July 10, 1981



UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
| | ) | |
| UNION ELECTRIC COMPANY | ) | Docket No. STN 50-483 OL |
| | ) | |
| (Callaway Plant, Unit 1) | ) | |

APPLICANT'S ANSWERS TO INTERROGATORIES
OF JOINT INTERVENORS (FIRST SET)

Applicant UNION ELECTRIC COMPANY, pursuant to 10 C.F.R. §2.740b, hereby submits the following responses to "Joint Intervenors' First Set of Interrogatories to Union Electric." These responses are subject to and without waiver of Applicant's previously filed objections to certain of these interrogatories. Neither is the provision of answers to these interrogatories a representation that Applicant considers the information sought to be relevant to the issues to be heard in this proceeding.

INTERROGATORY NO. 1. Were one or more Stop Work Orders issued on June 9, 1977 suspending the use of safety related embedments in concrete pours pending an inspection of all the safety related embedments onsite? If the answer is in the affirmative, please state:

(a) The exact terms of each Stop Work Order;

(b) The reasons why each Stop Work Order was issued.

ANSWER. Daniel International Corporation ("DIC") Stop Work Order No. 08 was issued on June 9, 1977 to stop all

structural concrete pour operations. DIC Stop Work Order No. 09 was issued on June 9, 1977 to stop the issuance from field supply of embedments supplied under Bechtel purchase order 10466-C-131-2.

(a) The terms of DIC Stop Work Order Nos. 08 and 09 are set forth in the copies of such Stop Work Orders produced in response to Joint Intervenor's document request no. 9.

(b) The reasons for the issuance of DIC Stop Work Order Nos. 08 and 09 are set forth in the copies of such Stop Work Orders produced in response to Joint Intervenors' document request no. 9.

INTERROGATORY NO. 2. During the process of evaluating after June 9, 1977 whether the embedded plates presented a safety-significant problem, did Union Electric determine that some exceptions to structural welding code standards would be permissible? If so, please state fully:

(a) What exception(s) would be permissible;

(b) For each exception in (a), the reason(s) for the determination that the exception was permissible;

(c) The identify and location of any documents in your possession relating to the determination that exceptions were permissible.

ANSWER. Yes.

(a) The following exceptions to the structural welding code, as identified in the Standard Plant FSAR Section 3.8.3.6.4.3, were determined to be permissible for welding between manually welded rod anchors and plates embedded in concrete:

1. Vertical leg of weld may be up to 1/16 inch smaller than that specified on d:awings.
2. Unequal legs are permitted.
3. Weld profile and convexity requirements for these welds need not be imposed.
4. An undercut of up to 1/16 inch for 10 percent of weld length may be permitted.

(b) The welding exceptions identified in answer to interrogatory no. 2(a) were determined to be acceptable for the following reasons:

1. Reduction of the vertical leg of the weld up to 1/16 inch is allowed because the reduced weld size is capable of supporting the design loads without exceeding the allowable stresses.
2. Unequal weld legs are permitted on weld studs because this exception to normal structural welding practice does not affect the performance of weld studs. Unequal legs are not generally permitted for linear (structural) welding since this condition can result in thermal stress in one of the pieces being welded. However, since the stud is free to move under differential thermal strains during the welding process, no thermal stresses are developed. The movement that occurs is very small and does no affect stud performance.

3. Weld profile and convexity are also measures of the amount of heat transmitted to the elements being welded and are therefore not critical in stud welding for the same reasons given in Response 2(b)(2).

4. An undercut of up to 1/16 inch for 10 percent of the weld length is permitted because the resulting reduction in the base metal cross section is small and the remaining base metal area can support the design loads within the allowable base metal stresses.

(c) The determination of these exceptions is based in part upon calculations and in part upon engineering experience and knowledge of welding procedures. The determination that exceptions were permissible is discussed in ULNRC-238 produced in response to document request nos. 5 and 6. Samples of the involved calculations were submitted to the NRC in ULNRC-361.

INTERROGATORY NO. 3. Regarding those embedded plates fabricated on or before June 9, 1977 for use at the Callaway Plant, which have stud anchors attached by automatically-timed stud welding equipment (mechanically-welded embeds), provide the following information, separately for each fabricator:

(a) The name and address;

(b) The date(s) the embeds were shipped and the number in each shipment;

(c) Identify all documents which reflect the information provided in answer to this interrogatory.

ANSWER. (a) Embedded plates with headed stud anchors, fabricated and shipped prior to June 9, 1977, were supplied only by Cives Steel Company, 8 Church Street, Gouverneur, New York 13642.

(b) See Table 3(b), attached hereto.

(c) The information provided in the response to Interrogatory 3(b) was obtained from the following shipping lists supplied by Cives Steel Company:

1. P5266 dated 10-6-76
2. 26180 dated 11-29-76
3. P5347 dated 2-76
4. P5351 dated 2-76
5. P7767 dated 5-3-77
6. P5638 dated 7-7-76
7. P5641 dated 7-9-76
8. P5665 dated 8-3-76
9. P5672 dated 8-6-76
10. P5673 dated 8-6-76
11. P5593 dated 6-21-76
12. P5679 dated 8-76
13. P5681 dated 9-9-76
14. P5682 dated 9-9-76
15. P5687 dated 9-9-76
16. P5691 dated 9-22-76
17. P5692 dated 9-28-76
18. P7510 dated 10-24-76
19. P7586 dated 1-7-77
20. P5697 dated 3-31-77
21. P7621 dated 2-9-77
22. P7658 dated 3-14-77
23. P7697 dated 4-18-77
24. P7943 dated 6-6-77
25. P7944 dated 6-6-77
26. P7683 dated 3-23-77
27. P5642 dated 7-14-76
28. P5643 dated 7-16-76
29. P5655 dated 7-23-76

INTERROGATORY NO. 4. Regarding the mechanically-welded embeds fabricated by the Cives Steel Company and received at the Callaway Plant site on or before June 9, 1977, provide the following information:

(a) State the number of mechanically-welded embeds in each shipment to the Callaway plant site, and the date of each shipment;

(b) State the number installed on or before June 9, 1977, in each Seismic Class I structure and system;

(c) Describe the function(s) of the embeds in each such structure and system;

(d) State separately with respect to each of the functions in each structure and system identified in response to paragraph (c) of this interrogatory, what would result if an embed were to fail after construction of the plant is completed.

ANSWER. (a) See Table 4(a), attached hereto.

(b) Auxiliary Building = 353

Control Building = 88

Reactor building = 18

(c) The embedded plates with headed stud anchors, fabricated and shipped by Cives Steel Company on or before June 9, 1977, are used for the attachment of structural steel framing, as well as the attachment of HVAC, cable tray, and pipe support steel, to concrete walls and slabs.

(d) It is highly unlikely that any major system failure would occur if a given embed failed. Further, the remote probability of such an embed failure has been demonstrated by analysis. (See ULNRC-238) Tests have also been conducted on plates, which demonstrate their capability to carry the intended design load.

INTERROGATORY NO. 5. Regarding those embedded plates fabricated on or before June 9, 1977, for use at the Callaway Plant, which have rod anchors or studs attached manually (manually-welded embeds), provide the following information for each fabricator:

(a) The name and address;

(b) A description of the types of embeds fabricated, including whether the studs are threaded; and the number of each type;

(c) The date(s) the embeds were shipped; and the number in each shipment;

(d) Identify all documents which reflect the information provided in the answer to this interrogatory.

ANSWER. (a) Embedded plates with manually welded rod anchors (identified in the response to Interrogatory 5(b)), fabricated and shipped prior to June 9, 1977, were supplied only by Cives Steel Company, 8 Church Street, Gouverneur, New York 13642.

(b) See Figure 5(b) (12 pages), attached hereto, for a description of the types of embedded plates with manually-welded rod anchors fabricated and shipped by Cives Steel Company on or before June 9, 1977.

(c) See Table 5(c), attached hereto.

(d) The information provided in the response to Interrogatory 5(c) was obtained from the following shipping lists supplied by Cives Steel Company:

| | |
|---|---|
| 1. P5350 dated 2-76 | 8. P5314 dated 2-76 |
| 2. P5352 dated 2-76 | 9. P5267 dated 1-16-76 |
| 3. P5607 dated 6-21-76 | 10. P5268 dated 1-16-76 |
| 4. P5640 dated 7-7-76 | 11. P5328 dated 2-76 |
| 5. P5676 dated 8-76 | 12. P5346 dated 2-76 |
| 6. P7945 dated 6-6-77 | 13. P5349 dated 2-76 |
| 7. P5315 dated 2-76 | 14. P5644 dated 7-16-76 |

INTERROGATORY NO. 6. Regarding the manually-welded embeds fabricated by the Cives Steel Company and received at the Callaway plant site on or before June 9, 1977, provide the following information:

(a) State whether all studs on the manually-welded embeds were threaded; if the answer is negative: (i) Describe those plates with studs not threaded; (ii) State the number of each (threaded and not threaded) at the Callaway plant site;

(b) State the number of manually-welded embeds in each shipment to the Callaway plant site, and the date of each shipment;

(c) State the number installed on or before June 9, 1977, in each Seismic Class I structure and system;

(d) Describe the function(s) of the embeds in each such structure and system; state in each case whether the embed studs were threaded or not;

(e) State separately with respect to each of the functions in each structure and system identified in response to paragraph (d) of this interrogatory, what would result if an embed were to fail after construction of the plant is completed.

ANSWER. (a) Embedded plates with manually welded rod anchors, fabricated and shipped by Cives Steel Company on or before June 9, 1977, contain threaded and unthreaded anchor rods.

(i) Of the subject embedded plates, only the plates identified with plate mark numbers EP112, EP112A, EP212, EP312 and EP412 have manually welded rod anchors without threads. These rod anchors, which are not threaded, were oversize in design to account for the possibility of threading. For details of these plates see Figure 5(b).

(ii) The numb of embedded plates with manually welded rod anchors, fabricated and shipped by Cives Steel

Company on or before June 9, 1977, totalled 689 with threaded rod anchors and 39 with unthreaded rod anchors.

(b) See Table 6(b), attached hereto.

| | | |
|---|---|---|
| (c) Auxiliary Building | = | 167 |
| Control Building | = | 65 |
| Reactor Building | = | 0 |

(d) The embedded plates with manually welded rod anchors, installed in the wall prior to June 9, 1977, are used for the attachment of structural steel framing members to the concrete.

(e) It is highly unlikely that any major system failure would occur if a given embed failed. Further, the remote probability of such an embed failure has been demonstrated by analysis. (See ULNRC-238) Tests have also been conducted on plates, which demonstrate their capability to carry the intended design load.

INTERROGATORY NO. 7. With respect to the period on and after June 9, 1977, please describe fully any inspection or test done by Union Electric or anyone else of the following embeds fabricated by Cives:

(a) Those installed in the Callaway Plant prior to the issuance of the Stop Work Orders on June 9, 1977;

(b) Those onsite at the Callaway Plant but not yet installed when the Stop Work Orders were issued on June 9, 1977;

(c) Any embedments that had been fabricated for the Callaway Plant, but were not yet received at the plant site by the time the Stop Work Orders were issued.

ANSWER. (a) See Report on Testing to Evaluate Welds of Anchor Rods and Studs to Embedded Plates, Bechtel Power Corporation, Gathersburg, Maryland, September 15, 1980, a copy of which will be produced at the same time and place as Applicant's document production.

(b) See ULNRC-238 and attachments thereto being produced in response to document request nos. 5 and 6.

(c) All embedments that had been fabricated for the Callaway Plant, but were not yet received at the plant site by the time the Stop Work Orders were issued, were reinspected by CIVES Steel Company at the CIVES shop or at the Callaway Plant. Inspection of manually welded rod anchors met, as a minimum, the requirements of AWS D1.1-75, with the exceptions identified in the answer to interrogatory no. 2(a). Inspection of headed stud anchors were to AWS D1.1-75 Section 4.30.2, prior to August 18, 1977, and to AWS D1.1-75, Section 4.30.1, after that date. However, all these headed stud anchors were inspected to insure a 360° fillet weld.

INTERROGATORY NO. 8. With respect to the preceding interrogatory and separately with respect to each subpart thereof, please describe fully:

(a) The results of any such inspections or tests;

(b) If no inspection was done, the reasons why no inspection was done;

(c) Any repairs, replacements, or other remedial or precautionary measures that were taken with respect to embeds, including an enumeration of the following:

(i) the number of manually welded embeds repaired on site;

(ii) the number of manually welded embeds returned to the fabricator for repair or replacement;

(iii) the number of mechanically welded embeds repaired on site;

(iv) the number of mechanically welded embeds returned to the fabricator for repair or replacement;

(d) The identity and location of any documents relating to any of the measures specified in answer to this interrogatory.

ANSWER. (a) See documents identified response to interrogatory no. 7(a), and (b).

(b) Not applicable.

(c) Objected to. Subject to and without waiver of its objection to subpart (c) of this interrogatory Union Electric Company provides the following information:

(ii) The number of manually welded embeds returned to the fabricator for repair or replacement:

(a) Installed in the Callaway Plant prior to June 9, 1977 = 0 (NOTE: <u>Installed</u> is taken to mean poured in concrete).

(b) Onsite at the Callaway Plant but not yet installed on June 9, 1977 = 12.

(c) Fabricated for the Callaway Plant but not yet received by the time of the Stop Work Order (June 9, 1977) - Those in transit on June 9, 1977 are included in (b) above.

(iv) The number of mechanically welded embeds returned to the fabricator for repair or replacement:

(a) Installed in the Callaway site prior to June 9, 1977 = 0 (Note: Installed is taken to mean poured in concrete).

(b) Onsite at the Callaway Plant but not yet installed on June 9, 1977 = 36.

(c) Fabricated for the Callaway Plant but not yet received by the time of the Stop Work Order (June 9, 1977) - Those in transit on June 9, 1977 are included in (b) above.

(d) Objected to. Subject to and without waiver of its objection to subpart (d) of this interrogatory Union Electric Company provides the following information:

(1) Stop Work Order #8 produced in response to document request no. 9.

(2) Welding quality control surveillance reports; various dates; inspection and repair reports of welded items; prepared by Daniel International Welding Quality Control; located at the Callaway Site. Welding quality control surveillance reports contain welding inspections done on embedded plates, round embedded sleeves, rectangular embedded sleeves, embedded frames, structural steel, miscellaneous

steel, handrail, and other civil-related items. They include inspections conducted in the plant proper, in the fabrication shop, and in laydown and warehouse areas.

(3) Engineered and Material Equipment Return #1077; 6/27/77; list of material returned to Cives Corporation as defective; prepared by Daniel International Warehouse Personnel; located at the Callaway Site.

INTERROGATORY NO. 9. State whether any embeds were cut out of wooden forms for inspection after June 9, 1977. If answer is in the affirmative:

(a) State how many embeds were cut out;

(b) Identify all documents which pertain to such inspections.

ANSWER. Embedded plates were removed from forms after June 9, 1977. The normal removal procedure would not involve cutting the form.

(a) The number of embeds removed from forms is unknown and was not recorded.

(b) See answer to interrogatory no. 8(d).

INTERROGATORY NO. 10. Regarding Nuclear Regulatory Commission (NRC) Report No. 50-483/78-01, what was the width of the crack mentioned on p. 20, entry 13a, in the "plant" north wall of the Control Building?

ANSWER. One-sixteenth inch (1/16") at the widest place.

INTERROGATORY NO. 11. Regarding Nuclear Regulatory Commission (NRC) Report No. 50-483/78-01, state the number of the "other cracks" referred to on page 20 of this report.

ANSWER. Fourteen (14).

INTERROGATORY NO. 12. State the location, length, width and shape of each crack counted in the answer to the preceding interrogatory.

ANSWER. See NRC 2-2081-C-A produced in response to document request no. 14.

INTERROGATORY NO. 13. State the reason or reasons why the cracks referred to on page 20 of NRC Report No. 50-480/78-01 are described as a "recurring problem," that is, why are such cracks recurrent?

ANSWER. It is assumed that reference to NRC Report No. 50-480/7801 is intended to read No. 50-483/78-01.

Small cracks of this nature are common and are expected to occur on a limited basis in reinforced concrete construction. No corrective or preventive measures known to the construction industry will absolutely assure similar cracking will not occur.

INTERROGATORY NO. 14. What, in the opinion of Union Electric, is the cause of

(a) the twelve foot long crack described on page 20, entry 13a of NRC Report No. 50-483/78-01;

(b) the "other cracks" described on page 20, entry 13a of NRC Report No. 50-483/78-01. Each crack should be separately addressed if there are different causes for different cracks.

ANSWER. (a) The twelve foot long crack described on page 20, entry 13a of NRC Report No. 50-483/78-01 was caused by the volume change and resulting strains associated with the drying shrinkage of the concrete during the normal curing process.

(b) The "other cracks" described on page 20, entry 13a of NCR Report No. 50-483/78-01 were caused by the volume change and resulting strains associated with the normal drying shrinkage of the concrete during the curing process.

INTERROGATORY NO. 15. What reason or reasons does Union Electric have to believe that concrete such as that described on page 20, entry 13a of NRC Report No. 50-483/78-01 will not develop additional cracks?

ANSWER. The cracks described on page 20, entry 13a of NRC Report No. 50-483/78-01 occurred at an age when the concrete was undergoing its most severe volume changes caused by drying shrinkage when the ultimate tensile strength was not yet fully developed. Generally, further cracking due to this condition does not continue as the concrete ages. Concrete is a non-homogenous construction material with very low tensile strength capacity and may exhibit additional cracking as the result of externally applied loads and temperature changes. The design methods used for reinforced concrete, however, consider the concrete to crack under tension loading.

INTERROGATORY NO. 16. What reason(s) does Union Electric have to believe that the cracks described on page 20, entry 13a of the NRC Report No. 50-483/78-01 will not increase in size.

ANSWER. It is possible for the cracks described on page 20, entry 13a of the NRC Report No. 50-483/78-01 to vary slightly in size, primarily as the result of temperature changes and to a lesser extent due to external loads, creep, and continued drying shrinkage. These changes, which in total

could either increase or decrease the reported dimensions, will be extremely small since the significant cracking mechanism in these walls, i.e., volume change and resulting strains due to concrete drying shrinkage, has already occurred.

INTERROGATORY NO. 17. Have any employees, agents or affiliates (including contractors and subcontractors) of Union Electric measured any of the cracks described on page 20, entry 13a of NRC Report No. 50-483/78-01 subsequent to the filing of Report No. 50-483/78-01? If so, state the following:

(a) identify all documents in the possession or control of Union Electric in which there is reference to the subsequently measured cracks;

(b) identify personnel who made the subsequent measurements;

(c) summarize Union Electric's assessment of the subsequent measurements.

ANSWER. Yes.

(a) See NCR 2-2173-C-A produced in response to document request no. 15.

(b) The personnel who made the subsequent measurements are Joe Kleiner (DIC Superintendent) and Dave Dunning (DIC Q.C. Inspector).

(c) Applicant believes that the measurements on Nonconformance Report 2-2173-C-A adequately address the extent and the location of the cracks.

INTERROGATORY NO. 18. Are the cracks described on page 20, entry 13a of NRC Report No. 50-483/78-01 still accessible to visual or instrument inspection?

ANSWER. On or about June 18, 1981, an inspection was conducted by Daniel Civil Engineering of all cracks that were

indicated on NRC Report No. 50-483/78-01 and delineated on Nonconformance Report No. 2-2081-C-A. During the inspection, the following information was obtained:

(a) For crack #1, the location of the crack is inaccessible from the Control Building side of the wall.

(b) For crack #2, the location is accessible.

(c) For cracks #3 and #4, the slab at Elevation 2056'-10" covers the area making these two cracks inaccessible.

(d) For crack #5, only that portion above the slab at Elevation 2056'-10" is visible from the Control Building. Below the slab, the area is between two block walls and cannot be checked.

(e) Crack #6 is visible above the slab at Elevation 2056'-10" from the Control Building side of the wall. Below this slab, the wall is painted and the crack does not show.

(f) Cracks #7, #8, #9 and #10 are all accessible.

(g) Cracks #11, #12 and #13 have been patched per Nonconformance Report No. 2-2173-C-A. The patches are visible.

(h) Crack #14 is visible as a hairline crack on the Communications Corridor side of the wall.

(i) Crack #15 has been patched and the area is accessible.

INTERROGATORY NO. 19. Regarding Nuclear Regulatory Commission (NRC) Report No. 50-483/78-03, what is meant on page 3 by the statement that NCR 2-2081-C-A was "superceded" by NCR 2-2173-C-A?

ANSWER. The statement on Page 3 of Nuclear Regulatory Commission (NRC) Report No. 50-483/78-03, that Nonconformance Report 2-2081-C-A was "superceded" by Nonconformance Report 2-2173-C-A, meant that Nonconformance Report 2-2081-C-A was voided and returned from Bechtel and would not be evaluated or dispositioned; and that Nonconformance Report 2-2173-C-A was being submitted to Bechtel for evaluation and disposition. The first Nonconformance Report contained information and described conditions which, upon subsequent investigation and review of criteria, were not considered nonconformances by either Daniel or Bechtel.

