tray Weldment Evaluation" was reviewed and found acceptable. Two conclusions are drawn by this analysis. They are: (1) the worst strength reduction found in the sample of straight ladder trays could be applied to any connection on the straight ladder trays and these components could still carry their design loads; (2) the worst strength reduction found in the sample of ladder fittings could be applied to any connection or any ladder fitting and these components could still carry their design loads. Therefore this analysis demonstates the structural adequacy of the ladder trays and the ladder tray fittings. This closes open item 454/84-32-07; 455/84-25-07.

g. <u>Cable Tray Hangers-Noncompliance 454/84-32-08; 455/84-25-08 (Open)</u>

In a number of cases deficiencies were identified in the welds associated with the cable tray hangers. These deficiencies lead to a series of inspection programs dealing with this issue.

(1) CECo and Sargent & Lundy initially inspected and evaluated approximately 300 welded connections. None of these connections exceeded applicable Code allowables for stress. Nevertheless some large strength reductions were apparent in this sample (53% strength reduction). The deficiencies causing these large strength reductions were of a nature that they could not be tolerated by all connections. Therefore, a second inspection program was started, based on the largest strength reduction found in the initial sample (53% strength reduction).

(2) The second program inspected and evaluated all connections which could not tolerate a 53% strength reduction. During this inspection a connection was found which had a significantly larger strength reduction (92% strength reduction). This was evidence that the 53% strength reduction was not the worst case. This lead to a much more comprehensive inspection program.

(3) In the third inspection program all connection types DV-8 and DV-8A were inspected for missing welds and all other accessible connections were inspected for missing welds. Under the provisions of this program, if a connection type was found to have a strength reduction greater than 53% then all of that connection type would be inspected for missing welds. At this time approximately 30,000 connections have been inspected. Approximately 550 connections classified as inaccessible now require inspection and remain to be completed. This noncompliance remains open (Reference 454/84-32-08; 455/84-25-08).

These inspection programs have been reviewed in all stages by the inspectors. These reviews included review of weld maps, weld evaluations, program plans, personnel testing, training and actual observation of welds. No noncompliances or deviations from commitments have been identified in these cable tray hanger inspection programs.

h. Observations

A number of observations were made during the review of these analyses. They are as follows:

1. Ladder Tray Fittings - In some configurations the pipe rung of a ladder tee or cross intersects the sidechannel at an angle of 45°. The original analysis for determining the strength of this connection did not take into account the reduction in effective throat at the 45° intersection. Sargent & Lundy was notified of this problem and performed a reanalysis which has been reviewed and found acceptable. Therefore, this observation has no effect on the conclusions drawn relative to ladder tray fittings.

2. Solid Bottom Cable Trays - In the original calculation "Effect of Missing Stiffeners on Cable Tray Design" the methodology of combining seismic response did not adhere to the methodology to which the Byron Plant is committed in the FSAR. Sargent & Lundy was notified of the problem and performed a reanalysis using the proper combination methodology. The reanalysis has been reviewed and found acceptable. Therefore, this observation has no effect on the conclusions drawn relative to solid bottom cable trays.

i. Conclusion

No items of noncompliance or deviations were identified. The inspection of the final analyses revealed no violation of FSAR commitments as they pertain to design and analysis. Also, the procedures dealing with the performance of these analyses were functioning properly. Therefore, the structural adequacy of the SCC supplied components covered in this report has been demonstrated.