INTERROGATORY NO. 20. State Union Electric's assessment of the safety significance of the twelve foot crack in the Control Building wall, referred to on page 3 of NRC Report No. 50-483/78-03 and on page 20 of the NRC Report No. 50-483/78-01.

ANSWER. There is no adverse safety significance resulting from the twelve foot crack in the control building wall, referred to on page 3 of NRC Report No. 50-483/78-03 and on page 20 of NRC Report No. 50-483/78-01. Concrete is a non-homogeneous construction material with a very low tensile strength capacity, and may exhibit surface and through-thickness cracking as a result of internal tensile stress caused by the volume change due to drying shrinkage, creep, externally applied loads and temperature changes. The

design methods used for reinforced concrete, however, consider the concrete to crack under tension loading.

INTERROGATORY NO. 21. Regarding Nuclear Regulatory Commission (NRC) Report No. 50-483/77-06, and the circumferential concrete crack in the Reactor Containment Building referred to on pp. 20-21, state the following:

(a) the length of the crack;

(b) the depth of the crack;

(c) the width of the crack;

(d) the proximity of the crack to reinforcing steel and other embedded materials.

ANSWER. (a) See NCR 2-0631-C-A produced in response to document request no. 16.

(b) See NCR 2-0631-C-A produced in response to document request no. 16.

(c) See NCR 2-0631-C-A produced in response to document request no. 16.

(d) The crack occurred in an unreinforced section of concrete, approximately one foot above the 45° bend of a #18 horizontal bar and between a vertical #11 and #18 bar approximately three feet each direction radially from the crack location. The embedded M4X13 is the only other embedded material in the area.

INTERROGATORY NO. 22. Identify any documents in the control or possession of Union Electric which make reference to the circumferential crack in the reactor cavity moat area, approximately 42 inches from cavity liner and extending through 270° arc, referred to on pp. 20-21 of NRC Report No. 50-483/77-06.

ANSWER. The following documents make reference to the circumferential crack in the reactor cavity moat area, referred to on pages 20-21 of NRC Report No. 50-483/77-06:

(a) NCR Number 2-0631-C-A- dated May 20, 1977;

(b) Trip Report May 10-11, 1977 -- Investigation of Concrete Crack at the Reactor Pit Moat Area by C. L. Miller, SNUPPS Civil Group Supervisor (Bechtel);

(c) Concrete Record of Repair-Grouting and Drypack Inspection Report dated June 2, 1977, Unit 1 Reactor Cavity Moat Blockout Elevation 1998'-5 3/4";

(d) Letters confirming telephone conversations

(1) BLUE 403 dated June 7, 1977

(2) BLUE 409 dated June 21, 1977

(3) BLUE 461 dated December 13, 1977;

(e) Letter BLUE 402 dated June 2, 1977;

(f) Stop Work Order No. 06, dated May 9, 1977;

(g) Telecon Report prepared by F. D. Field, Manager-QA, Union Electric Co. of telecon with D. W. Hayes, Chief-Projects Section, NRC-I&E/III, dated May 10, 1977; and

(h) Telecon Report prepared by F. D. Field, Manager-QA, Union Electric Co. of telecon with T. E. Vandel, Reactor Inspector, NRC dated May 12, 1977.

INTERROGATORY NO. 23. Summarize separately the content of each document listed in answer to the preceding interrogatory.

ANSWER: Items (a) through (c) listed in response to Interrogatory No. 22 have been produced in response to Document Request Nos. 16, 17 and 18, respectively. The enclosures to the letters listed as Item (d) contain summaries of telephone conversations between representatives of Union Electric Company, Bechtel Power Corp. and various other contractors. Those portions of the telephone conversation summaries relating to the concrete crack in the reactor cavity moat area are excerpted and set forth below.

From the Enclosures to BLUE 403:

5-10-77 UE (W. Zvanut and P. Divjak)

UE informed Bechtel that a concrete crack had occurred in the reactor cavity moat area in a circular pattern around a small embedded beam. The crack was detected 6 to 12 inches from the circular M 4 x 13 beam shown on drawing C-OL2911, Section F, and extended a circumferential distance of 270°. It was theorized that the crack may have been caused by the welding of the 1/4 inch liner plate to the embedded beam. The extension of the crack could not be determined without removing some line plate.

5-20-77 UE (D. Stecko and R. Lynch)

UE inquired about the progress of reports prepared by C. Miller regarding the concrete crack in the reactor cavity moat area...Bechtel will check C. Miller for status.

From the Enclosures to BLUE 409:

6-1-77 UE (W. Zvanut and R. Lynch)

UE inquired about Bechtel's report on the concrete crack in the moat area of the reactor cavity... Bechtel informed UE that... [it] will be telecopied to UE and A. S. Martin

in the morning of 6-2-77... Bechtel informed UE that NCR 2-0631-C-A for the concrete crack was signed off "approved as recommended" and telecopied to M. R. Hamby on 5-26-77.

6-3-77 UE (W. Zvanut, E. Swallow and R. Lynch)

Bechtel telecopied on 6-2-77 the report on the concrete crack in the moat area of the reactor cavity...

From the Enclosures to BLUE 461:

11-14-77 UE (W. Zvanut and R. Lynch)

UE inquired if Bechtel could furnish duplicate slides or 8" x 10" prints of the concrete crack in the moat area of the reactor cavity to satisfy an NRC request. If possible, UE would like four copies of 2 or 3 representative photos to be given to D. Capone in Gaithersburg today.

11-14-77 UE (D. Stecko and R. Lynch)

In regard to UE's earlier inquiry concerning photos or slides of the concrete crack in the moat area of the reactor cavity, Bechtel informed UE that they could not locate either. However, Bechtel will contact A. Martin at the jobsite to see if he still retained his slides.

11-16-77 UE (D. Stecko and R. Lynch)

Bechtel learned from A. Martin that he sent 8 original slides of the moat area crack to C. Proctor of UE on 6-7-77. A. Martin contacted C. Proctor on 11-15-77 about locating these slides. C. Proctor will search his files and also check the company photographer who used the originals for copies in June. D. Stecko will follow-up on this in-house.

(e) Bechtel letter BLUE 402, 6-2-77 transmitting Bechtel's trip report for inspecting the crack in the reactor cavity moat area.

(f) Stop Work Order No. 6, dated May 9, 1977. Stops work on erection and testing of all bottom liner plate and leak chase in the reactor cavity moat area (17-foot radius from centerline of reactor).

(g) Telecon report (5/10/77) Field to Hayes of the NRC, reporting a possible significant deficiency regarding a crack in the reactor cavity moat area.

(h) Telecon report F. Field to T. E. Vandel of the NRC (5/12/77) rescinding the significant deficiency regarding the crack in the reactor cavity moat area.

INTERROGATORY NO. 24. This interrogatory applies to the following sentence in NRC Report No. 50-483/77-06, pp. 20-21: "It was reported by licensee by telephone on May 10, 1977. . . <u>that an investigation had been initiated</u> to determine the safety related significance of the crack [in the reactor cavity moat area]" (emphasis added):

(a) State the names and titles of all persons involved in the investigation here referred to;

(b) Summarize the result of the investigation here referred to.

ANSWER. (a) (1) James A. Petsche - DIC Area Civil Engineer;

(2) Clayton Miller - Bechtel Civil Group Supervisor; and

(3) William Zvanut - UE Supervising Engineer.

(b) See document produced in response to document request no. 17.

INTERROGATORY NO. 25. Describe in detail the repairs made to the circumferential crack in the reactor cavity moat area, as further described in NRC Report No. 50-483/77-06, pp. 20-21.

ANSWER. Repairs made on the circumferential crack in the Reactor Cavity moat area involved chipping the entire crack

extent to sound concrete, and filling up to the top surface of the bottom flange of the embedded M4X13 with a flowable mixture of a non-shrink grout. The remaining area of removed concrete was replaced when the adjacent pour, 2C221S08, was placed. Dry-pack was used to repair the "tunnel-like" area at approximately 240° Azimuth which was chipped completely under the embedded M4X13 to determine whether the crack proceeded under the liner toward the Reactor Cavity. The section of liner plate which was removed and its associated leak chases were rewelded in accordance with a recommended welding sequence to minimize any weld shrinkage and distortion of the liner plate.

INTERROGATORY NO. 26. Identify all documents which relate to acceptance of the repair of the circumferential crack in the reactor cavity moat area by quality control personnel.

ANSWER. (1) Nonconformance Report 2-0631-C-A. See document produced in response to document request no. 16. (2) Grouting and Drypack Inspection Report. See document produced in response to document request no. 18. (3) DIC Compression Destructive Testing Reports dated June 5, 1977 and June 9, 1977. (4) DIC Concrete Pour Package 2C221S08.

INTERROGATORY NO. 27. State the cause of the circumferential crack in the reactor cavity moat area.

ANSWER. The probable cause of the circumferential crack in the reactor cavity moat area is discussed on Page 3 of the document produced in response to Document request no. 17.

INTERROGATORY NO. 28. Identify all documents which establish a reporting standard for cracks in permanent concrete at the Callaway Plant.

ANSWER. Construction Procedure - QCP-109 - Concrete Placement, Grouting and Post Pour.

INTERROGATORY NO. 29. Summarize the reporting standards set forth in the documents listed in answer to the preceding interrogatory.

ANSWER. See document produced in response to document request no. 19.

INTERROGATORY NO. 30. How many recurrences of cracks, as anticipated by NRC Report No. 50-483/78-01, page 20, have occurred since the date of that report?

ANSWER. Pursuant to the reporting standards in QCP-109, seventeen (17) cracks have been reported to DIC Engineering since the issuance of NRC Report Number 50-483/78-01.

INTERROGATORY NO. 31. Provide nonconformance report (NCR) numbers for any cracks counted in the answer to the preceding interrogatory.

(a) Describe any procedures, modifications of materials, modifications in design, implemented subsequent to NRC Report No. 50-483/78-01, which are intended to detect, lessen or eliminate the problem of concrete cracks at the Callaway Plant.

(b) State the dates on which items listed in the answer to paragraph (a) of this interrogatory were effective.

ANSWER. Of the seventeen (17) cracks reported in the answer to interrogatory no. 30, three (3) have been addressed on NCR's and are as follows: 2-1973-C; 2SN-2611-C; and 2SN-3593-C. Subsequent investigation revealed the other fourteen (14) cracks to be not nonconforming pursuant to the reporting standards in QCP-109.

(a) There have been no procedural changes made relating to NRC Report number 50-483/78-01. No modifications of materials or modifications in design have been implemented subsequent to NRC Report No. 50-483/78-01.

(b) Not applicable.

INTERROGATORY NO. 32. State separately for each reportable void repaired in the tendon access gallery ceiling its depth, width, length, location and shape:

(a) at the time of initial discovery

(b) after the chipping operation.

ANSWER. (a) At the time of initial discovery of the voids in the tendon gallery, it was not a requirement to document the condition prior to chipping.

(b) Subsequent to form removal in the tendon access gallery ceiling, nineteen (19) areas of honeycomb were addressed and chipped to sound concrete. These areas with respect to depth, width, length, location and shape were documented on Nonconformance report 2-0856-C-A, being produced in response to document request no. 31.

INTERROGATORY NO. 33. Regarding NRC Report No. 50-483/78-02, p. 4, and the statement that technical specification C-103(Q) was revised to differentiate major from minor defects which require approval prior to repair:

(a) summarize the differentiation between major and minor defects, as established by the revision of C-103(Q);

(b) state whether major and minor concrete defects were undifferentiated for purposes of repair approval prior to the January 23, 1978 revision of C-103(Q).

ANSWER. (a) The revision of Specification C-103(Q) referenced on page 4 of NRC Report No. 50-483/78-02 divides concrete defects into three classes. The first class, which is described in Section 15.1 of the specification, includes minor surface imperfections left by air bubbles as well as sand streaks. These defects are cosmetic in nature and require no repair.

The second class of defects, which is described in Section 15.3, includes relatively small defects which are of a size, nature and extent that can be repaired using the preapproved procedures outlined in this section of the specification. These defects are smaller in size than the defects described in Section 15.2. Although repairs of these defects are required and are documented by the constructor, they need not be reported to the design engineer.

The third class of defects, which is described in Section 15.2 of the specification, includes the following:

(1) Any void whose depth exceeds 1/3 the least dimension of the beam, column, wall or slab.

(2) Any void where reinforcing bars are exposed around their entire circumference for a length along the bar as follows: (a) #7 bars and smaller - 6 inches; (b) Larger than #7 bars - 9 inches.

(3) Any series of voids wherein the total exposed area of any one reinforcing bar is twice that given in (2) above.

The design engineer must approve the proposed repair procedures for this third class of defects before the constructor proceeds with the repairs.

(b) Prior to January 23, 1978, these classes of concrete defects were undifferentiated for purposes of repair approval.

INTERROGATORY NO. 34. State whether, in Union Electric's opinion, the honeycombing in the base mat resulted, in whole or in part, from the congestion of trumplate wall dowels, main steel, rebar supports, or form ties, or any or all of these singly or in combination, so that adequate vibration of the concrete mat was hampered.

ANSWER. As stated in NCR 2-0856-C-A, page 2, under "Cause of Nonconformance and Action to Prevent Recurrence" and in the Final Report to Mr. J. G. Keppler, September 29, 1977, the cause of the honeycombing was inadequate or insufficient vibration. This inadequate or insufficient vibration was due to the congested area over the tendon gallery; the mobility of crews was hampered by the trumplates, wall dowels, main steel, rebar supports, form ties, dense rebar matrix and the distance from the pour point to the bottom of the 10-foot thick slab.

INTERROGATORY NO. 35. Are there any areas of the base mat which are less marked than other areas by congestion as described in the preceding interrogatory, and, if the answer is affirmative, describe the differences in congestion by specific area.

ANSWER. Yes. The base mat was placed continuously in five (5) lifts, each approximately two feet in thickness. Of the five lifts, the middle three (3) are the least congested, being affected only by the shear ties, supports for the upper layers of reinforcing steel, an occasional anchorage for the NSSS supports and secondary shield walls and hooks on only a limited number of horizontal reinforcing steel bars in the top and bottom lifts.

Of the top and bottom lifts, the top lift is judged the least congested, primarily due to the fact that it has considerably less reinforcing steel than the bottom lift. Within the bottom lift, the area between the reactor cavity and the tendon gallery is the least congested, again because less reinforcing steel is required in this area. Comparing the area surrounding the reactor cavity and the area above the tendon gallery, the latter is judged the more difficult to consolidate, and on that basis is considered more congested. Both areas have five (5) layers of reinforcing steel, with the first four (4) layers being similar at both locations. Although the fifth layer of reinforcing steel is more concentrated near the reactor cavity, the spacing of reinforcing bars in this area allowed proper insertion of consolidation equipment and satisfies the ACI spacing requirements. The tendon gallery is considered more congested because of the presence of thick embedded plates, between the bottom surface and the reinforcing steel, and the trumpets attached to the embedded plates.

INTERROGATORY NO. 36. State Union Electric's conclusion as to the cause of honeycombing in the tendon access gallery concrete, attributing relative weight and probability to each cause if more than one cause is named. Also state:

(a) What actions were taken by Union Electric subsequent to NCR 2-0856-C-A to prevent the recurrence of voids in concrete such as those in the tendon access gallery, as described in NCR 2-0856-C-A;

(b) Whether Union Electric believes the preventive actions described in the answer to paragraph (a) of this interrogatory are adequate;

(c) The basis for Union Electric's conclusion as to the adequacy of preventive actions named in the answer to paragraph (a) of this interrogatory.

ANSWER. The honeycombing in the tendon access gallery concrete was caused by insufficient vibration of the toe of the first lift of concrete due to the congestion directly over the tendon access gallery. Mobility of the vibrator crews was hampered by the trumplates, wall dowels, main steel, rebar supports and form ties.

(a) Training sessions were given, on site, to all concrete placing crews. A training film was made in order to illustrate the proper methods to be used in the consolidation of concrete. In addition, in other highly congested areas, Daniel International Corporation requested, and received permission, to use a higher slump concrete to facilitate placement.

(b) Yes.

(c) The preventive actions stated in answer to paragraph (a) were adequate due to a lack of similar occurrences attributable to this same specific cause.

INTERROGATORY NO. 37. How often is vibrator frequency checked during concrete placements at the Callaway Plant?

ANSWER. Vibrator frequencies are checked and documented prior to use on each concrete placement and more often if determined to be necessary by the DIC Quality Control concrete placement inspector.

INTERROGATORY NO. 38. How often is vibrator frequency checked during concrete placements when vibrators are used continuously in concrete placement extending beyond one twelve-hour shift?

ANSWER. All vibratory units are checked and documented prior to each concrete placement, and periodically during the placement at the DIC Quality Control Inspectors' discretion, to verify that they are working properly. No specific time frame exists at Callaway for periodic in-process checks, with the exception of the Reactor base mat. These vibrators were checked and documented every twelve (12) hours.

INTERROGATORY NO. 39. Have technical specifications, quality control procedures or work control procedures relating to the checking of vibrator frequency been changed or modified during construction at Callaway Unit 1, and, if the answer is affirmative, state:

(a) the nature of the change or modification;

(b) the reason for the change or modification;

(c) the effective date of the change or modification;

(d) identify the document(s) effecting the change or modification.

ANSWER. Yes. Quality Control Procedures have been modified during construction at Callaway Unit 1.

(a) Modifications to Construction Procedure QCP-109 were made in direct proportion to the existing construction schedule and the number of concrete placements. The first progression was to upgrade the frequency of vibrator checks/documentation from monthly to bi-monthly. This modification was effective January 7, 1977 by revision 5 to QCP-109, paragraph 4.5.12. Revision 6 to QCP-109, dated March 11, 1977, added the requirement of minimum frequency stated in vibrations per minute; ACI 309-72 table 5.1.4. As a measure to increase control, ACI 309-72, paragraph 15.3 was incorporated in revision 8 on August 3, 1977, requiring vibrator oscillations be checked prior to each placement and documented for each safety related concrete pour. The final addition subsequent to the above, was revision 12 dated July 17, 1978. This incorporated the documentation of spare vibrator osciallations on the existing Procedural form.

(b) The Quality Control Procedure QCP-109 was in each case upgraded to incorporate more control as the construction work load increased.

(c) See answer to interrogatory no. 39(a).

(d) QCP-109.

INTERROGATORY NO. 40. Is Union Electric aware of any cold joining (improper bonding) in the concrete of the reactor base mat due to the drying (setting up) of concrete in certain areas before additional concrete was poured on top? If so, provide the following information:

(a) State how it was discovered;

(b) State the number of areas and describe the location of each such area.

(c) Does such a condition affect the strength of the concrete and, if so, describe how the strength is affected.

(d) Identify the document(s) which refer, in whole or part, to such conditions.

ANSWER. No.

(a) Not applicable.

(b) Not applicable.

(c) Not applicable.

(d) Not applicable.

INTERROGATORY NO. 41. The following interrogatory applies to the following sentence in NRC Report No. 50-4831 77-07, p. 4: "WP-109. . . does not identify either 'vibration', 'consolidation', or 'densification,' clarification of the procedureal requirements is required."

(a) State the implications for safety and quality assurance in the lack of clarification of procedural requirements as herein referred to;

(b) State whether the clarification herein referred to has been provided;

(c) Summarize separately the nature of each clarification which has been provided for vibration, consolidation, and/or densification, respectively;

(d) Identify the documents containing the clarification referred to in the answer to the preceding paragraph of this interrogatory, and the effective date of each clarification.

ANSWER. It is assumed in the answer given that the NRC document referred to is Report #50-483/77-07, not #50-4831 77-07.

(a) Procedural controls were in existence in WP-109, Revision 6, at the time NRC Report No. 50-483/77-06, page 3, identified this item. WP-109, Revision 6, incorporated by reference Section 9 and Section 10 c Bechtel Specification 10466-C103(Q) (Technical Specification for Forming, Placing, Finishing and Curing Concrete). This incorporation by reference adequately implemented the specification requirements and was therefore in accordance with 10 C.F.R. 50, Appendix B, Criteria V.

(b) Yes.

(c) The clarification provided was the adding of the word "vibration" or "consolidation" in the section that referenced Bechtel Technical Specification 10466-C103(Q). In Daniel International WP-109, this was done by adding a Note to Section 4.2.3. In Daniel International QCP-109, this was done by stating "verify concrete is thoroughly consolidated by suitable means during placement... as per Reference 2.13 (ACI 309 - Recommended Practice for Consolidation of Concrete) and Reference 2.9 (Bechtel Specification 10466-C103(Q))".

(d) Clarification is contained in the following documents:

(1) Daniel International Corporation WP-109; 10/24/80: Procedure for Concrete Placement, Grouting and Post-Pour, prepared by Daniel International Corporation Civil Engineering Department.

(2) Daniel International Corporation QCP-109; 3/4/81: Procedure for Inspection of Concrete Placement, Grouting and Post-Pour, prepared by Daniel International Corporation Quality Control (Civil) Department.

Note: Dates given above are dates of current revisions. The effective date of the clarification, as stated in NRC Report No. 50-483/77-09, p.3 is 11/3/77 for WP-109, Revision 7, and 11/7/77 for QCP-109, Revision 9.

INTERROGATORY NO. 42. Did repairs of the honeycombing in the tendon access gallery conform to the repair procedure suggested by Daniel International, as described in NCR 2-0856-C-A, and if the answer is negative:

(a) State the nature of deviations from the suggested procedure;

(b) State the reason(s) for deviations from the suggested procedure;

(c) Identify all documents relating to repairs and summarize their contents.

ANSWER. The Bechtel-approved repair procedure for the tendon access gallery varied from the repair procedure suggested by DIC in some instances.

(a) See NCR 2-0856-C-A, produced in response to document request no. 31, "Comments on Daniel Intl. NCR 2-856-C-A."

(b) Due to the uniqueness of this repair (overhead pressure grouting) Bechtel imposed supplementary requirements.

(c) The documents that relate to repairs are as follows:

(1) NCR 2-0856-C-A, produced in response to document request no. 31.

(2) WJE No. 77401, produced in response to document request no. 32.

(3) Grout Placement Cards for Pours #2C221Z07 (9-26-78), 2C221Z07-C (9-28-78), 2C221Z07-1 (9-29-78) and Drypack Placement Card for 2C221Z07-B (12-7-78); placed on dates shown in parentheses; these are the documents required by Procedure in order to place safety-related concrete, grout, or drypack.

(4) PCE-488; 5/12/77; Describes the types of repair necessary in the Tendon Access Gallery and the methods to be used in their repair; prepared by W. H. Bunt, Project Civil Engineer.

(5) PCE-531; 6/21/77; Request that repairs greater than 6" deep or where the chipped portion extends past the back of the trumplate be delayed, prepared by W. H. Bunt, Project Civil Engineer.

INTERROGATORY NO. 43. Regarding NRC Report No. 50-483/77-06, p. 22, state whether <u>repair</u> of the honeycombing therein referred to was hampered by limited mobility of work crews due to the trumplate wall dowels, the main steel, rebar supports and form ties.

ANSWER. Trumplates, wall dowels, the main steel, rebar supports and form ties did <u>not</u> hamper the mobility of work

crews making the repairs to the honeycombing referred to in NRC Report No. 50-483/77-06, p. 22.

INTERROGATORY NO. 44. If the answer to the preceding interrogatory is in the negative, describe the repair procedures utilized and the reason(s) why work crew and equipment mobility was not an impediment in these repairs.

ANSWER. The repair procedure utilized is as outlined in the Bechtel approved disposition to Nonconformance Report 2-0856-C-A. Mobility of work crews and equipment was not a factor in the repair of the honeycombing as the repair of the honeycombing took place from the Tendon Access Gallery. The trumplates, wall dowels, main steel, rebar supports and form ties are embedded in the concrete which forms the ceiling above the Tendon Access Gallery.

INTERROGATORY NO. 45. Regarding the reference to a "chipping operation" in NRC Report No. 59-483/77-06, p. 22, describe the chipping operation in the tendon access gallery, the chipping tool(s) used and the reason why "minor rebar damage" resulted from the operation.

ANSWER. The "chipping operation" referred to in NRC Report No. 59-483/77-06, p. 22 is a normal construction method used to remove unsound concrete. The chipping operation involves the use of a chipping hammer, which is a steel chisel driven by compressed air. The chipping hammer is not able to be precisely controlled such that "minor rebar damage" (<u>i.e.</u>, nicks) can occur when the bit of the chipping hammer contacts the reinforcing steel. Often a piece of concrete will fracture and immediately expose the embedded rebar with the bit of the chipping hammer in close proximity to the reinforcing steel.

INTERROGATORY NO. 46. Regarding the statement in NRC Report No. 50-483/77-07, p. 1[illegible], that "dry-pack grout was not being tested as required. . . because. . . Specification C-191. . . failed to include such a test," state whether the untested dry-pack grout herein referred to, or the repairs involving this grout, have been tested subsequent to NRC Report No. 50-483/77-07.

ANSWER. Neither were tested subsequent to NRC Report No. 50-483/77-07.

INTERROGATORY NO. 47. If the answer to the preceding interrogatory is in the affirmative, identify all documents which refer to test of the dry-pack grout, and summarize separately the contents of each such document. If the answer is in the negative, state the reason why no tests have been performed.

ANSWER. The drypacks referred to in the preceding interrogatory were identified as nonconforming on Nonconformance Report No. 2-1176-C-A. Bechtel Power Corporation, in the approved disposition of Nonconformance Report No. 2-1176-C-A did not require any testing to be performed, as the subject repairs were deemed to be structurally insignificant.

INTERROGATORY NO. 48. Does Union Electric believe that after repairs of voids in the tendon access gallery, the condition of the base slab has no adverse safety implications?

ANSWER. Yes.

INTERROGATORY NO. 49. State the bases for Union Electric's conclusion as to the preceding interrogatory.

ANSWER. The repair of the voids in the tendon access gallery results in a condition at the repaired areas which is at least as good as the original design requirement [illegible]is is

the result of (1) using material for repair which equalled or exceeded the quality and strength requirements of the original design and (2) utilizing procedures that provide for proper placement and bonding. Although all the defective areas were found by visual observations, soniscope methods were employed on a large sample of additional trumplates to provide added assurance that hidden defects were not a concern.

INTERROGATORY NO. 50. Explain if the word "trumplate" as used in NRC Report 50-483/77-06, p. 21 is synonymous with "bearing plate" as used in Figure 3.8-15 FSAR-SNUPPS. If not, define "trumplate."

ANSWER. The word "trumplate" refers to the combined trumpet and bearing plate which are welded together to form a unit.

INTERROGATORY NO. 51. State the tensile force in the vertical tendons of the Reactor Building:

(a) Under normal operating conditions;

(b) Under the 60 psig "design accident pressure load" condition.

ANSWER. (a) and (b) The tensile force in the vertical tendons of the Reactor Building under normal operating conditions is essentially the same as the tensile force under the 60 psig "design accident pressure load" conditions. Pressurization of the containment results in a small growth of the containment due to expansion. The increase of force in the tendons due to this growth is insignificant and therefore not calculated.

The tensile force in the tendons varies along the length of the tendon due to friction forces and varies with time due to prestressing losses such as elastic shortening of the concrete, shrinkage and creep of the concrete, and relaxation of the prestressing steel. The minimum force is at least 971,000 pounds and occurs at the apex of the dome. The force at the anchorages after stressing is at least 1,401,000 pounds and no greater than 1,601,000 pounds.

INTERROGATORY NO. 52. State the compression stress in the concrete above the bearing trumplates in the ceiling of the tendon access gallery:

(a) Under normal operating conditions;

(b) Under the 60 psig "design accident pressure load" condition.

ANSWER. (a) and (b) Adequacy of bearing plates and concrete to resist the bearing pressure is verified by testing. Bearing plates are tested to a load equal to the guaranteed ultimate tensile strength of the tendon, or approximately 2,000,000 pounds.

INTERROGATORY NO. 53. Regarding the report on the soniscopic study of the base slab by Wiss, Janey, Estner and Associates, Inc., dated August 1, 1977, state the following:

(a) The identity of the NRC inspector, referred to on page 13 of NRC Report #50-843/77-07, who inspected this report.

(b) The manufacturer, model number and specifications of the soniscope instrument;

(c) The minimum volume of a detectable air void, and the maximum depth in concrete at which a void is detectable;

(d) The qualifications and training of the personnel operating the device;

(e) Whether the transducer frequency utilized was the optimal for detecting the size voids anticipated. If the answer is in the affirmative, explain the basis for the answer.

(f) Whether the technique is capable of detecting an air void directly underneath a shallower void.

(g) Whether the soniscope instrument was used in the tendon access gallery or on the mat floor above the gallery.

(h) At what other specific locations of the base mat soniscope testing was performed.

ANSWER. (a) A. A. Verela.

(b) Manufacturer - James V-Scope; Model - C-4960; Specifications - see Table 53(b), attached hereto.

(c) While it is not possible to define a minimum detectable void volume nor a maximum depth in concrete at which a void is detectable, the manufacturer of the particular soniscope used at Callaway estimates that the minimum detectable void size on a 50 foot path length would be approximately a 24 inch diameter plane, and an 8 inch diameter plane on a 5 to 10 foot path length.

(d) See Attachment 53(d) (5 pages), attached hereto.

(e) The transducer frequency utilized was considered optimal for the interrelated requirements of detecting small voids while maintaining adequate signal quality during soniscope work on a thick concrete structure. Based upon observation of the voids in the tendon access gallery, the transducer frequency utilizied was deemed appropriate for detection of voids the size of those anticipated.

(f) Yes.

(g) Soniscope instrument was used in the tendon access gallery and on the base mat floor.

(h) Soniscope testing was also performed on the outside edge of the base mat.

INTERROGATORY NO. 54. State separately for each void or imperfection which has occurred in the Reactor Building dome, including but not necessarily limited to the seven areas of imperfection referred to in NRC Report No. 50-483/80-30, pp. 3-4, the depth, width, length, location and shape of each such imperfection.

ANSWER. A total of fourteen (14) voids and/or imperfections on the reactor dome are documented. The following is a breakdown of the above, giving Nonconformance Report numbers and attachments stating depth, width, length, location and shape of each imperfection.

(a) NCR's 2SN-3155-C, 2SN-3406-C and 2SN-3433-C. These imperfections are the seven (7) listed on NRC Report No. 50-483/80-30 pp. 3-4. Sheets 5 of 7 and 6 of 7 show additional chipping (per G. Goddard - Bechtel) after NRC disposition approval.

(b) NCR's 2SN-3630-C and 2SN-3652-C. Seven (7) total.

(c) This sketch shows the plan view for location reference of the fourteen (14) imperfections.

These documents will be produced at the same time and place as Applicant's document production.

INTERROGATORY NO. 55. State whether, in the opinion of Union Electric, the imperfections and voids in the concrete of the dome, described in NRC Report 50-483/80-30, pp. 3-4, are attributable to the same cause(s) as the voids in the base mat.

ANSWER. They are not.

INTERROGATORY NO. 56. State the bases for the conclusion in the preceding interrogatory.

ANSWER. The causes for the honeycombing in the base mat are stated in the response to interrogatory no. 34. The imperfections in the concrete dome were caused by intermittent, sporadic subsidence of the concrete during construction as a result of the concrete "boiling over" prior placements of concrete and to localized occurrences of unconsolidation in difficult to consolidate areas. The term "boiling over" refers to the downward movement of the concrete over the edge of the previous, lower placement of concrete due to the combined action of gravity and vibration of the concrete above. Of the two causes of the imperfections, the major cause is attributed to subsidence. The occurrence of localized unconsolidation is only incidental to the imperfections. Also, the dome pour did not contain the congested areas that were experienced in the pouring of the base mat.

INTERROGATORY NO. 57. State Union Electric's conclusion as to the cause(s) of honeycombing in the Reactor Building dome described in NRC Report No. 50-483/80-30, pp. 3-4, attibuting (sic) relative weight and probability to each cause if more than one cause is named.

ANSWER. The referenced NRC Report makes no reference to honeycombing in the Reactor Building dome. The report refers to concrete imperfections and unsound material. See response to interrogatory no. 56.

INTERROGATORY NO. 58. What is the basis in NCR 2SN-2790-C for specifying the imperfections in the concrete of the dome unreportable?

ANSWER. It is assumed that "unreportable" is used in this interrogatory to mean not reportable to the NRC under the criteria outlined in 10 C.F.R. 50.55(e). The areas of concrete imperfections identified in Nonconformance Report 2SN-2790-C were discovered upon removal of temporary, partially embedded material which was used to support formwork and equipment for the pour. The cause of these areas was thought to be inadequate consolidation due to limited accessability for vibration, which in combination with the extent of the imperfections identified at the time, did not represent a condition which was deemed reportable as defined by 10 C.F.R. 50.55(e).

INTERROGATORY NO. 59. Identify the design specifications governing the thickness of the exterior walls of Callaway Unit 1 Reactor Building dome.

ANSWER. The thickness of the exterior walls of the Callaway Unit 1 containment is determined by the design requirements included in BC-TOP-5-A, Revision 3, dated February 1975, including the documents referenced therein.

INTERROGATORY NO. 60. How thick are the exterior walls of the dome according to the design specifications listed in the answer to the preceding interrogatory.

ANSWER. Hemispherical Dome -- 3'-0";

Cylindrical Shell -- 4'-0".

INTERROGATORY NO. 61. In the opinion of Union Electric, how extensive can honeycombing be before the integrity of the containment building dome of Callaway Unit 1 is compromised?

ANSWER. The effect of honeycombing on the dome depends upon its location with respect to both meridional and vertical azimuth, its depth within the section, the shape of the honeycombed area and the intensity of the honeycombing. Considering the possible number of situations which could occur within each of these parameters and the multiplicity of parameters involved makes such an analysis impractical and unrealistic. No situation regarding honeycombing has been found on the dome to suggest a need for such analysis.

INTERROGATORY NO. 62. This interrogatory pertains to the following sentence in NRC Report No. 50-483/80-30, p. 4: "[L]icensee personnel attributed the occurrence of the imperfections to the <u>complex nature</u> of those portions of the dome slab where the imperfections had occurred". (emphasis added). State separately for each portion of the dome where an imperfection occurred the nature of the "complexity" here referred to.

ANSWER. The complexities referred to on page 4 of NRC Report 50-483/80-30 were the result of the extensive system of walkways and supports for concrete conveyance equipment required to distribute concrete over an area with the characteristics of the Reactor Dome.

Of the four areas referred to in this interrogatory which appeared to be unconsolidated concrete, these complexities were

thought to be partial obstruction - walkways for three of the areas and conveyor tower support legs for the fourth area.

INTERROGATORY NO. 63. Why have actions taken by Union Electric subsequent to the discovery of voids in the tendon access gallery, and designed to prevent recurrence of voids in concrete, proved inadequate to prevent voids in the dome?

ANSWER: The identified cause of the "voids" in the dome was entirely different from the cause of the "voids" in the tendon access gallery ceiling. The voids in the dome were not and could not have been prevented by the action to prevent recurrence prescribed for the tendon access gallery.

INTERROGATORY NO. 64. State any and all reasons Union Electric has to believe that imperfections in the concrete of the dome are limited to areas identified in NRC Report No. 50-483/80-30, pp. 3-4, and identify all documents and tests which form a basis for this conclusion.

ANSWER. The imperfections in the concrete of the dome are not necessarily limited to the areas identified in NRC Report No. 50-483/80-30, pp. 3-4. The <u>Final Report of Containment Dome Concrete Imperfections at Callaway Unit 1</u> (presently undergoing further revision and incorporation of additional data) demonstrates that although the imperfections may occur sporadically at other locations, they are of a minor nature and do not impair the structural integrity of the dome concrete. See document produced in response to document request no. 38.

INTERROGATORY NO. 65. By what testing methods did Union Electric determine the extent of imperfections in the concrete of the dome.

ANSWER. The extent of imperfections in the dome concrete was determined by personal observation and inspection,

selective excavation of concrete, boroscopic examination, microseismic examination, and nuclear densometer testing.

INTERROGATORY NO. 66. State whether ice was used in lieu of water in the Reactor Building dome concrete mix. If so, provide the following additional information:

(a) Explain why ice was used;

(b) Describe how ice was used;

(c) State the duration of the pour;

(d) State during which hours ice was used.

(e) State whether the use of ice in lieu of water in concrete mix violates any procedures, regulations, or requirements applicable to construction at Callaway Unit 1?

(f) Identify all procedures, regulations, or requirements which form a basis for the answer to paragraph (e) of this interrogatory, and summarize separately the content of each item listed.

ANSWER. Yes.

(a) Ice was used as required to reduce concrete temperatures to those required by section C paragraph 10.1.12. Bechtel technical specification 10466-C-101.

(b) Ice was batched with the other ingredients at the batch plant when the concrete was mixed.

(c) The pour lasted approximately 20 hours.

(d) Ice was used during the entire pour, or from 22:17 hours on 07-21-80 to 18:10 hours on 07-22-80.

(e) It does not.

(f) (1) 10466-C-101 (Q); 09-27-79; Technical Specification for contract for providing onsite batch plant and standarized Nuclear Unit Power Plant System (SNUPPS); section 10.1.2 Hot Weather concreting describes methods to be used when placing concrete in hot weather and allowable concrete temperatures; prepared by the Bechtel Corporation.

(2) 10466-C-103 (Q) produced in response to document request no. 22.

(3) ACI 305-72: 1972; Recommended Practice for Hot Weather Curing; ACI Publication defining Hot Weather, the effects of Hot Weather on concrete, and the steps necessary to produce Quality Concrete in periods of Hot Weather; prepared by American Concrete Institute (ACI) Committee 305.

(4) DIC Work Procedure WP-109; 10-24-80; Concrete Placement, Grouting and Post-Pour; Daniel Work Procedures that outlines, in section 4.0 the steps to be taken during Hot Weather concreting; prepared by DIC Civil Engineering Department.

INTERROGATORY NO. 67. This interrogatory applies to the following sentence in NRC Report No. 50-483/80-30, p. 5: "The licensee has committed to and undertaken <u>actions</u> to address the reactor dome concrete imperfection issue." (emphasis added).

(a) State separately the nature of each action referred to;

(b) State the date of each action undertaken or anticipated;

(c) State the extent to which each action listed in the answer to paragraph (a) of this interrogatory has resolved the dome concrete imperfection issue.

ANSWER. (a) and (b)

(1) Visual inspections of the excavated areas by Bechtel personnel from the Gaithersburg, Maryland office. (12/10/80 and 12/17/80).

(2) Interviews by Bechtel personnel from the Gaithersburg, Maryland office with the constructor's field engineers, a quality control inspector, the concrete superintendent and numerous craftsmen involved. (12/10/80 and 12/17/80).

(3) A study of the videotape of the dome placement. (12/10/80 and 12/17/80).

(4) Nuclear Density Meter Testing. (12/30/80 to 1/15/81).

(5) Boroscopic holes. (12/30/80 to 1/15/81).

(6) Microseismic Testing. (1/22/81 to 2/12/81).

(7) Excavation of two (2) test areas. (2/21/81).

(c) See documents produced in response to document request no. 38.

INTERROGATORY NO. 68. Regarding NRC Report No. 50-483/80-27, p. 21, state the cause of "flaking" on a concrete repair therein referred to.

ANSWER. The "flaking" of the concrete repair referred to in NRC Report No. 50-483/80-27, p. 21 was caused by the overlapping of a cosmetic repair onto the adjacent sound concrete. Since the adjacent sound concrete did not receive or require surface preparation and the overlapping portion was extremely thin, no bonding occurred between the surface of the sound concrete and the overflow from the repairs.

INTERROGATORY NO. 69. State whether the matter of flaking concrete, referred to in the preceding interrogatory, has been closed by subsequent inspection, and if the answer is affirmative, identify and state the number of the closing Nuclear Regulatory Commission Report.

ANSWER. To the best of Applicant's knowledge it has not been closed.

INTERROGATORY NO. 70. Identify by number, date and relevant pages all documents which pertain, in whole or in part, to nonconformance with concrete cover requirements through the fifth lift of the Reactor Building and as to each report listed, further state:

(a) the nature of the nonconformance;

(b) the location of the nonconformance;

(c) the date on which conformance was achieved or explain other disposition of the nonconformance;

(d) identify documents which verify conformance.

ANSWER. See NCR 2-2055-C-A and NCR 2SN-2007-C produced in response to document request nos. 46 and 47. NCR 2-0971-C-A, NCR 2-2229-C-A and NCR 2-2879-C-A are also being provided in response to this interrogatory and will be produced at the same time and place as Applicant's document production.

INTERROGATORY NO. 71. State the date on which Union Electric first communicated with the NRC Staff regarding the NRC interpretation of concrete cover requirements with regard to the Reactor Building wall at Callaway Unit 1, and further describe in detail the communication.

ANSWER. On January 5, 1978, the NRC was conducting a special, announced investigation into allegations regarding, among other things, improper concrete cover for reinforcement. Concrete cover requirements were interpreted differently by the NRC and Union Electric and its contractors.

The NRC made a verbal request that Union Electric chip two areas down to the depth of the minimum required concrete cover for the areas, to verify that adequate concrete cover is present.

Union Electric believed there was no legitimate reason for the chipping, and stated that no action would be taken without a formal written request from the NRC. Since the issue was not of safety significance, Union Electric was advised that a written request for the chipping would not be made.