Commonwealth Edison
Byron Nuclear Station
4450 North German Church Road
Byron, Illinois 61010

ENCLOSURE 6

October 10, 1984

LTR: PM-84-71

Mr. D. J. McDonald
Director of Inspections
National Board of Boiler and
Pressure Vessel Inspectors
1044 Crupper Avenue
Columbus, Ohio 43220

SUBJECT: National Board Audit of the Byron Nuclear Station Units 1 & 2

REFERENCE: (i) Commonwealth Edison (V. I. Schlosser) Letter Dated September 10, 1984, to National Board of Boiler and Pressure Vessel Inspectors (D. J. McDonald)

(ii) National Board of Boiler and Pressure Vessel Inspectors (D. J. McDonald) Letter Dated September 21, 1984, to Commonwealth Edison (C. Reed)

Dear Mr. McDonald:

Reference (ii) above summarized the results of the National Boiler Board Audit Team's activities at the Byron Nuclear Station during September, 1984. This letter provides the status of corrective actions to findings and observations in preparation for the National Board's site visit for closeout of open items during the week of October 29, 1984.

3.0 Hunter Corporation

Article 3.1

Approved revisions to Sections 2 and 4 of Hunter Corporation's Quality Assurance Manual and Site Implementation Procedure (S.I.P.) 4.000 were included in Reference (i) and are currently being implemented on site.

Article 3.2

The approved revision to Hunter Corporation's S.I.P. 6.501 was included in Reference (i) and is currently being implemented.

Article 3.3

A copy of the approved revision to the N.D.E. interface agreement was included in Reference (i). This agreement has been implemented by Hunter Corporation.

Article 3.4

Item 1)

The jurisdictional authority (State of Illinois, Department of Nuclear Safety) was notified of the adoption of Code Case N-302 by the attached letter from D. Elias (Project Engineering) to J. Blackburn (Illinois Department of Nuclear Safety) dated September 10, 1984. The jurisdictional authority normally does not respond formally to this type of notification; however, an attempt will be made to obtain a response.

Item 2)

Field Change Requests F-33,684 through F-33,686 were written to revise Hunter Corp., NISCo, and Powers-Azco-Pope installation specifications to allow the use of Code Case N-302.

Item 3)

The Byron Final Safety Analysis Report (F.S.A.R.), as reviewed and approved by the Nuclear Regulatory Commission, provides for the use of code cases included in Regulatory Guide 1.84. A special notification of the N.R.C. is included in the F.S.A.R. only when code cases not approved by or beyond the limitations of Reg. Guide 1.34 are intended to be used.

Item 4)

The appropriate Hunter, NISCo, and Powers-Azco-Pope N-5 Data Reports have been revised to indicate the use of Code Case N-302.

Articles 3.5 through 9.0 have been closed by the National Board Audit Team.

Summary

The information requested by the National Board Audit Team has been sent per Reference (i). Co:rective action for open items in Articles 3.1 through 3.4 ı.as bːen implemented by Hunter Corp. We believe these open items can be closed during the National Board Audit Team Byron Site visit that is scheduled to start on October 29, 1984.

V. I. Schlosser
Project Manager
Byron Station

VIS/ML/sg:106kj

cc: V. Schlosser
G. Sorensen
R. Tuetken
M. Lohmann
K. Hansing
W. Shewski
B. Shelton
E. Hemzy
J. Streeter, NRC
J. Hinds, NRC
C. Allison, NBB
M. Sullivan, NBB
R. Holt, NBB
D. Stewart, HSB
B. Rainey, HSB
M. Somsag, Hunter
R. Larkin, PAP
D. Stringer, NISCo

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

September 10, 1984

Mr. J. Blackburn
Illinois Department of Nuclear Safety
3150 Executive Park Drive
Springfield, Illinois 62706

Subject: Byron Station Units 1 & 2
ASME Code Case N-302

Dear Mr. Blackburn:

Commonwealth Edison Company, at its Byron Nuclear Station is using ASME Division 1 Section III Code Case N-302 "Tack Welding". This Code Case has been accepted with no clarification by the NRC in Reg. Guide 1.84, revision 22 dated July, 1984.

Very truly yours,

D. Elias

D. Elias
Project Engineer

BA/sb/4977b
cc: B. Annis
B.R. Shelton
M. Lohmann
Mr. Don McDonald - The National Board of B&PV Inspectors
1055 Crupper Avenue
Columbus, Ohio 43229

ENCLOSURE 7

O'shan NRR
h/m

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

OCT 10 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 closeout inspection conducted by Mr. R. S. Love of this office on September 17-21, 1984, of activities at Byron Station, Units 1 and 2 authorized by NRC Construction Permit No. CPPR-130 and No. CPPR-131 and to the discussion of our findings with Mr. K. J. Hansing 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.

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

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 and the enclosed inspection report will be placed in the Public Document Room.

~~8410310277~~

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

Sincerely,

Roger D Walker for

R. L. Spessard, Director
Division of Reactor Safety

Enclosure: Inspection Report
No. 50-454/84-69(DRS) and
No. 50-455/84-47(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

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-454/84-69; 59-455/84-47(DRS)

Docket No. 50-454; 50-455 License 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 Site, Byron, Illinois

Inspection Conducted: September 17-21, 1984

Inspector(s): R. S. Love RS Love 10/10/84
Date

Approved By: C. C. Williams, Chief C.C. Williams 10/10/84
Plant Systems Section
Date

Inspection Summary

Inspection on September 17-21, 1984 (Report No. 50-454/84-69; 50-455/84-47(DRS)

Areas Inspected: Routine, unannounced inspection of licensee actions on previous inspection findings, 10 CFR 50.55(e) reports and IE Bulletins. This inspection involved a total of 37 inspection-hours on-site by one NRC inspector, including 2 inspection-hours during off-shifts.
Results: Of the areas inspected, no items of noncompliance or deviations were identified.

~~8410310296~~

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

*K. J. Hansing, Quality Assurance Superintendent
*D. L. Vandgrift, Project Quality Control Engineer
*J. W. Rappeport, Quality Assurance Engineer
*J. L. Bergner, Quality Assurance Supervisor
*E. T. Sager, Electrical Field Engineer
*M. V. Dellabetta, Quality Assurance Engineer
*J. O. Binder, Project Electrical Supervisor
R. B. Klinger, Project Quality COntrol Supervisor

Hatfield Electric Company (HECo)

A. Smith, QA/QC Manager
S. Bindenagel, Assistant QC Supervisor
T. Ahlquist, Lead QC Inspector

Sargent and Lundy (S&L)

T. B. Thorsell, Senior Electrical Project Engineer

The inspector also contacted and interviewed other licensee and contractor personnel during this reporting period.

*Denotes those persons present at the exit interview on September 21, 1984.

2. Licensee Action on Previously Identified Items

a. (Closed) Noncompliance (454/84-27-01; 455/84-19-01): During a previous inspection it was identified that the licensee failed to identify and control nonconforming cable tray hangers during the hanger reinspection required by HECo nonconformance report (NCR) 407R. As a result of the inspector's concerns, 295 hangers were reinspected. This reinspection resulted in 2 HECo NCRs, 1 CECo NCR, and 44 HECo deficiency reports (DR) being prepared to document potential discrepancies. During a previous inspection (454/84-47; 455/84-41), the inspector reviewed 19 of the closed DRs and found the corrective aciton to be adquate. During this inspection, the inspector reviewed the following closed NCRs and DRs:

(1) DR 5419, dated July 17, 1984. Only 1 tube steel section installed and the drawing indicated that 2 tube steel sections should be installed. Field Change Report (FCR) 25193 was issued to correct the drawing. The DR was closed on August 16, 1984.

(2) DR 4925, dated May 10, 1984. Tube steel thickness was 1/16" undersized. FCR 25116 was issued to accept the tube steel as installed. The DR was closed on July 27, 1984.

(3) DR 4929, dated May 10, 1984. Tube steel length was shorter than specified on the drawing. FCR 25075 was issued to correct the drawing. The DR was closed on July 27, 1984.

(4) DR 4921, dated May 10, 1984. Oversized tube steel was installed. FCR 25085 was issued to accept the tube steel as installed. The DR was closed on July 27, 1984.