INTERROGATORY NO. 72. This interrogatory pertains to the following sentence in NRC Report 50-483/77-11, p. 4: "[At 340 degrees azimuth of the third lift of the Reactor Building] concrete cover was less than that required by NRC interpretation of the <u>concrete cover requirements</u>, but within the concrete cover requirements as interpreted by the licensee and contractors." (emphasis added):

(a) Describe the relationship and location of the reinforcing steel and concrete cover in dispute;

(b) Identify all documents setting forth the "concrete cover requirements" referred to in the above sentence, and summarize separately the contents of each document;

(c) Identify all documents setting forth the NRC's interpretation of the concrete cover requirements referred to in the above sentence, and summarize separately the contents of each document;

(d) Identify all documents setting forth the licensee's and contractors' interpretation of the above-mentioned concrete cover requirements, and summarize separately the contents of each document.

ANSWER. (a) The design location of the reinforcing steel and concrete cover are shown in Figure 72(a) sheets 1 and 2 attached hereto. These figures show the relationship between the outside face reinforcing steel and the cover.

(b) The concrete cover requirements for construction purposes are given in Bechtel Specification No. 10466-C112(Q), Revision 9, and ACI 318-71.

(c) The NRC's interpretation of the concrete cover requirements is given in the meeting minutes attached to NRC Report No. 50-483/78-01.

(d) The documents setting forth the licensee's and contractors' interpretation of concrete cover requirements are given in the meeting minutes attached to NRC Report No. 50-483/78-01 (produced in response to document request no. 41), and in Enclosure B to BLSE 5526. A summary of the technical content of Enclosure B to BLSE 5526 will be produced at the same time and place as Applicant's document production.

INTERROGATORY NO. 73. Regarding the conflict between the NRC's interpretation of concrete cover requirements at 340 degrees azimuth and the licensee/contractors' interpretation, as indicated in NRC Report No. 50-483/77-11, p. 4, state which interpretation prevailed at 340 degrees azimuth.

ANSWER The interpretation of the licensee and contractors prevailed at azimuth 340 degrees on the fourth lift.

INTERROGATORY NO. 74. This interrogatory applies to the following sentence in NRC Report 50-483/77-11, p. 4: "This matter [of concrete cover requirements] will be resolved by the sixth lift of the reactor containment wall." State:

(a) The estimated additional construction costs which would have been incurred in adhering to the NRC interpretation of maximum and minimum concrete cover requirements starting with the third lift had such adherence been required as of the date of NRC Report No. 50-483/77-11;

(b) The repair process which would have been involved in adhering to the NRC interpretation of concrete cover requirements starting with the third lift, had such adherence been required as of the date of NRC Report No. 50-483/77-11;

(c) The estimated additional construction costs which would have been incurred in changing the first and second lifts to conform to the NRC interpretation.

ANSWER. (a) The cost of adhering to the NRC interpretation of minimum and maximum concrete cover starting with the third lift has not been estimated.

(b) On February 24, 1978, which was the date of NRC Report No. 50-483/77-11, both the third and fourth lifts of the Callaway Unit #1 Reactor Building exterior walls had been placed. Further, it is assumed that "starting with the third lift" means at the bottom of the third lift as "By the sixth lift" meant by the bottom of the sixth lift in NRC Report No. 50-483/77-11, p. 4. To attain a 2" absolute minimum cover for the third lift, it would have been necessary to remove concrete and rework reinforcing steel in the preceding lift.

(c) The additional cost of adhering to the NRC interpretation has not been estimated.

INTERROGATORY NO. 75. This interrogatory applies to the following sentence in NRC Report 50-483/77-11, p. 10: "Bechtel Power Corporation personnel repeated that their interpretation of the cover requirements was that the two-inch cover requirement can be reduced to an absolute minimum of an inch and one third per a provision of the specifications which allows reduction of the specified cover by one-third". (emphasis added):

(a) Identify the provision which allows reduction of the specified cover by one-third, and summarize its contents;

(b) State whether Union Electric believes the one-third reduction can be utilized without adverse safety implications, and the basis for Union Electric's belief.

ANSWER. (a) The one-third reduction in concrete cover was given by Section 7.10 of Bechtel Specification No. 10466-C112(Q), Revision 9, which referenced ACI 318-71 Section 7.3.2. See Section 7.10 of Revision 10 of Bechtel Specification No. 10466-C-112(Q) produced in response to document request no. 45 which did not change from Revision 9.

(b) Union Electric does believe that the one- third reduction can be utilized without adverse safety implications. In reinforced concrete members the concrete cover for reinforcing steel serves two specific purposes: (1) to protect the steel from corrosion in an adverse environment; and (2) to provide adequate bond for rebar development. Corrosion protection over a long period of time is provided by controlling cracking through the concrete cover. Crack control is within the scope of ACI Committee 224, which has stated "the

cracking mechanism in two-way action slabs and plates is controlled primarily by the steel stress level and the spacing of the reinforcement in the two perpendicular directions and only to a small extent by the magnitude of the concrete cover."

The reduced cover actually serves to reduce the potential crack width, according to the findings of ACI Committee 224, Control of Cracking in Concrete Structures - ACI Committee 224 Report, ACI Journal, December, 1972.

The second consideration in regard to concrete cover is to assure that the reinforcing steel near the surface has sufficient cover so as not to reduce the load-carrying capacity of the reinforcing steel. A discussion of this subject was presented in the ACI Journal, Orangun, O.C. Jirsa, J.O., Brown, J.E. ACI Journal March, 1977. The empirical expressions formulated from that work, together with the appropriate rebar spacing and development lengths provided in the shell, show that the minimum concrete cover is acceptable.

INTERROGATORY NO. 76. Provide identifying information regarding "a draft Code case" involving the matter of concrete cover, as referred to on page 10 of NRC Report No. 50-483/77-11; and state whether the Code change under consideration was implemented.

ANSWER. The "draft code case" referenced apparently pertains to a question submitted by Brown & Root, Inc. to the ACI/ASME Code Committee in a letter dated June 22, 1976. A draft response was prepared by a working group committee and

was in the possession of the NRC on January 23, 1978. Applicant has been unable to confirm that Brown & Root has since received a reply. Code Interpretation III - 2-77-13, responds to a similar question. This interpretation was issued by ASME on October 21, 1977 and published in September, 1978. No code change has yet been implemented.

INTERROGATORY NO. 77. This interrogatory applies to the following sentence in NRC Report No. 50-483/77-11, p. 11: "[A] two-inch minimum concrete cover will be required for the sixth and subsequent lifts, utilizing the fifth lift as a transition area." Explain what is meant by utilizing the fifth lift as a transition area.

ANSWER. "Utilizing the fifth lift as a transition area" as stated in NRC Report No. 50-483/77-11, p. 11 means the reinforcing steel in the fifth lift would be gradually sloped radially inward toward the center of the containment building to allow an absolute minimum concrete cover of two inches (2") by the bottom of the sixth lift.

INTERROGATORY NO. 78. NRC Report No. 50-483/77-11, indicates that "Union Electric is evaluating" the requirement that the company comply with concrete cover requirements at the sixth and subsequent lifts:

(a) State the outcome of Union Electric's evaluation;

(b) Identify documents which pertain to Union Electric's evaluation, and summarize separately the contents of each document.

ANSWER. (a) Compliance with the NRC's interpretation of the concrete cover requirements was agreed to by Union Electric starting at the sixth lift. Design documents were revised to reflect this decision.

(b) DLUC 2551 dated February 24, 1978. Summary: The conclusion to that evaluation was that it is possible to maintain two inches of cover in the shell starting with sixth lift.

INTERROGATORY NO. 79. Identify, and provide report numbers and dates, for all nonconformance reports pertaining, in whole or in part, to concrete placement in the third lift area, including but not necessarily limited to the 23 reports mentioned in NRC Report No. 50-483/77-10, at p. 19, and summarize separately the contents of each report.

ANSWER. See documents produced in response to document request no. 47. See further, ten (10) NCR's also being provided in response to this interrogatory which will be produced at the same time and place as Applicant's document production.

INTERROGATORY NO. 80. Insofar as NRC Report No. 50-483/77-10 mentions 23 nonconformance reports on the third lift concrete pour and describes this as "an unusually large number," explain any and all factors which could be considered to have contributed to such a large number of reports.

ANSWER. Factors which contributed to the number of Nonconformance Reports for the Reactor Building exterior wall, third lift (Pour 2C231W03) are as follows:

(1) Extreme congestion in the area due to the large amounts of #11, #14 and #18 bars present in the section along with tendon sheathing, the auxiliary access hatch, liner plate stiffener embeds and liner plate penetrations.

(2) The proximity of the third lift to the penetrations for main steam, main feed water and blowdown lines and

the resultant interferences of these penetrations with the bar tails.

Note, however, that of the 23 NCRs mentioned in NRC Report No. 50-483/77-10, one was superseded. Of the remaining 22 NCRs, 16 were the result of minor conflicts between the detail and design drawings due to the complex geometry of the reinforcing steel, and the inherent difficulty in detailing.

The remaining six (6) NCRs are broken down as follows: two (2) cases where reinforcing steel was incorrectly installed (one report per bar); a broken welded stud requiring repair; a sheathing drain not installed; a bent rebar; and a damaged rebar.

INTERROGATORY NO. 81. Identify the ten nonconformance reports pertaining to the third lift of the Reactor Building wall which were still outstanding the evening of November 21, 1977.

ANSWER. Only nine NCR's were found to be outstanding on November 21, 1977. These NCR's are as follows:

| | |
|---|---|
| 2-1470-C-D | 2-1595-C-D |
| 2-1511-C-A | 2-1605-C-B |
| 2-1532-C-B | 2-1613-C-A |
| 2-1582-C-A | 2-1634-C-A |
| | 2-1637-C-D |

INTERROGATORY NO. 82. This interrogatory applies to information in item 3. a. (8), page 8-9 of NRC Report No. 50-483/78-01, where "several areas" of concrete cover are mentioned as "less than two inches as specified on placement drawings."

(a) Describe the location and size of each of the "several areas" and describe the location of rebar within each area;

(b) Summarize the discussion at the January 23, 1978 meeting between Union Electric and the NRC in Bethesda, Maryland, as to the areas of nonconforming concrete cover described in item 3.a.(8);

(c) State the decision reached at the January 23, 1978 meeting as to whether the two inch cover required by Bechtel Topical Report BC-TOP-5, Section CC-3533.1 of Appendix C could be reduced by one-third pursuant to specification No. C-112.

ANSWER. (a) No information is available, since design specifications were not violated.

(b) See minutes of January 23, 1978 meeting, produced in response to document request no. 41. See also response to interrogatory no. 75(b).

(c) See page 6 of Enclosure 1 to NRC Report No. 50-483/78-01, produced in response to document request no. 41.

INTERROGATORY NO. 83. This interrogatory applies to information in item 3.a.(7), page 8, of NRC Report No. 50-483/78-01, involving "several areas" of "concrete cover. . . of 12 to 13 inches which appeared to be more than permitted."

(a) Describe the location and size of the "several areas" herein referred to;

(b) Summarize the discussion in the January 23, 1978 meeting between Union Electric and the NRC in Bethesda, Maryland, as to the areas of nonconformance described in item 3.a.(7);

(c) State the decision which was reached in the January 23, 1978 meeting as to the areas in which concrete cover exceeded the maximum allowed by BC-TOP-5, Section CC-3534 of Appendix C.

ANSWER. (a) Applicant's best information is that the "several areas" referenced by the NRC are in the area of the

electrical penetrations. These areas are currently identified in FSAR Section 3.8.1.6.2.3.

(b) See Enclosure 1 to NRC Report No. 50-483/78-01, produced in response to document request no. 41.

(c) See Enclosure 1 to NRC report No. 50-483/78-01, produced in response to document request no. 41.

INTERROGATORY NO. 84. Regarding variations beneath the sixth lift of the Reactor Building from the two-inch concrete cover requirement established by Section CC-3533.1 of Appendix C to BC-TOP-5 for #6 through #18 reinforcing steel, state the following:

(a) The number of variations from Section CC-3533.1 of Appendix C to BC-TOP-5 for #6 through #18 reinforcing steel, beneath the sixth lift;

(b) The location of each such variation;

(c) The safety implications of such variations when considered cumulatively.

ANSWER. (a) No information is available since the requirements of Bechtel Specification No. 10466-C112(Q) in effect at the time of the pour were met.

(b) See answer to interrogatory no. 84(a) above.

(c) See answer to interrogatory no. 75(b) above.

INTERROGATORY NO. 85. This interrogatory applies to a 1 1/2 inch placement tolerance described in NRC Report No. 50-483/78-01, item 3.b.(14), p. 11, and to NCR 2-2055-C-A which was "dispositioned 'use as is' by Bechtel prior to concrete placement":

(a) State the basis for Bechtel's "use as is" disposition in NCR 2-2055-C-A;

(b) State Union Electric's conclusion as to the safety implications of the "use as is" disposition of NCR 2-2055-C-A, and the basis for this conclusion.

ANSWER. (a) The "use as is" disposition of NCR 2-2055-C-A does not adversely affect the safety of the plant. The purpose of the crack control provisions in Topical Report BC-TOP-5a is primarily intended to be applicable for the main body of the shell "to control <u>general</u> face cracking" (emphasis added). Over local areas however, a strict adherence to the placing provision is neither intended nor technically required. None of the areas affected by increased reinforcing cover are exposed to outside environmental conditions. Rather, the exterior wall of the reactor building shell in these areas is enclosed within the environment of the auxiliary building.

Although the crack control criterion of ACI Committee 224 Report is not a requirement for the reactor building shell, it is satisfied. Using the analytical methods included in the report for the local areas under question and considering the case where field tolerances increase the cover beyond that required by the Topical Report, crack width is again of no consequence.

Considering the fact that the building is prestressed, the structural integrity of the containment shell will not be compromised where the maximum cover limits are exceeded in these local areas.

(b) Applicant has reviewed the Bechtel "use as is" disposition and agrees with it. See answer to interrogatory no. 85(a).

INTERROGATORY NO. 86. State whether Union Electric, Bechtel Power Corporation, Daniel International Corporation, or SNUPPS, alone or in concert, have objected to the requirement at the Callaway Plant of a minimum concrete cover of two inches over reinforcing steel on the outer face of the reactor containment with no placement tolerance on that minimum dimension at or above the sixth lift, and if the answer is affirmative, further state:

(a) The bases for objection to the requirement;

(b) The details of any alternative requirement proposed by any of the above parties;

(c) The reasons set forth by the proposing party for adoption of any alternative requirement named in answer to the preceding paragraph of this interrogatory.

ANSWER. No, not after receipt of the minutes from the January 23, 1978, NRC meeting.

INTERROGATORY NO. 87. State whether Union Electric, Bechtel Power Corporation, Daniel International Corporation, or SNUPPS, alone or in concert, have objected to the requirement at the Callaway Plant of a maximum concrete cover on face reinforcing steel of ten inches at or above the sixth lift, and if the answer is affirmative, further state:

(a) The bases for objection to the requirement;

(b) The details of any alternative requirement proposed by any of the above parties;

(c) The reasons set forth by the proposing party for adoption of any alternative requirement named in answer to the preceding paragraph of this interrogatory.

ANSWER. No, not after receipt of the minutes from the January 23, 1978, NRC meeting.

INTERROGATORY NO. 88. State to what extent Union Electric believes that a reduction in the two-inch minimum concrete cover on reinforcing steel in the lower five lifts of the Callaway Reactor Building exterior wall may have reduced or compromised the following properties of the structural system:

(a) Protection against corrosion of the steel if exposed to weather;

(b) Protection against excessive heat;

(c) Assurance of adequate bond for rebar development.

ANSWER. (a) See answer to interrogatory no. 75(b).

(b) The reduction of the specified cover on reinforcing steel in the first five lifts will not adversely affect the performance of the structure within its specified design temperature requirements.

(c) See answer to interrogatory no. 75(b).

INTERROGATORY NO. 89. State to what extent Union Electric believes that exceeding the ten-inch maximum concrete cover on reinforcing steel below the sixth lift of the Reactor Building may have reduced or compromised the structural system's ability to control cracking.

ANSWER. The ten-inch maximum concrete cover on reinforcing steel below the sixth lift of the reactor building has been exceeded only in the local areas surrounding the electrical penetrations. Exceeding the maximum cover requirements in these local areas will not compromise the structural system's ability to control cracking such that the structure will be prevented from performing its intended design function.

INTERROGATORY NO. 90. State whether Union Electric made any changes in its minimum and maximum concrete cover requirements or procedures for reinforcing steel in outside faces of buildings other than the Reactor Building following the NRC meeting of January 23, 1978. If the answer is affirmative, cite specific changes and identify relevant documents.

ANSWER. Applicant has made no such changes.

INTERROGATORY NO. 91. For the 23 NCRs regarding the third lift referred to in NRC Report No. 50-483/77-10, page 19, state on what day and what hour each was resolved or closed out.

ANSWER. It is not possible to determine on what hour each NCR was closed, since records of this nature are not kept by DIC Engineering or DIC Quality Control. Specific dates on which the NCR's were closed can be determined from the NCR's themselves, all of which have been produced in response to document request no. 47.

INTERROGATORY NO. 92. State separately for each of the first six lifts of the Reactor Building exterior wall how many days were spent installing the reinforcing steel.

ANSWER.

Pour #1 - 39 days

Pour #2 - 32 days

Pour #3 - 33 days

Pour #4 - 50 days

Pour #5 - 42 days

Pour #6 - 44 days

INTERROGATORY NO. 93. With respect to NRC Report No. 50-483/80-10 and the allegations, investigations, and inspections upon which it is based, provide the following information:

(a) Who was the vendor of the spool piece which was the subject of NRC Report No. 50-483/80-10?

(b) Who manufactured and supplied the pipe to the vendor?

(c) When was the pipe manufactured?

ANSWER. (a) DRAVO Corporation, Marietta, Ohio.

(b) Armco Steel Corporation.

(c) October 28, 1977 (Date material was certified on the Certified Material Test Report).

INTERROGATORY NO. 94. Are fusion welded tubular products, other than SA 312, intended for use in safety-related systems at Callaway? (i.e., Pipe made in accordance with SA 249, SA 333, SA 334 and fittings made in accordance with SA 403.)

ANSWER. At Callaway, there are no safety-related pipes made in accordance with SA 249, SA 333, or SA 334. In dealing with fittings made in accordance with SA 403, the fittings may be fusion welded or forged.

INTERROGATORY NO. 95. Were defective welds found in preassembled piping formations manufactured by Gulf & Western (G&W), fabricated, delivered, and in some cases installed at Callaway Unit One as indicated in SNUPPS letter SLNRC 79-20 dated Nov. 29, 1979 and the report referenced therein? If affirmative, state the following:

(a) Who made the visual examination of the preassembly Formation A-9-111 in March, 1979 at the Wolf Creek job site?

(b) Was this a required examination?

(c) How many G&W preassembled piping formations were delivered to the Callaway Plant prior to November 2, 1979?

(d) State how many such formations were installed at the Callaway Plant prior to November 2, 1979, describe individually each formation installed and state where in the plant (including identification of systems and service) each formation was installed.

(e) State whether any of the subject formations were inspected upon receipt at the Callaway Plant site. If so, state the number inspected, method(s) of inspection and by whom the inspection(s) were made;

(f) How many of the subject formations delivered to the Callaway Plant site were found to have defective welds?

(g) How many of the subject formations delivered to the Callaway Plant site were reworked?

ANSWER. Yes.

(a) Richard Cook, DIC Certified Level II Mechanical/Welding Inspector, Certified Lead NDE Inspector, and Acting QC Supervisor (as of March, 1979); Lead Mechanical/Welding Quality Inspector, Certified Lead NDE Inspector (currently).

(b) No.

(c) 31 safety-related formations.

(d) See table 95(d), attached hereto.

(e) A total of thirty-one (31) formations were received at the Callaway Project and all formations received receipt inspection. Visual inspections were conducted by DIC certified quality control inspectors in accordance with defined DIC procedures.

(f) Fifteen (15) safety-related formations.

(g) Twenty-seven (27) formations were reworked. The term "rework" as used in this response is defined as any work required to restore the formation to a condition acceptable to the governing code and/or specifications.

INTERROGATORY NO. 96. With regard to SNUPPS FSAR, Section 6.3.2.1, page 6.3-2, the statement, ". . . a minimum of three

accumulators. . .ensure adequate core cooling in the event of a design basis LOCA as to provide boration in the event of a steam or feedwater break accident," provide the following information:

(a) Does the SNUPPS design call for four accumulators?

(b) Should one of the four accumulators become inoperable, are the remaining three sufficient? If not, please explain the quoted statement.

ANSWER. (a) As shown on FSAR Figure 6.3-1, four accumulators are provided.

(b) The quoted statement from FSAR Section 6.3.2.1 is self-explanatory. A minimum of three of the four accumulators, plus the additional minimum ECCS equipment, ensure adequate core cooling in the event of a design basis LOCA or boration in the event of a steam or feedwater line break accident.