(5) DR 4945, dated May 15, 1984. Wrong connection detail was utilized. FCR 25086 was issued to accept the connection detail as installed. The DR was closed on July 27, 1984.

(6) DR 4946, dated May 11, 1984. One tube steel section intalled and the drawing indicated that 2 sections should be installed. FCR 4946 was issued to correct the drawing. The DR was closed on July 27, 1984.

(7) DR 4944, dated May 14, 1984. Tube steel length was shorter than specified on the drawing. FCR 25087 was issued to correct the drawing. The DR was closed on July 27, 1984.

(8) DR 4941, dated May 11, 1984. Wrong connection detail was utilized. FCR 25072 was issued to accept the connection detail as installed. The DR was closed on July 27, 1984.

(9) DR 5028, dated May 10, 1984. East vertical tube steel added, was not shown on the drawing. FCR 25089 was issued to correct the drawing. The DR was closed on July 27, 1984.

(10) DR 4942, dated May 11, 1984. Oversized tube steel was installed. FCR 25088 was issued to accept the tube steel as installed. The DR was closed on July 27, 1984.

(11) DR 4927, dated May 10, 1984. Wrong connection detail utilized and tube steel length was shorter than specified. FCR 25112 was issued to accept the detail as installed and to correct the tube steel length on the drawing. The DR was closed on July 27, 1984.

(12) DR 5013, dated May 10, 1984. Wrong connection detail utilizied. FCR 25112 was issued to accept the detail as installed. The DR was closed on July 27, 1984.

(13) DR 5027, dated May 14, 1984. DV-85 connection detail plate size reduced. FCR 25076 was issued to accept the plate as is. The DR was closed on July 27, 1984.

(14) DR 5018, dated May 11, 1984. Welds were rusty. Welds were cleaned and painted, and the DR was closed on August 16, 1984.

(15) DR 4923, dated May 10, 1984. Wrong connection detail utilized, plate size was increased. FCR 24867 was issued to accept the plate as installed. The DR was closed on July 14, 1984.

(16) DR 4933, dated May 10, 1984. Wrong connection detail utilized and welds rusty. FCR 25113 was issued to acccept the detail as installed and the welds were cleaned and painted. The DR was closed on August 6, 1984.

(17) DR 5003, dated May 10, 1984. Eight one inch return welds missing. FCR 25126 was issued to accept the welds as installed. The DR was closed on July 28, 1984.

(18) DR 4934, dated May 10, 1984. Wrong size tube steel was installed. FCR 25130 was issued to accept the hanger as installed. The DR was closed on July 27, 1984.

(19) DR 5026, dated May 11, 1984. DV-84 connection was not installed per detail. FCR 25084 was issued to accept the hanger as installed. The DR was closed on July 27, 1984.

(20) DR 4932, dated May 10, 1984. DV-84A connection was not installed per detail, clearance violation. FCR 25074 was issued to accept the hanger as installed. The DR was closed on July 27, 1984.

(21) DR 5025, dated May 14, 1984. Clip angle length was reduced 1/4". FCR 25119 was issued to accept the hanger clips as installed. The DR was closed on July 27, 1984.

(22) DR5023, dated May 14, 1984. Auxiliary steel connection was not per drawing. FCR 25121 was issued to accept the auxiliary steel as installed. The DR was closed on July 27, 1984.

(23) DR5022, dated May 14, 1984. DV-84A connection was not installed per detail, clearance violation. FCR 25083 was issued to accept the hanger as installed. The DR was closed on July 27, 1984.

(24) DR5017, dated May 11, 1984. Auxiliary steel alignment, off-center, violates tolerance for DV-84A connection. FCR 25082 was issued to accept the hanger as installed. The DR was closed on July 27, 1984.

(25) DR5007, dated May 11, 1984. Hanger weld was rejected for lack of penetration. Weld was repaired and the DR was closed on September 6, 1984.

(26) HECo NCR 989, dated May 14, 1984. Ninety one hangers were found with excessive gap on the DV-84 connection details. ECN 7824 was issued to increase the allowable gap to 3/4". FCR 25115 was issued to accept the hangers as installed. The NCR was closed on September 20, 1984.

(27) HECo NCR 990, dated May 14, 1984. During verification of pan hanger attachment (NCR 407R), 19 hangers were identified as being inaccessible due to concrete or block walls covering the

hanger attachments. CECo NCR F923 was prepared to transmit the HECo NCR to S&L for disposition. The disposition on these NCRs, 390 and F923, was to accept the hangers without reinspection based on the results of the total reinspection effort, (4000 + hangers). Both NCRs were closed September 20, 1984.

The corrective action on the above listed DRs and NCRs appears to be adequate. This item is closed.

b. (Closed) Noncompliance (454/84-27-02; 455/84-19-02): During a previous inspection it was observed that the HECo procedures failed to address the inspection of cable trays to verify the minimum separation requirements. As a result of the inspector's concerns, reinspection of cable tray installed since February 1983 was initiated by HECo. Cable tray installed prior to February 1983 had been 100% reinspected for minimum separation requirements under a previous reinspection program. To supplement HECo's reinspection effort, the licensee directed S&L to perform a reinspection of all safty-related trays to verify separation requirements between safety-related and non-safety-related cable trays. On September 26, 1984, Mr. E. T. Sager (CECo) telephonically informed Mr. R. S. Love (Region III) that S&L had completed their reinspection effort on September 19, 1984. Mr. Sager also stated that an ECN would be issued to direct HECo to install cable tray covers as required. The installation of covers reduces the minimum separation required to one inch. Based on the HECO and S&L reinspections and the program in place to verify installation of tray covers, this item is closed.

c. (Closed) Noncompliance (454/83-49-04; 455/83-35-04): During a previous inspection it was identified that electrical cable grips were not being properly installed in cable tray risers. It was also identified that HECo Procedure 10, "Class I Cable Installation", did not address the requiremetn for QC to verify the proper installation of cable grips. During a previous inspection (454/84-47; 455/84-41), the inspector was able to satisfy all concerns in this area except, procedure revision and the proper installation of the last cable grip prior to termination. When cables enter a panel from the bottom, a cable grip failure could cause excessive stress on the terminations.

During this inspection, it was observed that the licensee had reworked the cable grips in the control room panels where cable entry is from the bottom. The cable grips inspected appeared to be providing adequate support to the cables so as not to stress the terminations during a seismic event. The inspector also reviewed draft Revision 22 to Procedure 10. This procedure now requires QC to inspect cable grip for proper installation and document this inspection on Form HP-105. Based on the above observations, this item is closed.

3. Licensee Action on 10 CFR 50.55(e) Reports

a. (Closed) 50.55(e) Report (454/83-14-EE; 455/83-14-EE): As a result of Region III inspector's concerns (454/83-49-04; 455/83-35-04) and

CECo NCRs F-852 and F-869 in the area of electrical cable grip installations, the licensee filed a potential 50.55(e) report. Based on the information contained in Paragraph 2.c above, this item is closed.

4. Licensee Action on IE Bulletins

a. (Closed) Bulletin (454/80-20-BB): "Failure of Westinghouse Type W-2 Spring Return to Neutral Control Switches." This bulletin was issued when discrepancies (intermitten contact of neutral contacts) were observed in the W-2 spring-return-to-neutral control switches. In the licensee's response of April 30, 1981 (T. R. Tramm, CECo, to James G. Keppler, Region III), it was indicated that all safety-related W-2 switches would be replaced at the Byron Station. Based on this information, personnel interviews, and review of records, the Region III inspector closed this item in Inspection Report 454/84-23 and 455/84-16. On August 29, 1984, the licensee amended his response of April 30, 1981 to indicate that 118 W-2 switches were not replaced for one or more of the following reasons:

(1) The switch is of the maintaining contact type, not the spring-return-to-normal type described in IE Bulletin 80-20.