INTERROGATORY NO. 97. What is the sequence of use of the four accumulators during an event for which they would be used?

ANSWER. In the event of a LOCA that results in a significant reduction of reactor collant system pressure, the accumulators begin to inject borated water when the pressure falls below approximately 600 psi. Each accumulator injects borated water at approximately the same time since each is independently connected to one of the reactor coolant system loops (see FSAR Figure 6.3-1).

INTERROGATORY NO. 98. If the accumulators work in sequence, in the event of malfunction of one accumulator what is the delay time until a second accumulator would be brought into use?

ANSWER. Each accumulator is essentially independent of the others. Failure of one accumulator to inject its contents does not affect the other three or cause a delay in their operation.

INTERROGATORY NO. 99. What inspection procedure will be used to insure that the welds and pipes in the accumulator system do not weaken over time?

ANSWER. Welds in the accumulator system will be examined under a pre-service inspection program prior to operation and an in-service examination periodically during operation. Welds will be examined either by a combination of ultrasonic and surface examinations or surface examination only depending on system classification, pipe size, and wall thickness. These examinations are performed and evaluated in accordance with ASME Boiler and Pressure Vessel Code Section XI.

INTERROGATORY NO. 100. After discharge of accumulators during a design basis LOCA or a steam or feedwater break accident, when will the accumulators be refilled?

ANSWER. The accumulators are not required to be refilled after they are discharged during a design basis accident for which they are needed. They will be refilled at a convenient time after the transient and before return to normal plant operation.

INTERROGATORY NO. 101. What pressure is normally expected in the section of pipe coming from the accumulator to the Reactor Cooling System cold leg loop between the motorized control valves and the check valve when the plant is operating?

ANSWER. During the normal plant operation the section of piping between motor-operated valves EP HV8808A through D and

check valves EP 8956 A through D would be at a pressure approximately equal to that of the associated accumulator. As stated in FSAR Table 6.3-1, normal operating pressure of the accumulators is 650 psig.

INTERROGATORY NO. 102. If hafnium is used in the control rods instead of silver, indium and cadmium, what changes in the fission, activation, and corrosion products might be expected?

ANSWER. Because the control rods are clad with stainless steel, no change in corrosion products would be expected if hafnium is used instead of silver - indium - cadmium. No fission product changes would be expected. The activation level of hafnium would be lower than that of silver - indium - cadmium.

INTERROGATORY NO. 103. At what Westinghouse reactors has there been experience with hafnium control rods?

ANSWER. Hafnium has been used as control rod material at the following Westinghouse designed reactors:

1. Shippingport;
2. Yankee Rowe.

INTERROGATORY NO. 104. Has Union Electric or has anyone in its behalf made any calculations of releases of radioactive materials in gaseous and liquid effluents other than calculations based on the PWR-GALE Code-NUREG-0017 and/or Regulatory Guide 1.21 and/or Regulatory Guide 1.112? If so, identify all reports of such calculations by title and date, and state who presently has such report in his custody or possession.

ANSWER. No.

INTERROGATORY NO. 105. Explain the reasons for the change from an expected failure rate of 0.25 percent of the fuel rods (Final Environmental Statement-NUREG 75/011, p. 3-10) to 0.12 percent (FSAR-SNUPPS, Table 11.1 A-1).

ANSWER. The value of 0.12 percent failed fuel used in the SNUPPS FSAR for calculating radioactive effluents from the plant is in accordance with the NRC current guidance provided in NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code).

INTERROGATORY NO. 106. What will be the amount (curies per year) of tritium released each year in the liquid effluent? How is this figure derived?

(a) How much of this figure will be produced by fission?

(b) How much by activation?

(c) Of the amount produced by fission, describe fully the liquid pathway whereby the tritium will be released into liquid effluent.

ANSWER. As indicated in Table 11.1-2 of the FSAR, the annual tritium release from the plant in liquid effluents is expected to be 410 curies. This value was calculated in accordance with NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code).

(a) The methodology for calculating tritium release rates is described in Section 2.2.19 of NUREG-0017. This method employs an empirical relationship derived from data from operating plants that is a function of the reactor's thermal rating. The method does not identify specific tritium production mechanisms.

(b) See 106 (a).

(c) Since specific tritium production mechanisms are not identified, as discussed in (a) above, source-specific tritium pathways are also not identified. NUREG-0017 considers tritium released via the liquid pathway as part of the nonrecyclable waste streams for the boron recovery, clean waste and dirty waste systems and the turbine building floor drain discharge.

INTERROGATORY NO. 107. State the derivation of the estimate of 410 curies of tritium as the average release per year in the liquid effluent of a 1000-megawatt pressurized water reactor using zirconium-alloy-clad fuel rods. Include an account of the amount of boric acid estimated to be used per year in the reactor vessel.

ANSWER. The annual tritium release rate from the plant in liquid effluents of 410 curies per year, as indicated in FSAR Table 11.1-2, was calculated in accordance with NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code) which is the recommended method provided by the Nuclear Regulatory Commission in Regulatory Guide 1.112, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors.

Inasmuch as this method does not derive the expected tritium production and release on a mechanistic basis, an estimate of the amount of boric acid to be used per year in the reactor vessel is not available.

INTERROGATORY NO. 108. Has any estimate or study been prepared which concludes that the operation of the Callaway Plant or a plant like the Callaway Plant could produce in edible fish in the Missouri River (or any river) an amount of radionuclides that could be dangerous to the health of a person eating the fish? If so, identify the study or estimate by title, author and date and state who now has such report in his possession or custody.

ANSWER. Union Electric is unaware of any study similar to that described in this interrogatory.

INTERROGATORY NO. 109. Are there any noble gases at all such as xenon-127, xenon-133, krypton-81 or krypton-85 released from a pressurized water reactor such as Callaway Plant Unit One? If so, state which gases and the amount released in terms of curies per year in (a) the liquid effluent and (b) gaseous emissions.

ANSWER. FSAR Table 11.1-2 provides the calculated annual radioactive releases in liquid and gaseous effluents, including noble gases.

As indicated on sheet 1 of Table 11.1-2, no significant amount of noble gases is expected in liquid effluents. Sheet 2 of Table 11.1-2 indicates the isotopes and quantities of noble gases expected in gaseous effluents.

INTERROGATORY NO. 110. Will any of the radionuclides to be released from the Callaway Plant Unit One be released in particulate form as opposed to fully dissolved form? If so, please identify such radionuclides and the amount to be released in terms of curies per year:

(a) With the liquid effluent.

(b) To the atmosphere.

ANSWER.

(a) A number of the radionuclides in liquid effluents identified in FSAR Table 11.1-2 are

expected to occur in an insoluble form; however, the method used to calculate these expected values, NUREG-0017, does not specifically differentiate between soluble and insoluble isotopes in the liquid effluents.

(b) Of the gaseous effluents given in FSAR Table 11.1-2, the Mn-54, Fe-59, Co-58, Co-60, Sr-89, Cs-134 and CS-137 are expected to be in particulate form, as described in NUREG-0017.

INTERROGATORY NO. 111. Will the amount of tritium produced by the Callaway Plant per year increase as the plant gets older? If so, estimate the amount or rate of increase and state the bases for the estimate.

ANSWER. The amount of tritium assumed to be produced at the Callaway Plant, as given in FSAR Table 11.1-2, is derived by the method recommended by the Nuclear Regulatory Commission in NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). Section 2.2.19.1 thereof states, in part,

> The data obtained at Ginna (1970-1972) show that after three years of operation the plant tritium inventory had not reached steady state and that in 1973 approximately 50% of the tritium produced was released. For this reason, we consider that the data in Table 2-22 indicate a lower tritium production rate than would actually be experienced after steady state is attained, and consequently the apparent production rate has been doubled and a release rate obtained of 0.4 Ci/yr per MWt, which considers both liquid and vapor pathways.

Therefore, the initial tritium production rate is assumed to be the expected steady-state value and that value is considered to remain constant throughout the plant life.

INTERROGATORY NO. 112. In what way, if any, is cooling tower blowdown water treated prior to discharge? What is the treatment designed to accomplish and how is it performed?

ANSWER. Cooling tower blowdown water is not treated prior to discharge.

INTERROGATORY NO. 113. List all alpha emitting radionuclides that will be released in the liquid effluent and give the estimated single reactor amount per year of such emission in terms of curies per year.

ANSWER. None of the nuclides in liquid effluents identified in FSAR Table 11.1-2 which were calculated in accordance with NUREG-0017 is an alpha emitter.

INTERROGATORY NO. 114. State the following information regarding the liquid effluent for the Callaway Plant Unit One:

(a) How many gallons of liquid effluent will be released each day of reactor operation which contain radioactive isotopes including tritium and dissolved and entrained noble gasses.

(b) How much of the total liquid effluent containing such radioactive materials will be cooling tower blowdown water?

(c) How much will be domestic sanitary wastes?

(d) How much will be demineralized water system regenerant wastes?

(e) How much will be "process" or "processed" water other than demineralized water system regenerant wastes.

(f) What will be the source or sources of this other processed water?

ANSWER.

(a) Approximately 136,000 gallons per day.

(b) None of the amount given in (a).

(c) None of the amount given in (a).

(d) None of the amount given in (a).

(e) All water containing radioactive isotopes will be processed in the radwaste systems as required to meet the Technical Specifications.

(f) The assumed sources of the amount specified in (1) are listed in FSAR Tables 11.1A-2 and 11.1A-5.

INTERROGATORY NO. 115. To what extent has Union Electric assessed the impact of reduced flow of the Missouri River by reason of drought or ice jams or by reason of future diversion projects (e.g., irrigation or major industrial sites) with respect to the potential for dilution of the Callaway radioactive discharge?

ANSWER. Low flow considerations are discussed in SNUPPS FSAR Site Addendum Section 2.4.11.

INTERROGATORY NO. 116. In making projections of future Missouri River low flows in SNUPPS FSAR Callaway Site Addendum pp. 2.4-37 and 2.4-38, why was not the more conservative model developed by MRBCS used, rather than the model of the U.S. Corps of Engineers, especially since the latter predicts low flows greater than those already obtained at Hermann, Missouri? Do these models take into account the anticipated construction of a coal gasification plant at Yates, Missouri?

ANSWER. The so-called "more conservative" MRCBS model for projecting future Missouri River low flows, published in 1969, included all possible withdrawals that could be conceived of at the time the study was made, regardless of how feasible or viable the projected use might be. The best available projections of future Missouri River low flows have been computed by

Claire (1974) and Duscha (1974) at the Reservoir Control Center of the U.S. Army Corps of Engineers, Omaha district office. These were later updated to take into account withdrawals in the upper Missouri River Basin for conceivable coal gasification projects. Since the Corps of Engineers' projections removed the more speculative or impractical projected future withdrawals contained in the MRCBS model, the Corps' projections were considered more realistic and, therefore, were used.

The projected low flow at Hermann, which is slightly higher than the historic low flow at Hermann, is considered realistic in view of the greater control over flow possible by the Corps of Engineers now than when the historic low flow occurred.

References: Claire, M., Corps of Engineers, Reservoir Control Center, Missouri River Division, Department of the Army, personal communication (January 3, 1974).

Duscha, L. A., Chief, Corps of Engineers, Engineering Division of Missouri River Division, Department of the Army, Omaha, Nebraska, written communication (January 17, 1974).

INTERROGATORY NO. 117. What are the nature and degree of reliability of the high and low-level monitors in the Missouri River "provided to give sufficient notice to the main control room operators to allow an orderly reduction or shutdown of plant operation" (SNUPPS FSAR CALLAWAY Site Addendum p.

2.4-40)? How often are these monitors inspected, checked for obstructions, etc.?

ANSWER. The cited statement at p. 2.4-40 of SNUPPS FSAR CALLWAY Site Addendum is incorrect. There is no high level monitor, and the low-level monitor is not required for plant operations.

INTERROGATORY NO. 118. Do Corps regulations require a minimum Missouri River flow of 6000 cfs at Kansas City? What would the flow at Hermann, Missouri be under these minimum conditions?

ANSWER. The U.S. Army Corps of Engineers is required, in order to meet water quality requirements in the Missouri River, to maintain a minimum release of 6,000 cfs at river mile 811.1 at Gavins Point Dam. This results in a flow of 9,000 cfs at Kansas City at river mile 356 (Abraham, 1981). Since there are several major tributaries between Kansas City and Hermann at river mile 98 (e.g., the Lamine River, Grand River, Osage River, Cedar Creek, Gasconade River), the low flow at Hermann would exceed 9,000 cfs.

Reference: Abraham, C., Personal communication: U.S. Army Corps of Engineers, Omaha district office (June 19, 1981).

INTERROGATORY NO. 119. What is the flow of water required for radwaste dilution of tritium waste to dilute it below the recommended EPA levels of 20,000 pCi/l per year? Are the approximately 10,000 GPM currently allocated for radwaste dilution (SNUPPS FSAR Callaway Site Addendum Fig. 2.4-15) sufficient to achieve both tritium and other isotopic radwaste dilution to meet EPA/NRC maximum levels?

ANSWER. Approximately 10,000 gpm of flow in the Missouri River in addition to the 10,000 gpm plant dilution flow would be required to achieve a tritium concentration of 20,000 pCi/l during times when radioactive discharge to the river is occurring from both units. This can be compared to a current average river flow of about 30,000,000 gpm. Therefore about 0.03% of average river flow is required for dilution. This amount of dilution flow would meet all EPA and NRC regulations.

INTERROGATORY NO. 120. Considering that the intake and discharge pipes to and from the Callaway reactor are approximately 5 miles long, what precautions are being taken to assure minimum flow through the pipes?

ANSWER. There are no requirements for a constant minimum flow in either pipeline. There is a requirement for 5,000 gpm minimum flow in the discharge line when discharging radwaste effluent. Radwaste system outlet valve will not open unless atleast 5,000 gpm flow is provided.

INTERROGATORY NO. 121. What will be the amount (curies per year) of tritium released each year in gaseous emissions? How is this figure derived?

(a) How much of this figure will be produced by fission?

(b) How much by activation?

(c) Of the amount produced by fission, describe fully the pathway whereby the tritium is released to the environment.

ANSWER. As given in FSAR Table 11.1-2, the expected annual tritium release in gaseous effluents is 1,000 Ci. This value was derived in accordance with the guidance provided by

the Nuclear Regulatory Commission in NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). As discussed in response to Interrogatory 106, this method does not identify tritium source terms mechanistically, therefore the information requested by parts (a), (b) and (c) is not available.

INTERROGATORY NO. 122. State the dispersion characteristics of solid particulates which may be released from a pressurized water reactor such as Callaway Unit One during a short term accident.

ANSWER. The short-term atmospheric dispersion characteristics used for accident analysis at Callaway are provided in FSAR Table 15A-2.

INTERROGATORY NO. 123. Describe the model assumed in answering the previous interrogatory.

ANSWER. The model assumed in answering the previous interrogatory is described in SNUPPS FSAR CALLAWAY Site Addendum § 2.3.4.

INTERROGATORY NO. 124. Regarding SNUPPS FSAR, Section 12.2.2, explain the bases for using data on in-plant radioactive contamination obtained at the Fort Calhoun nuclear plant to estimate in-plant contamination at Callaway.

ANSWER. Data on in-plant radioactive contamination at Fort Calhoun is not used to estimate in-plant contaimination at Callaway.

INTERROGATORY NO. 125. This interrogatory pertains to Environmental Report Operating License Stage, Volume II, Section 6.1.3.2.1 Radiological Dispersion Models. Equations

[6.1.3-8], [6.1.3-9] and [6.1.3-10] predict infinite concentrations at the plume centerline if the wind speed at the 10-meter level is zero. State the ground-level relative concentrations at the plume centerline during a short-term accident at a time of no detectable air motion.

ANSWER. The discussion presented in the Environmental Report concerning the referenced equations states than when calm conditions occur, a positive value of wind speed equal to the highest starting speed of either wind sensor is substituted for the recorded value of wind speed. The ground-level relative concentrations are then determined as described in the discussion. The equations referenced do not predict relative concentrations when wind speed is zero, since in mathematical operations, division by zero is not defined.

INTERROGATORY NO. 126. Environmental Report, Page 6.1-23, last paragraph states, "For calm conditions, a wind speed is assigned equal to the vane or anemometer starting speed, whichever is higher". State the basis for assigning the higher starting speed as the wind speed rather than the lower.

ANSWER. In Regulatory Guide 1.XXX, it is stated that during calm conditions, a wind speed would be assigned equal to that of the anemometer or vane starting speed, whichever is higher. This procedure is repeated in Regulatory Guide 1.145, which was released for comment in August of 1979.

INTERROGATORY NO. 127. State whether the "starting speeds" referred to in the preceding interrogatory are the "threshold" speeds of Table 6.1-6, Environmental Report Operating License Stage Vol. II.

ANSWER. The terms "starting speeds" and "threshold speeds" have identical meanings when used as described in this interrogatory.

INTERROGATORY NO. 128. Explain the term "spatial variations in stability", as used in Environmental Report, Page 6.1-29, top line.

ANSWER. Hourly meteorological input to the PUFF model can include stability class indices from up to 30 arbitrarily located stations. Spatial variations in stability for a given hour will result when stations at different locations provide different stability class indices. Hourly stability class indices from the various stations are interpolated onto a uniform rectangular grid within the PUFF model using inverse distance weighting. For each grid point, a weighted average is computed from the data at stations within a specified distance from the grid point. When stability is determined from a single meteorological station, as in the PUFF modeling for the FSAR, stability will not vary spatially within a given hour.

INTERROGATORY NO. 129. Describe the annual average concentration at the exclusion area boundary to be expected if effluents are released into the atmosphere at two discrete positions within the plant site.

ANSWER. The annual average concentration at the exclusion area boundary is not calculated.

INTERROGATORY NO. 130. Regarding Environmental Report, p. 6.1-30, paragraph 2; explain the criteria used to judge when a puff can no longer make a "significant" contribution to the concentration.

ANSWER. Two criteria are used to determine if the contribution of a puff to a receptor should be calculated. These criteria are based upon proximity of the puff to the receptor and upon the concentration at the center of the puff.

If the puff is not proximate to the receptor, its impact is not calculated. If the puff is proximate but its center point relative concentration ($\chi_p$) is below a specified value ($\chi_{min}$), then the impact is not calculated. Both of these criteria are built into a radius of influence ($r_p$) for each puff:

$$r_p = \sigma_y \left[-2 \ln \frac{\chi_{min}}{\chi_p}\right]^{1/2} \quad \text{or} \quad \gamma_c \sigma_y,$$

whichever is smaller, where "$\sigma_y$" is the horizontal standard deviation of the puff effluent mass distribution, and "$\gamma_c$" is the number of horizontal standard deviations from the puff center to the puff edge. In the PUFF modeling for the FSAR, $\gamma_c$ = 3.0. In the PUFF model, the above equation is evaluated only when $\chi_{min} < \chi_p$ This is also the criterion referred to on Page 6.1-30 of the Environmental Report. Puffs can no longer contr bute a significant concentration at any grid point, and are therefore discarded, when $\chi_{min}$ is is greater than $\chi_p$. In the PUFF modeling for the FSAR, $\chi_{min}$ = 1.0x10-$^{11}$ hr/m$^3$.

INTERROGATORY NO. 131. This interrogatory pertains to "Environmental Report Operating License Stage Vol. I" pg. 2.3-7. In light of the precipitation measuring equipment's malfunction at the Callaway plant site, state the precise correlation coefficient between the average monthly rainfall at Columbia and that at the plant site.

ANSWER. Due to the equipment malfunctioning at the Callaway Plant site, not enough site-specific data are available to determine the precise correlation coefficient between data obtained at the two monitoring locations.

INTERROGATORY NO. 132. This interrogatory pertains to SNUPPS, Callaway Site Addendum, Section 11.3.3.4.1. NRC Regulatory Guide 1.111 states "Deposition of radionuclides over large bodies of water is not considered in this guide. Such deposition will be analyzed on a case-by-case basis." Provide this analysis for the Callaway Unit I impact on the Missouri River.

ANSWER. The Missouri River is not a large body of water in the sense intended in Regulatory Guide 1.111. Therefore, this analysis was not performed.

INTERROGATORY NO. 133. Clarify the discrepancy between what Table 6.1-10 is stated to contain in the first sentence of paragraph 3, pg. 6.1-37 and the title of the table, "Water Quality Parameters Measured in Samples Taken During Intake/Discharge and Barge Slip Construction" (SNUPPS FSAR).

ANSWER. Assuming the intended citation is to the Environmental Report-Operating License Stage, the reference to Table 6.1-10 in the first sentence of paragraph 3, page 6.1-37 of the ER-OLS is a typographical error. The correct reference is Table 6.1-8.