(2) A failure of the neutral position contact will not affect the operation of safety-related equipment because the contact is not used in a control circuit.

(3) The switch does not perform a safety-related control function.

(4) The switch is used for testing purposes only.

(5) The switch is located on a switchgear cubicle and is functional only when the breaker is in the test position.

Based on a review of the amended response by Region III Operations and Engineering personnel, this response was found acceptable and this item is closed.

b. (Closed) Bulletin (455/82-04-BB): Deficiencies in Primary Containment Electrical Penetration Assemblies. The purpose of this bulletin was to inform licensees about findings concerning electrical penetrations supplied by the Bunker Ramo Company. For Byron Station, Bunker Ramo electrical penetrations are only installed in Unit No. 2. Based on CECo's analysis and inspections of the Bunker Ramo penetrations, the following corrective actions were taken:

(1) Penetrations 2SI01E-2P1E and 2SI02E-2P2E were replaced with Conax Adapter Modules.

(2) Replaced a total of 8 conductor termination lugs that failed the pull test in the following penetrations:

- 2SI08E-2K4R, replaced 4 lugs
- 2SI04E-2C2E, replaced 1 lug
- 2SI03E-2C1E, replaced 2 lugs
- 2SI07E-2K3R, replaced 1 lug

(3) Prepared NCR F-788, dated February 23, 1983. This NCR documents that ring torque termination lugs on instrumentation penetrations are not crimped tightly on the conductor insulation. Based on the pull test of 6,454 connectors these lugs were accepted as installed. There were 8 safety-related and 2 non-safety-related failures. The NCR was closed on June 2, 1983.

Complete details of CECo's inspection effort at Byron Station is contained in NUREG/CR-3795.

During this inspection, the following observations were made by the Region III inspector:

(a) During a review of records, it was determined that inspection reports were not prepared for the initial inspections required by the subject bulletin. During interviews with CECo personnel, the inspector was informed that the inspections were performed by a CECo field engineer. The inspector was unable to verify that the subject field engineer was in fact certified to perform the penetration inspections.

(b) During a review of noncomformance reports, it was observed that NCR F-788 was prepared to document that inproper terminations were made on instrumentation penetrations *2SI03E, *2SI04E, *2LV01E, 2LV02E, 2LV03E, 2LV04E, *2SI05E, *2SI06E, *2SI07E, *2SI08E, 2LV05E, 2LV06E, 2LV07E, and 2LV08E (Ref. Paragraph 4.b.(3) above). THe asterisk denotes safety-related penetrations.

It was also observed that NCRs had not been prepared on the 4 penetrations where one or more of the manufacturer's terminations failed the pull test and had to be replaced (Ref. Paragraph 4.b.(2) above). Also, NCRs had not been prepared on the two Bunker Ramo penetrations that were replaced with Conax adapter modules (Ref. Paragraph 4.b.(1) above). Because the licensee tracked this matter in the context of an open Bulletin item and took all of the appropriate corrective actions (also see CECo QA Surveillance Report 6503) over a long period of time, the omission of a nonconformance report is not, in this instance, considered an enforcement matter.

(c) During the inspection of terminations per Bulletin 82-04, the licensee observed that the terminal block screws on the vendor terminations could not be retorqued to 18 $\pm$ 3 inch-pounds per the vendor drawings without damaging and deforming the screw heads. This was documented on CECo NCR F-789, dated February 23, 1984. The resolution was to torque the screws to 10 inch-pounds. The NCR was closed September 23, 1983.

(d) Reviewed HECo inspection reports on the replacement of faulty vendor terminations and found them to be adequate (Ref. Paragraph 4.b.(2) above).