INTERROGATORY NO. 134. Regarding SNUPPS FSAR, pg. 6.1-17, paragraph 3, state the limitations imposed on the quality and completeness of the Callaway meteorological data caused by the malfunction of the digital data acquisition.

ANSWER. As in the previous interrogatory, the correct reference is the Environmental Report.

The digital data acquisition system was inoperative during the second data collection period (March 16, 1978 to March 16, 1979). Since the analog recording system utilized during this period is an acceptable method for the recording of data according to U.S. NRC Regulatory Guide 1.23, the malfunction of

the digital system imposed no limitation on the quality of the Callaway meteorological data.

The limitation on completeness of the collected data base is the amount of data lost due to malfunction of the analog recording system. These data losses are listed below for each parameter for the data collection period in question.

| PARAMETER | NUMBER OF HOURS OF DATA LOST | PERCENT OF TOTAL DATA FOR THIS PERIOD |
|---|---|---|
| Wind Speed (10 m) | 0 | 0.0 |
| Wind Direction (10 m) | 0 | 0.0 |
| Wind Speed (60 m) | 0 | 0.0 |
| Wind Direction (60 m) | 0 | 0.0 |
| Wind Speed (90 m) | 2 | 0.02 |
| Wind Direction (90 m) | 8 | 0.09 |
| Temperature | 613 | 7.00 |
| Dew Point | 613 | 7.00 |
| Delta Temperature (60-10 m) | 613 | 7.00 |
| Delta Temperature (90-10 m) | 671 | 7.66 |
| Precipitation* | -- | -- |

* National Weather Service data were used.

INTERROGATORY NO. 135. Explain the term "qualitative agreement" in the second sentence of paragraph 1, page 2.3-11, Environmental Report Operating License Stage Vol. I.

ANSWER. This term was used by Holzworth to indicate that the result of a study performed to predict the incidence of days with high meteorological potential for air pollution correlated with the number of episode days recorded during that period, although the numbers of the two events were not identical. Since the definitions of the two events differ somewhat, quantitative agreement is expected to be less than rigorous.

Reference: Holzworth, 1972, Mixing heights, wind speeds and potential for urban air pollution throughout the contiguous United States. United States EPA, no. AP-101.

INTERROGATORY NO. 136. State the longest period that the critically limiting conditions described on page 2.3-10 of the Environmental Report continuously occurred.

ANSWER. An investigation of the document by Holzworth, referenced in the previous answer, does not indicate exactly how long these limiting conditions continuously occurred; however, from the data presented by Holzworth, it can be seen that the longest period in which these conditions could have occurred was from 2.5 to 3 days.

INTERROGATORY NO. 137. Explain the choice of 1960-64 as the basis for arriving at a typical number for the frequency of occurrence of episode days (Environmental Report).

ANSWER. According to Holzworth, referenced in the answer to Interrogatory No. 135, this data base was limited to 5 years for economy, and to pre-1965 because the required hourly surface observations were available on punched cards only through 1964.

INTERROGATORY NO. 138. List all alpha emitting radionuclides that will be released in the gaseous emission and give the estimated single reactor amount per year of such emission in terms of curies per year.

ANSWER. None of the nuclides in gaseous effluents identified in FSAR Table 11.1-2 which were calculated in accordance with NUREG-0017 is an alpha emitter.

INTERROGATORY NO. 139. Will there be any tritium in the spent fuel pool? If so, how much will be added (curies per year) in each year of the normal expected or planned operation of the Callaway Plant? Will such additions increase as the plant gets older? If so, estimate the amount or rate of increase and the bases for the estimate.

ANSWER. The basis for the calculation for tritium production and releases for the Callaway Plant, NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR - GALE Code), predicts that tritium levels will increase to a steady state value after several years of operation and provides for adjustments in the calculation to account for the buildup. The tritium release rates for gaseous and liquid effluents given in FSAR Table 11.1-2 are the predicted steady state values and are, therefore, the predicted tritium production rate.

FSAR Table 12.1-11 states the number of curies of tritium expected to be released in gaseous effluents from the spent fuel pool. Since there are no significant liquid effluents from the spent fuel pool, this number reflects the expected annual tritium addition to the spent fuel pool.

INTERROGATORY NO. 141. References to temporary storage of spent fuel are made in Sections 9.1.4.2.1 (p. 9.1-28), 9.1.4.2.3 (p. 9.1-38), 9.1.4.2.3.1. Phase V (p. 9.1-46) and 12.2.1.7 (p. 12.2-5) (SNUPPS, FSAR Vols. 6 and 9). State what changes in design parameters will be required if storage time is increased beyond the estimated temporary storage time in the spent fuel pools.

ANSWER. No such evaluation has been performed.

INTERROGATORY NO. 142. Referring to Table 9.1-4 (SNUPPS FSAR Vol. 6) "Fuel Pool Cooling and Cleanup System Design

Parameters," state how evaporation rates can be controlled and whether any changes in the evaporation rate are to be expected due to longer storage time of spent fuel.

ANSWER. The spent fuel pool evaporation rates given in FSAR Table 9.1-4 are a function of the pool water temperature and the fuel building HVAC air temperature and relative humidity. The fuel pool water temperatures given in Table 9.1-4 are the maximum design values for normal operation and during plant shutdown. These are maximum values and are not effected by length of fuel storage time.

No specific actions are taken to control evaporation rates other than to maintain the spent fuel pool water temperature up to the maximum allowable value.

INTERROGATORY NO. 143. State whether there will be any monitoring of the amount of tritium being formed when neutrons from the spent fuel react with boron in the spent fuel pool.

ANSWER. There will not be any.

INTERROGATORY NO. 144. State the expected uranium, transuranic, and fission product leaching rates from exposed fuel pellets resulting from increased fuel rod cladding failure caused by long-term storage in spent fuel pools.

ANSWER. Storage of spent fuel in the spent fuel pool for the duration discussed in the FSAR is not expected to result in increased fuel rod cladding failure.

INTERROGATORY NO. 145. State the time periods that Union Electric estimates spent fuel rods can be stored in spent fuel pools before degradation of cladding and structural parts and increased compacting provide an increased risk of criticality incident.

ANSWER. As stated in FSAR 9.1.2, the spent fuel storage facility is designed to maintain the fuel assemblies in a

subcritical array as required by the NRC in General Design Criteria 62 of 10 CFR 50, Appendix A. The subcriticality requirement will be satisfied throughout the duration of spent fuel storage.

INTERROGATORY NO. 146. State what percentage of fuel cladding is expected to fail after one year's storage in the spent fuel pool; after five years' storage; after thirty years' storage.

ANSWER. No specific estimates have been made by the Union Electric Company of the percentage of fuel cladding that is expected to fail after one, five and thirty years of storage; however, the NRC's findings in NUREG-0575, Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, indicate that zircaloy cladding, such as that used in the Callaway plant, is highly resistant to water corrosion.

INTERROGATORY NO. 147. State how frequently the water of the spent fuel pool will be monitored and for what specific radioactive isotopes it will be monitored.

ANSWER. As stated in FSAR Table 9.3-4, the spent fuel pool water will be sampled at a frequency of once per week. These samples will be analyzed for gross beta and gamma activity.

INTERROGATORY NO. 148. State which radioactive isotopes are expected to be released to the liquid waste effluent from the spent fuel pool.

ANSWER. There are no radioactive isotopes expected to be released to the liquid waste effluent from the spent fuel pool.

INTERROGATORY NO. 149. Identify the amount of fuel expected to be held in the spent fuel pool and the assumed duration of time on which the estimates in Table 12.2-11 were based (SNUPPS, FSAR, Vol. 9).

ANSWER. The storage capacity of the Callaway Plant spent fuel racks is provided in FSAR Table 9.1-2. The parameters given in FSAR Table 12.2-11 are not a function of the duration of time that spent fuel has been stored in the spent fuel pool.

INTERROGATORY NO. 150. State what provisions have been made by Union Electric to insure adequate pool water quality during extended fuel storage periods.

ANSWER. A fuel pool cooling and cleanup system, described in FSAR section 9.1.3, is provide to maintain the desired spent fuel pool water quality. This system is designed to maintain spent fuel pool water quality throughout plant life.

INTERROGATORY NO. 151. State what design parameters have been included in the fuel storage racks to prevent swelling due to build up of hydrogen gas, subsequent jamming of the fuel assemblies, and rupturing of the fuel rods.

ANSWER. There are no design parameters which have been included in the fuel racks to prevent swelling due to build up of hydrogen gas.

INTERROGATORY NO. 152. Are radioactive particles readily adsorbed on suspended river sediment, which in turn is carried by the river many miles downstream?

ANSWER. Some radioactive particles will be.

INTERROGATORY NO. 153. Why were not each of the following potential pathways of exposure to 'man' from radionuclides in liquid effluents considered in Section 5.2, Vol. II Environmental Report Operating License Stage?

(a) Crop irrigation;

(b) River dredging;

(c) Use of gravel and sand for construction and other purposes;

(d) Domestic use for drinking, cooking, cleaning and bathing;

(e) Water purification plants, i.e., retention and accumulation of radioactive material in filters, ion exchange columns, sludge settling basins during water and sewage treatment processes.

ANSWER. Only pathways which were considered significant as defined in Regulatory Guide 1.109, Revision 1, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, were evaluated.

INTERROGATORY NO. 154. Have the Union Electric estimates of levels of radioactivity in 'man' due to fish consumption taken into account:

(a) that some fish feed at the bottom of rivers, where radioactive isotopes in the sediment may be much more concentrated than closer to the surface?

(b) That specific radioactive materials (e.g., Cs137 and Sr90 may become highly concentrated in shellfish, fish and waterfowl?

(c) If the answer to either or both of the above paragraphs is in the negative, explain why this was not taken into account.

(d) If the answer to either or both paragraph (a) and (b) of this interrogatory is affirmative, explain how it was taken into account and and identify all documents which reflect its consideration.

ANSWER.

(a) Yes.

(b) Yes.

(c) Not Applicable.

(d) Bioaccumulation Factors are used in the dose calculations which take into account the fact that some radionuclides are concentrated in aquatic biota. Sedimentation and resuspension processes are treated implicitly in theE Boiler and Pressure Vessel Code Section XI. bioaccumulation factors; these factors are averages from many observations that have been made and include reconcentrations by bottom feeders and other organisms that are eaten by fish. See Technical Note ORP/EAD-76-4, USEPA, Oct. 1976, "A Computer Code (RVRDOS) to Calculate Population Doses from Radioactive Liquid Effluents and an Application to Nuclear Power Reactors on the Mississippi River Basin."

INTERROGATORY NO. 155. Is nickel present as a corrosion product in nuclear reactors?

ANSWER. As stated in FSAR Section 5.2.3, nickel is one of the materials of construction of the reactor coolant system pressure boundary, and therefore could potentially be a corrosion product.

INTERROGATORY NO. 156. Why, in Table 11.1-2 (SNUPPS FSAR) are radioactive isotopes of Nickel (e.g., Ni59 and Ni63) not listed among the "corrosion and activation products"?

ANSWER. The radioactive isotopes listed in FSAR Table 11.1-2 were identified in accordance with the guidance provided by the Nuclear Regulatory Commission in NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). This method, which is based on experience from operating reactors, does not identify radioactive isotopes of nickel in the expected gaseous and liquid effluents.

INTERROGATORY NO. 157. Regarding SNUPPS, Callaway Site Addendum: p. 2.3-66, state the bases for the choice of 2.26 days 8 days as the half-lives used in the PUFF model calcu ions.

ANSWER. These half-lives were used because they were specified in Nuclear Regulatory Guideline 1.111 as acceptable for conservative estimates of radioactive decay for short-lived noble gases (2.26 days) and for all iodines (8 days) released to the atmosphere.

INTERROGATORY NO. 158. Identify all expert witness that are expected to testify for Union Electric at each hearing on Joint Intervenors' Contentions in this matter, and state separately for each person identified:

(a) The subject matter on which the expert is expected to testify;

(b) The substance of the facts and opinions to which the expert is expected to testify;

(c) A summary of the grounds for each opinion.

ANSWER. Union Electric has not yet determined who will testify as an expert on the subject of Joint Intervenors' Contentions 1 or 2.

INTERROGATORY No. 159. Identify, separately for each of the above interrogatories and the subparts thereof, the person(s) providing the answer.

ANSWER. Donald Schnell, General Manager, Union Electric Company, 1901 Gratiot Street, St. Louis, Missouri 63166, provided information in response to the following interrogatories:

No. 48
No. 49

William H. Zvanut, Supervising Engineer - Nuclear, Union Electric Co., 1901 Gratiot Street, St. Louis, Missouri 63166, provided information in response to the following interrogatories:

No. 22
No. 23
No. 34
No. 36(a)-(c)
No. 53(a)-(h)
No. 55
No. 56
No. 67(a)-(c)
No. 70
No. 71
No. 75(b)
No. 85(b)

Joseph Laux, Supervising Engineer - Quality Assurance, Union Electric Company, Callaway Plant, Fulton, Missouri 65251 provided information in response to the following interrogatory:

No. 1(a) and (b).

Alan C. Passwater, Superintendent, Licensing, Union Electric Company, 1901 Gratiot Street, St. Louis, Missouri 63166, provided information in response to the following interrogatories:

No. 69
No. 74(a) and (c)

Richard Wendling, Supervising Engineer - Nuclear, Union Electric Company, 1901 Gratiot Street, St. Louis, Missouri 63166, provided information in response to the following interrogatory:

No. 94

Kenneth W. Kuechenmeister, Supervising Engineer - Nuclear Construction, Union Electric Co., Callaway Plant, Fulton, Missouri 65251, provided information in response to the following interrogatories:

No. 8(c) and (d)
No. 9(a) and (b)
No. 95(c)

Neal Slayton, Supervising Engineer - Environmental, Union Electric Company, 1901 Gratiot Street, St. Louis, Missouri 63166, provided information in response to the following interrogatory:

| | |
|---|---|
| No. 104 | No. 117 |
| No. 108 | No. 119 |
| No. 112 | No. 120 |
| No. 114 | No. 152-154 |
| No. 115 | No. 158 |

Dr. Bernard Meyers, Project Manager, Bechtel Power Corporation, P.O. Box 607, 15740 Shady Grove Road, Gaithersburg, Maryland 20760, provided information in response to the following interrogatories:

No. 2(a)-(c)
No. 3(a)-(c)
No. 4(c) and (d)
No. 5(a)-(d)
No. 6(a), (d) and (e)
No. 7(a) and (c)
No. 8(a) and (b)
No. 14(a) and (b)
No. 15
No. 16
No. 20
No. 22
No. 24(a) and (b)
No. 25
No. 27
No. 33(a) and (b)
No. 35
No. 42(a) and (b)
No. 48
No. 49
No. 50
No. 51(a) and (b)
No. 52(a) and (b)
No. 56
No. 57
No. 59
No. 60
No. 61
No. 64
No. 65
No. 67(a)-(c)
No. 72
No. 73
No. 75(a) and (b)
No. 76
No. 78(a) and (b)
No. 80
No. 82(a)-(c)
No. 83(a)-(c)
No. 84(a)-(c)
No. 85(a) and (b)
No. 86
No. 87
No. 88(a)-(c)
No. 89
No. 96(a) and (b)
No. 97
No. 98
No. 100
No. 101
No. 102
No. 103
No. 105
No. 106
No. 107
No. 109
No. 110
No. 111
No. 113
No. 121
No. 124
No. 138
No. 139
No. 140
No. 141
No. 142
No. 143
No 144
No. 145
No. 146
No. 147
No. 148
No. 149
No. 150
No. 151
No. 155
No. 156

R. David Neal, Project Civil Engineer, Daniel Construction Company, Callaway Plant, P.O. Box 108, Fulton, Missouri 65251, provided information in response to the following interrogatories:

| | |
|---|---|
| No. 8(c) and (d) | No. 40 |
| No. 9(a) and (b) | No. 41(a)-(d) |
| No. 10 | No. 42(c) |
| No. 11 | No. 43 |
| No. 12 | No. 44 |
| No. 13 | No. 45 |
| No. 17(a)-(c) | No. 46 |
| No. 18 | No. 47 |
| No. 19 | No. 54 |
| No. 21 | No. 58 |
| No. 24(a) | No. 62 |
| No. 26 | No. 63 |
| No. 28 | No. 66(a)-(f) |
| No. 29 | No. 68 |
| No. 30 | No. 74(b) |
| No. 31(a) and (b) | No. 77 |
| No. 32(a) and (b) | No. 79 |
| No. 33(b) | No. 80 |
| No. 36(a)-(c) | No. 81 |
| No. 37 | No. 90 |
| No. 38 | No. 91 |
| | No. 92 |

Robert A. Hagar, Project Services Quality Control Engineer, Daniel Construction Company, Callaway Plant, P.O. Box 108, Fulton, Missouri 65251, provided information in response to the following interrogatories:

No. 4(a) and (b)
No. 6(b) and (c)
No. 7(b)
No. 93(a)-(c)
No. 95(c)

Raymond A. Somers, Project Mechanical Quality Control Engineer, Daniel Construction Company, Callaway Plant, P.O. Box 108, Fulton, Missouri 65251, provided information in response to the following interrogatories:

No. 95(a)-(c) and (e)

George M. Stephens, Construction Manager, Daniel Construction Company, Callaway Plant, P.O. Box 108, Fulton, Missouri 65251, provided information in response to the following interrogatory:

No. 95(d)
No. 99

Patrick H. Roney, Project Civil Quality Control Engineer, Daniel Construction Company, Callaway Plant, P.O. Box 108, Fulton, Missouri 65251, provided information in response to the following interrogatory:

No. 39

Stanley J. Seiken, Quality Assurance Manager, SNUPPS, 5 Choke Cherry Road, Rockville, Maryland 20850, provided information in response to the following interrogatory:

No. 99

Theodore Totten, Project Manager, Cives Steel Company, 8 Church Street, Gouverneur, New York 13642, provided information in response to the following interrogatories:

No. 3(b) and (c)
No. 5(c) and (d)

D.W. Pfeifer, Wiss, Janney, Elstner & Associates, Inc. 330 Pfingsten Road, Northbrook, Illinois 60062, provided information in response to the following interrogatory:

No. 53(b)-(h)

Dr. Donald L. Ballman, Associate, Dames & Moore, 1650 Northwest Highway, Park Ridge, Illinois 60068, provided information in response to the following interrogatories:

No. 116
No. 118
No. 122
No. 123
No. 125-137
No. 157

TABLE 3(b)

Embedded Plates With Headed Stud Anchors

Fabricated And Shipped By Cives Steel Company

On Or Before June 9, 1977

| SHIPPING DATE | QUANTITY |
|---|---|
| 1/16/76 | 240 |
| 2/76 | 75 |
| 2/76 | 109 |
| 6/21/76 | 12 |
| 7/07/76 | 279 |
| 7/09/76 | 351 |
| 7/14/76 | 280 |
| 7/16/76 | 80 |
| 7/23/76 | 8 |
| 8/06/76 | 234 |
| 8/06/76 | 381 |
| 8/19/76 | 390 |
| 9/09/76 | 307 |
| 9/03/76 | 294 |
| 9/03/76 | 364 |
| 9/22/76 | 278 |
| 9/28/76 | 279 |
| 10/14/76 | 501 |
| 11/29/76 | 600 |
| 1/07/77 | 624 |
| 2/09/77 | 1,000 |
| 2/23/77 | 324 |
| 3/14/77 | 390 |
| 3/31/77 | 500 |
| 4/18/77 | 500 |
| 5/03/77 | 23 |
| 6/07/77 | 94 |
| TOTAL | 8,517 |

TABLE 4 (a)