(e) Reviewed records for the replacement of Bunker Ramo penetration 2SI01E-2P1E with Conax adapter modules. Following is the sequence of events as determined by the records reviewed and personnel interviews:

- CECo to HECo "Speed letter dated March 7, 1984, informed HECo of the penetration modules to be replaced.
- March 22, 1984, Bunker Ramo penetration feed throughs were replaced with Conax feedthroughs.
- CECo to HECo "Speed Letter" dated April 5, 1984, directed HECo to remove Port A on this penetration and return it to Conax for repair because of excessive leakage. No NCR was prepared to document this, however, these issues were being tracked as an open Bulletin item as described in sub-paragraph (b) above and in open inspection reports.

- April 12, 1984, Port A was removed per HECo Work Request No. 1922.
- July 12, 1984, Port A was reinstalled per HECo Work Request No. 1922 and QC inspected as documented on HECo Supplemental Report No. 48. The manifold was pressurized to 20 pounds, however, no leak rate test was performed at this time.
- On September 21, 1984, a satisfactory leak rate test was performed on this penetration and inspection reports No. 48 and No. 48 Supplement were sign-off as complete.

(f) Reviewed records for the replacement of Bunker Ramo penetration 2SI02E-2P2E with Conax adapter modules. The sequence of events were basically the same as for penetration 2SI01E-2P1E discussed in paragraph (e) above. The diffferences being: (1) this penetration was replaced on March 16,1984, and (2) Port D had to be returned to Conax for repair.

(g) On July 23, 1984, CECo prepared NCR F-926 to document the fact that polysulfone bushing portion of Conax support bushing subassembly Adapter Modules have cracks in the polysulfone material. Penetrations affected are: 1AP85EA; 2AP84EB; 2AP85EC; 2AP85ED; 2RD12E; 2RD13E; 2RD15E; 2RD16E; 2RD17E; 2RY04E; 2RY05E; 2RY06E; 2RY07E; *2SI01E; and *2SI02E (* indicates safety-related). Conax telex dated June 27, 1984, states that stainless steel replacement support bushings will be manufactured and shipped to both Byron and Braidwood Stations which will be used in place of the existing polysulfone bushings which have experienced cracking. The Construction Deficiency Evaluation (by CECo Project Engineering Department-off-site) attached to this NCR indicates that this item is not reportable per the requirements of 10 CFR 50.55(e). The date of this evaluation is August 2, 1984. Note: The Conax telex appears to be in response to Braidwood NCR L-626, dated June 14, 1984, as referenced in S&L to CECo letter dated July 18, 1984. As of September 21, 1984, this NCR is still open.

(h) September 20, 1984, the inspector visually inspected the following safety-related penetations and no discrepancies were identified:

- 2SI04E-2C2E, #14 AWG, observed that the termination lug for the wire landed on TB34, termination 12 had been replaced.
- 2SI02E-2P2E, #2 AWG, Conax Adapter Module installed.
- 2SI07E-2K3R, #16 AWG.
- 2SI03E-2C1E, #14 AWG, observed that the termination lugs on wires landed on TB6, terminations 3 and 9 had been replaced.
- 2SI01E-2P1E, #2 AWG, Conax Adapter Module installed.
- 2SI05E-2K1R, #16 AWG

This inspection resulted in the review of 2 of 4 penetrations with #16 wire, 2 of 2 with #14 wire, and 2 of 2 with #2 wire.

(i) September 21, 1984, CECo prepared QA Surveillance Report 6503 to document the Region III inspectors concerns associated with the IE Bulletin 82-04 review. Pending a review of this surveillance report for adequate corrective action and corrective action to prevent recurrence, this item is open (455/84-47-01).

5. Open Items

Open items are matters which have ben 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. An open item identified during this inspection is discussed in Paragraph 4.6.

6. Exit Interview

The Region III inspector met with the licensee representatives (denoted under Paragraph 1) at the conclusion of the inspection on September 21, 1984. The inspector summarized the purpose and findings of the inspection. The licensee acknowledged this information.