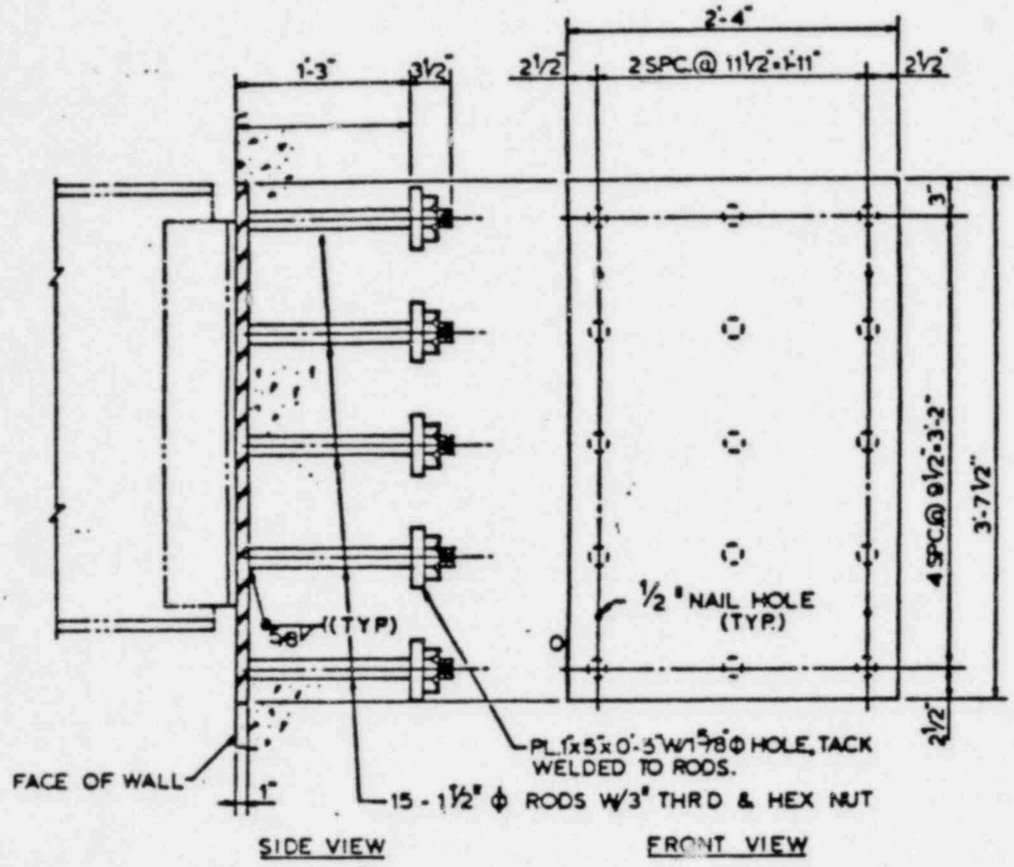
Embedded Plates With Headed Stud Anchors

Received At The Callaway Plant

On Or Before June 9, 1977

| RECEIVING DATE | QUANTITY | |
|---|---|---|
| 1/20/76 | 240 | |
| 2/02/76 | 75 | |
| 2/02/76 | 109 | |
| 6/24/76 | 12 | |
| 7/12/76 | 280 | OS&D 258 |
| 7/14/76 | 350 | OS&D 257 |
| 7/16/76 | 280 | |
| 7/26/76 | 80 | |
| 7/26/76 | 8 | |
| 8/9/76 | 234 | |
| 8/9/76 | 381 | |
| 8/24/76 | 390 | |
| 9/13/76 | 307 | |
| 3/16/76 | 294 | |
| 9/16/76 | 364 | |
| 10/4/76 | 278 | |
| 10/5/76 | 279 | |
| 10/18/76 | 501 | |
| 12/9/76 | 600 | |
| 1/21/77 | 624 | |
| 2/12/77 | 1,000 | |
| 2/24/77 | 324 | |
| 3/16/77 | 390 | |
| 4/05/77 | 500 | |
| 4/21/77 | 500 | |
| 5/05/77 | 23 | |
| 6/09/77 | 94 | |
| TOTAL | 8,517 | |

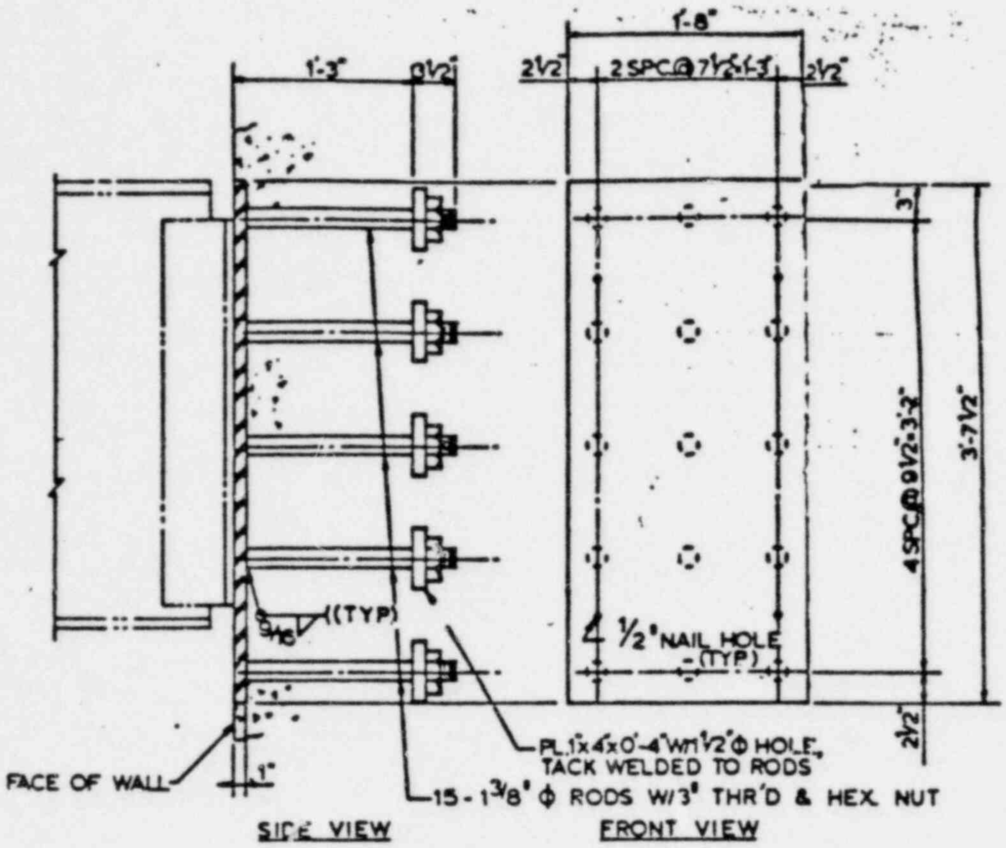
FIGURE 5(b)




EP111

FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS




EP 211

FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS

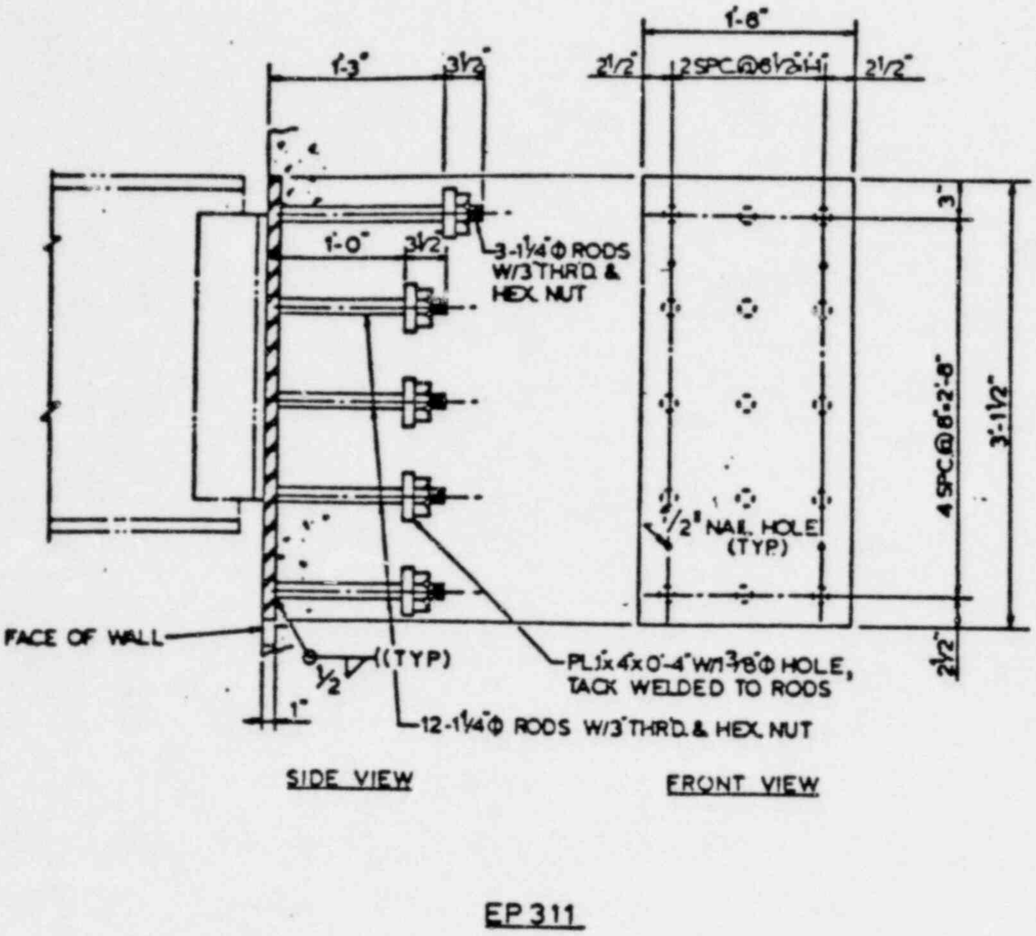
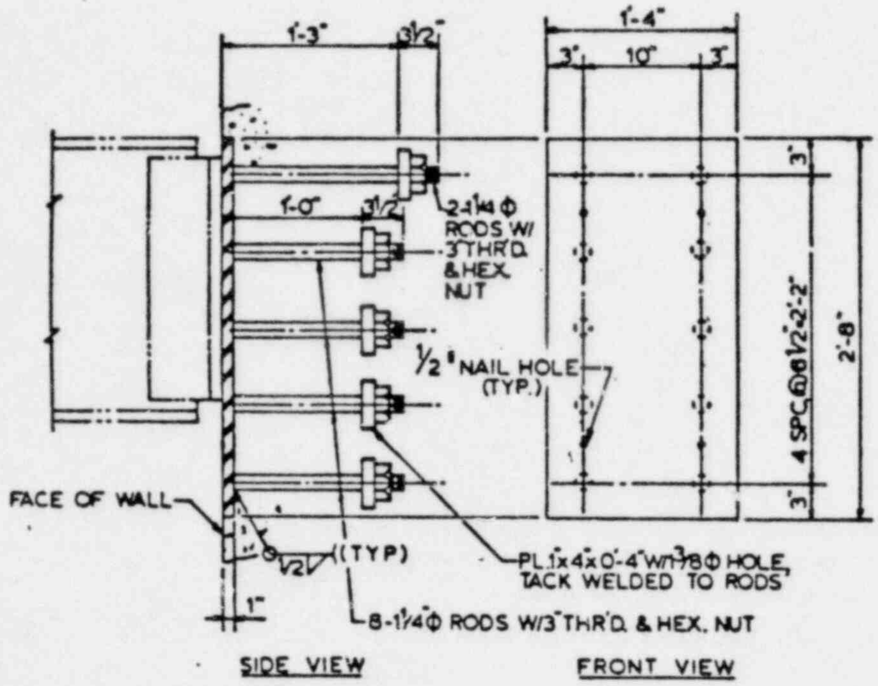



FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS




EP411

FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS

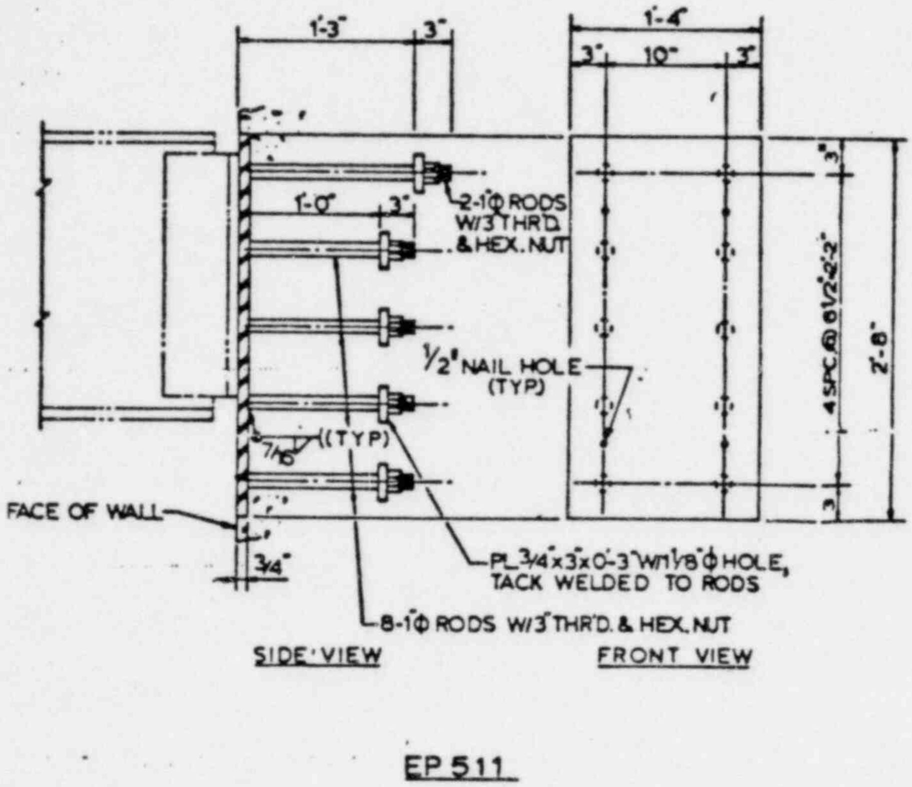
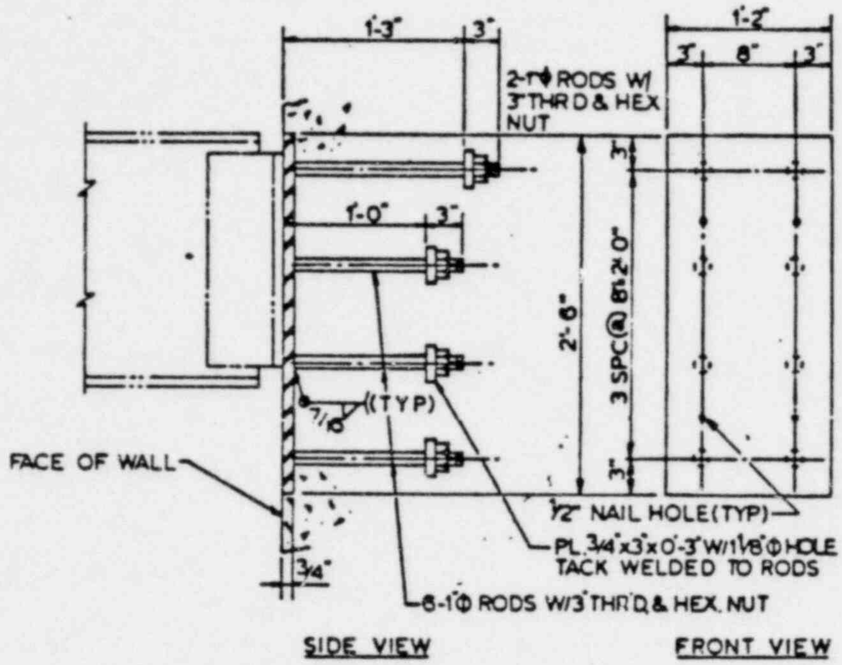



FIGURE 5(b)

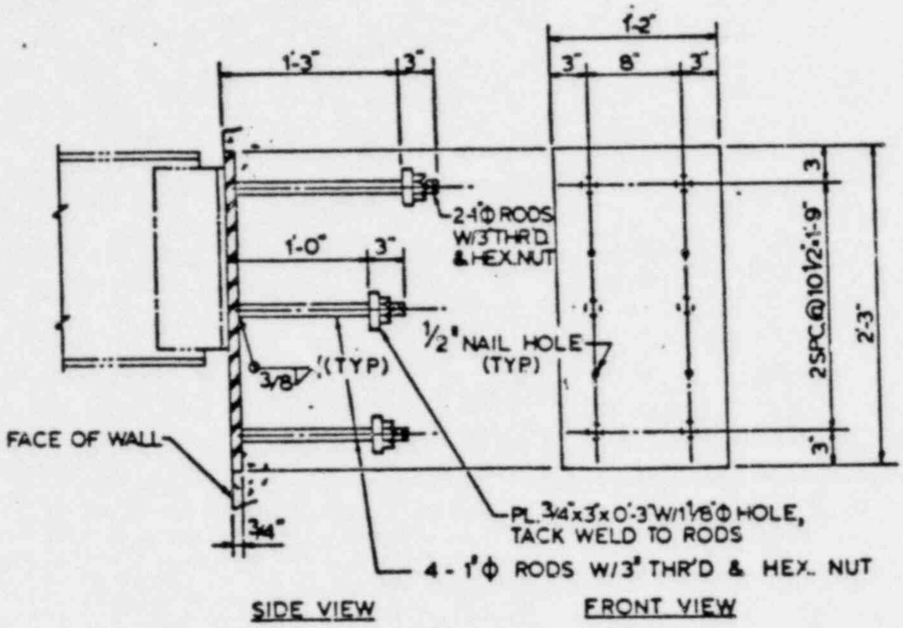
EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS




EP611

FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS




EP711

FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS

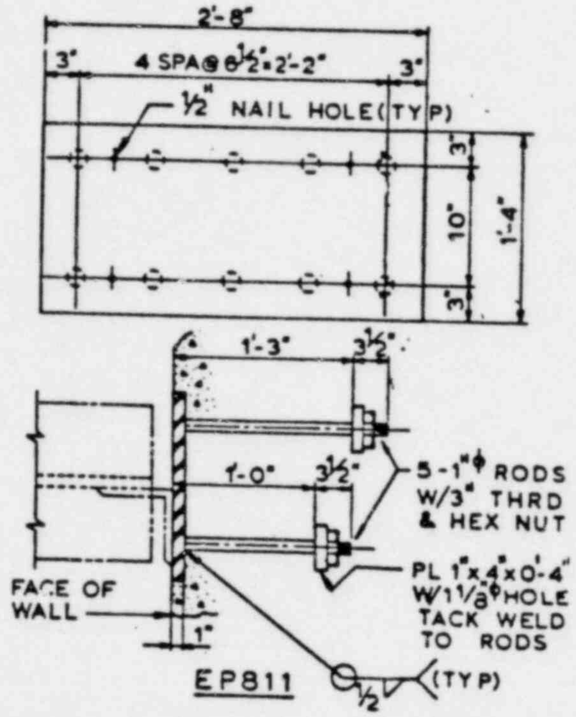
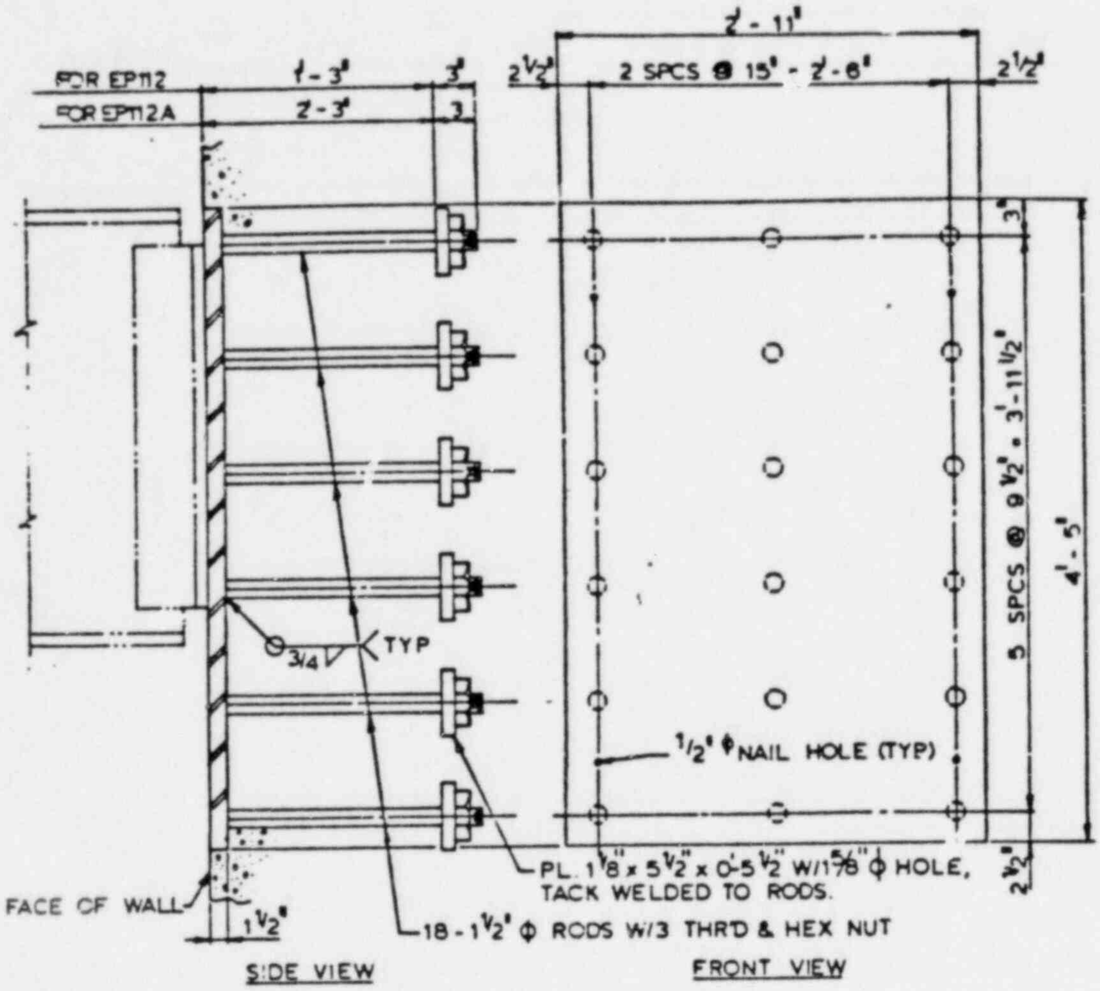



FIGURE 5(b)

EMBEL , PLATES WITH MANUALLY WELDED ROD ANCHORS




EP112 & EP112A

FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS

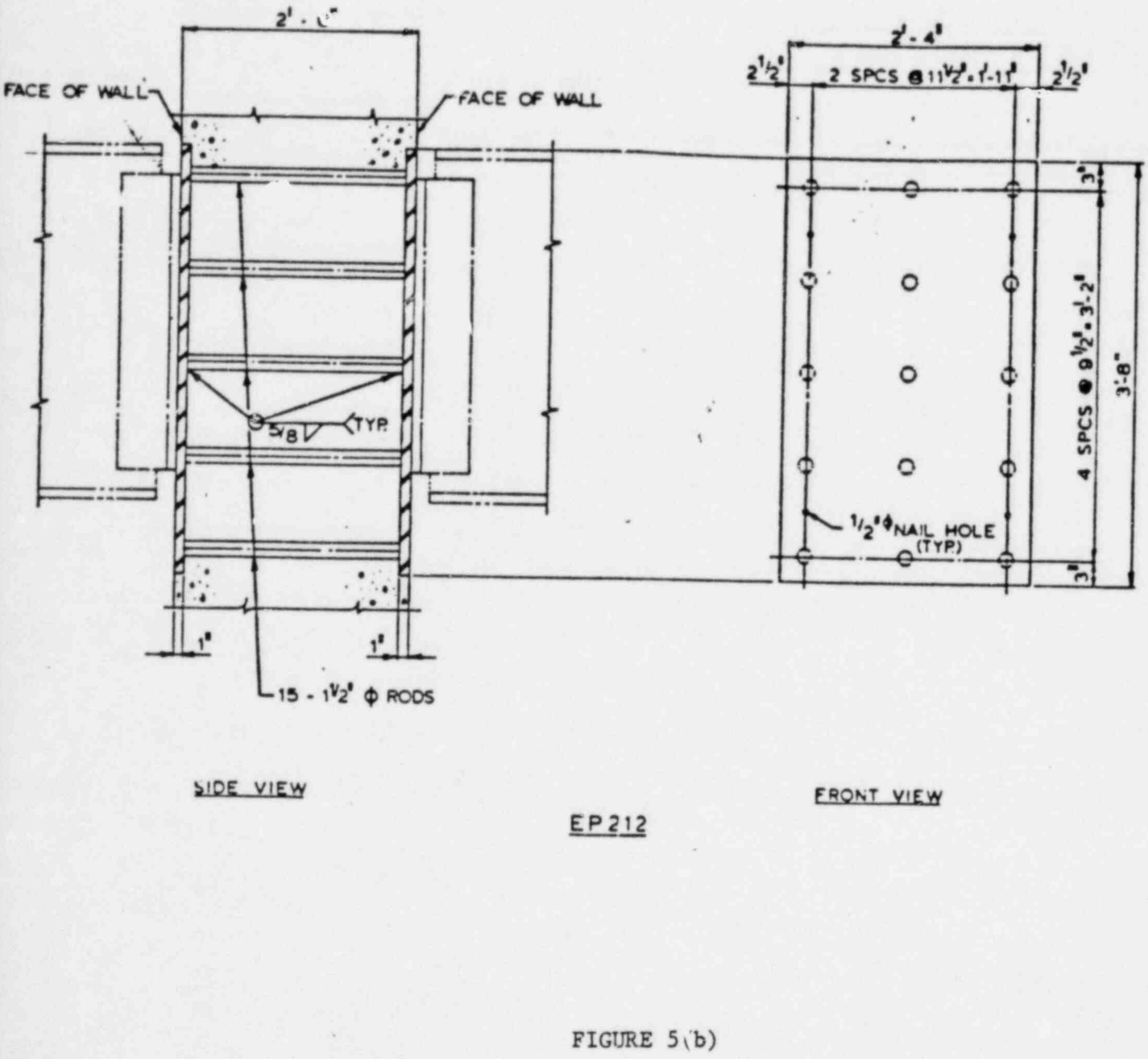



FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS

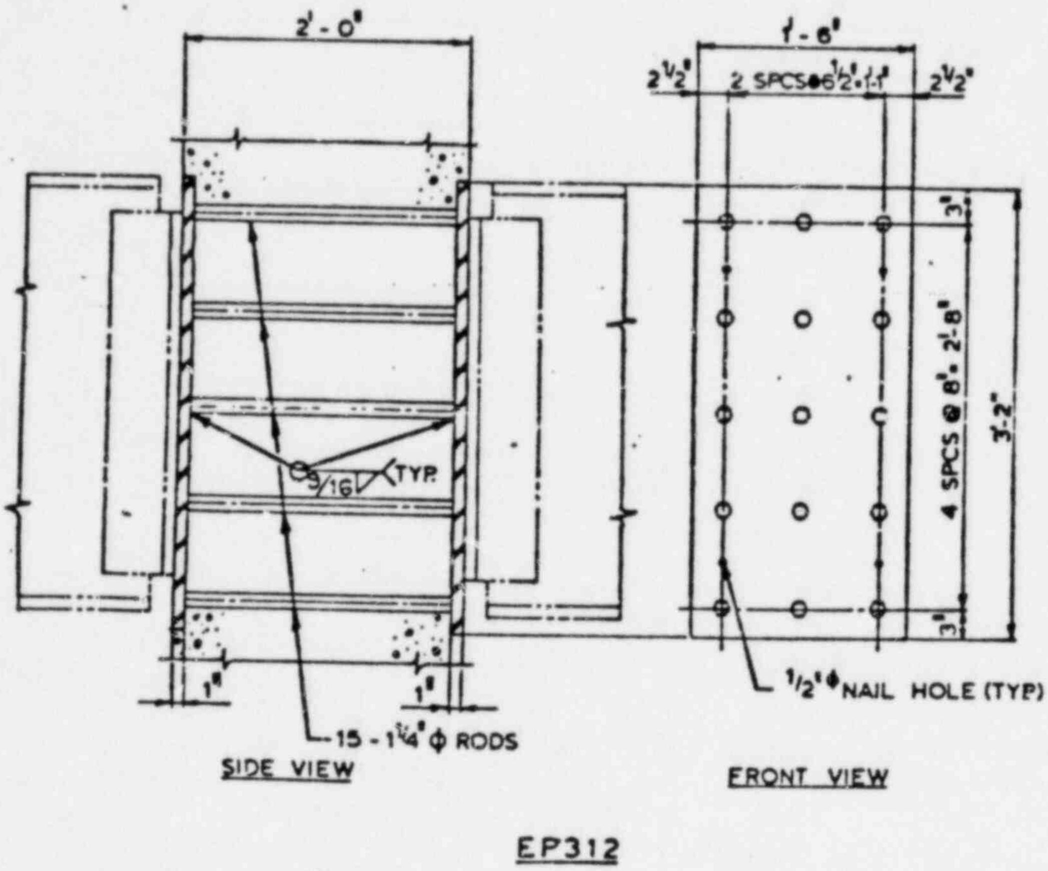



FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS

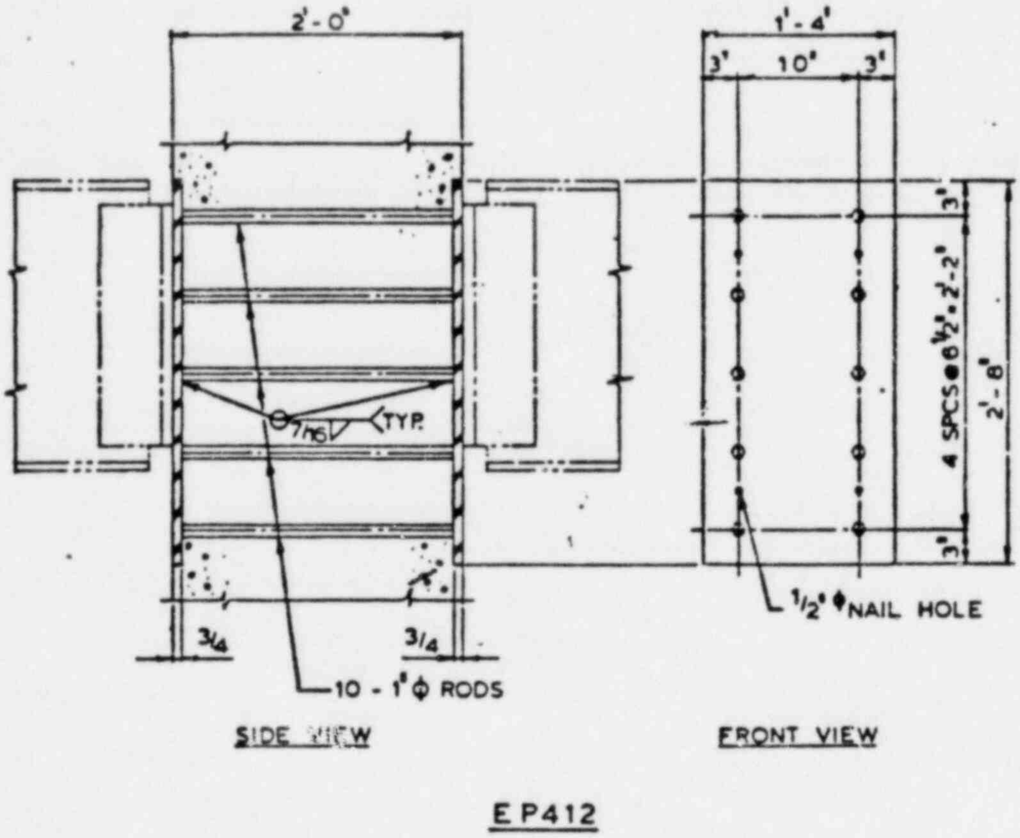



FIGURE 5(b)

EMBEDDED PLATES WITH MANUALLY WELDED ROD ANCHORS

TABLE 5 (c)

Embedded Plates with Manually Weld Rod Anchors

Fabricated and Shipped by Cives Steel Company

On Or Before June 9, 1977

| SHIPPING DATE | QUANTITY | |
|---|---|---|
| 10/06/76 | 44 | |
| 02/76 | 190 | |
| 02/76 | 146 | |
| 02/76 | 264 | |
| 6/21/76 | 8 | |
| 7/07/76 | 31 | OS&D 258 (5 too many) |
| 7/16/76 | 17 | OS&D 261 (1 too many) |
| 8/19/76 | 1 | |
| 6.07/77 | 44 | |
| TOTAL | 745 | |

TABLE 6 (b)

Embedded Plates with Manually Welded Rod Anchors

Received at the Callaway Site

On Or Before June 9, 1977

| RECEIVING DATE | | QUANTITY |
|---|---|---|
| 1/20/76 | | 44 |
| 2/2/76 | | 190 |
| 2/2/76 | | 146 |
| 2/2/76 | | 264 |
| 6/24/76 | | 8 |
| 7/12/76 | | 26 |
| 7/26/76 | | 16 |
| 8/24/76 | | 1 |
| 6/09/77 | | 44 |
| | TOTAL | 739 |

TABLE 53(b)

SPECIFICATIONS

A. ELECTRICAL

| | |
|---|---|
| 1. Operating frequency range | 15 to 300 KC |
| 2. Time measurement | Range - 5K-500 µ sec in 1/2 µ sec stop<br>Range - 1K-1000 µ sec in 1 µ sec stop<br>Range - 5K-5000 µ sec in 5 µ sec stop |
| 3. Timebase accuracy | 100KC crystal ± .005% |
| 4. Amplifier sensitivity | 10 microvolts/centimeter @ 20 KC |
| 5. Power input requirements | 105-130 volts, 50-60 cps AC<br>50 watts standby - 100 watts operating |
| 6. Input impedance to transmitting transducer | 2800 ohms |
| 7. Power output | Peak 1200 watts, Avg. 3.25 watts |
| 8. Pulse rate | 69 pulses per second |
| 9. Output impedance to transmitting transducer | 3800 ohms |

B. PHYSICAL

| | |
|---|---|
| 1. Temperature (Instrument & transducers) | Up to 120°F |
| 2. Package | Aluminum case with plastic handle |
| 3. Dimensions | 8 1/2" x 11" x 15" |
| 4. Weight | 23 lbs. |

ATTACHMENT 53(d)

Wiss, Janney, Elstner and Associates, Inc.

NAME:

Richard Stephen Vyskocil
2124 Brentwood, Northbrook, IL 60062
Phone (312 498-2849

TITLE: Senior Technician

CITIZENSHIP:

United States

BIRTH:

October 15, 1947

Mr. Vyskocil joined Wiss, Janney, Elstner and Associates in October of 1966. During his eleven years with the firm he has participated in numerous investigations of materials and distressed structures under the supervision of qualified engineers. He is a graduate of De Vry Institute of Technology where he earned a electronics technician degree. He was initially trained by Mr. Hedien and has additionally received review under the supervision of Mr. Oleson. Mr. Vyskocil has aided in the development of the WJE Soniscope Training Course and has conducted it twice in recent months. Principle jobs in which Mr. Vyskocil was a principal investigator include Fire Damaged Bridge in Napa, Idaho, Quaker Oats facility in Mexico City, McCormick Place, Soldier Field, Hydroelectric plant in Wenatchee, Washington.

Mr. Vyskocil has had considerable experience in the use of electronic equipment in relation to strain instrumentation. His experience, relative to the use of the soniscope is extensive. He has used the soniscope for approximately six years on concrete investigation.

Mr. Vyskocil is in good health and is physically fit to perform chores associated with this project.

NAME:

Gregory G. Hedien
9042 Cumberland Avenue - Niles, Illinois 60648

Phone -(312)- 296 - 2873

TITLE: Engineering Assistant

CITIZENSHIP:
United States

BIRTH:
August 16, 1938

Mr. Hedien joined Wiss, Janney, Elstner and Associates in August of 1968. Prior to joining our firm he was employed for eight (8) years at Portland Cement Association.

His duties have included extensive work in aggregates, concrete and instrumentation. He has been used on numerous building failures, fire damaged buildings and distressed structures. He has completed two years at Northwestern University.

Mr. Hedien's experience in the use of the soniscope dates back to his employment by PCA, where he worked with Mr. C. C. Oleson in the use of the use of the soniscope in nondestructive testing of concrete.

Some of the more prominent jobs he has used the soniscope on are:

Northern State Power - Red Wing, Minnesota
Hannover Street Bridge Investigation - Baltimore, Maryland
Soldier Field Stadium Investigation - Chicago, Illinois
Corps of Engineers - Bureau of Record Building - Fire
Zion Nuclear Station - Zion Illinois
Gunpowder Bridge Investigation - Baltimore, Maryland

Mr. Hedien is in good health and is physically fit to perform chores associated with this project.

KIMBALL JAMES BEASLEY

STRUCTURAL MATERIAL ENGINEER

ENGINEERING EXPERIENCE:

Since 1973, Kimball James Beasley has been involved in a wide range of experimental research investigations, structural analysis and engineering design projects with Wiss, Janney, Elstner and Associates, Inc.

Mr. Beasley's professional experience has been divided between concrete design and research for nuclear generating plants in the midwest, nondestructive investigation of in situ concrete throughout the United States and in the Bahamas. Mr. Beasley has over three years of field experience with the soniscope. His initial training was received under the direction of Mr. Hedien and Mr. Vyskocil. His field experience include:

| | |
|---|---|
| Soldier Field Stadium | Chicago, Illinois |
| Woolworth Building | New York, New York |
| Bahama Oil Refining Company | Freeport, Grand Bahama Island |
| Hanford No. 2 Nuclear Plant | Hanford, Washington |
| Portland Cement Plant | Echo, Texas |
| Lincoln Tunnel | New York, New York |
| Wrigley Building | Chicago, Illinois |
| Hennepin County Courthouse | Minneapolis, Minnesota |
| Fairlane Taconite Plant | Hibbing, Minnesota |
| St. Joe Lead Plant | Pittsburgh, Pennsylvania |
| Fargo Building | Boston, Massachusetts |
| Byron Nuclear Plant | Rockford, Illinois |
| LaSalle Nuclear Plant | Seneca, Illinois |

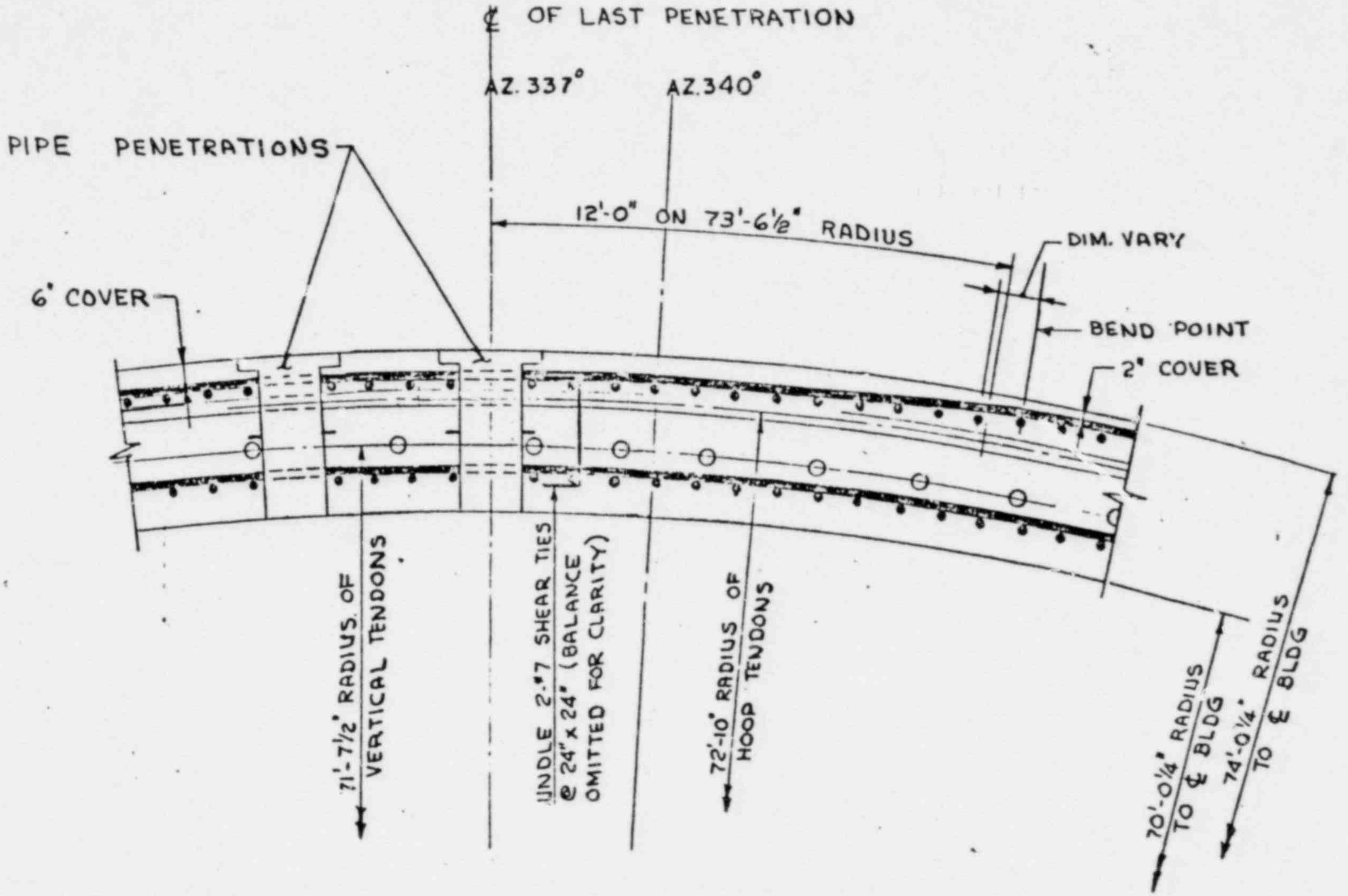
| | |
|---|---|
| Brandeis Parking Garage | Lincoln, Nebraska |
| Boston Store | Chicago, Illinois |
| Field Museum | Chicago, Illinois |
| Grant Park Garage | Chicago, Illinois |
| Federal Post Office | Phoenix, Arizona |

Also, Mr. Beasley has gained experience in legal procedures for the expert witness and has given testimony in binding arbitration.

Mr. Beasley is an Associate Member of the American Society of Civil Engineers and an Associate Member of the American Concrete Institute.

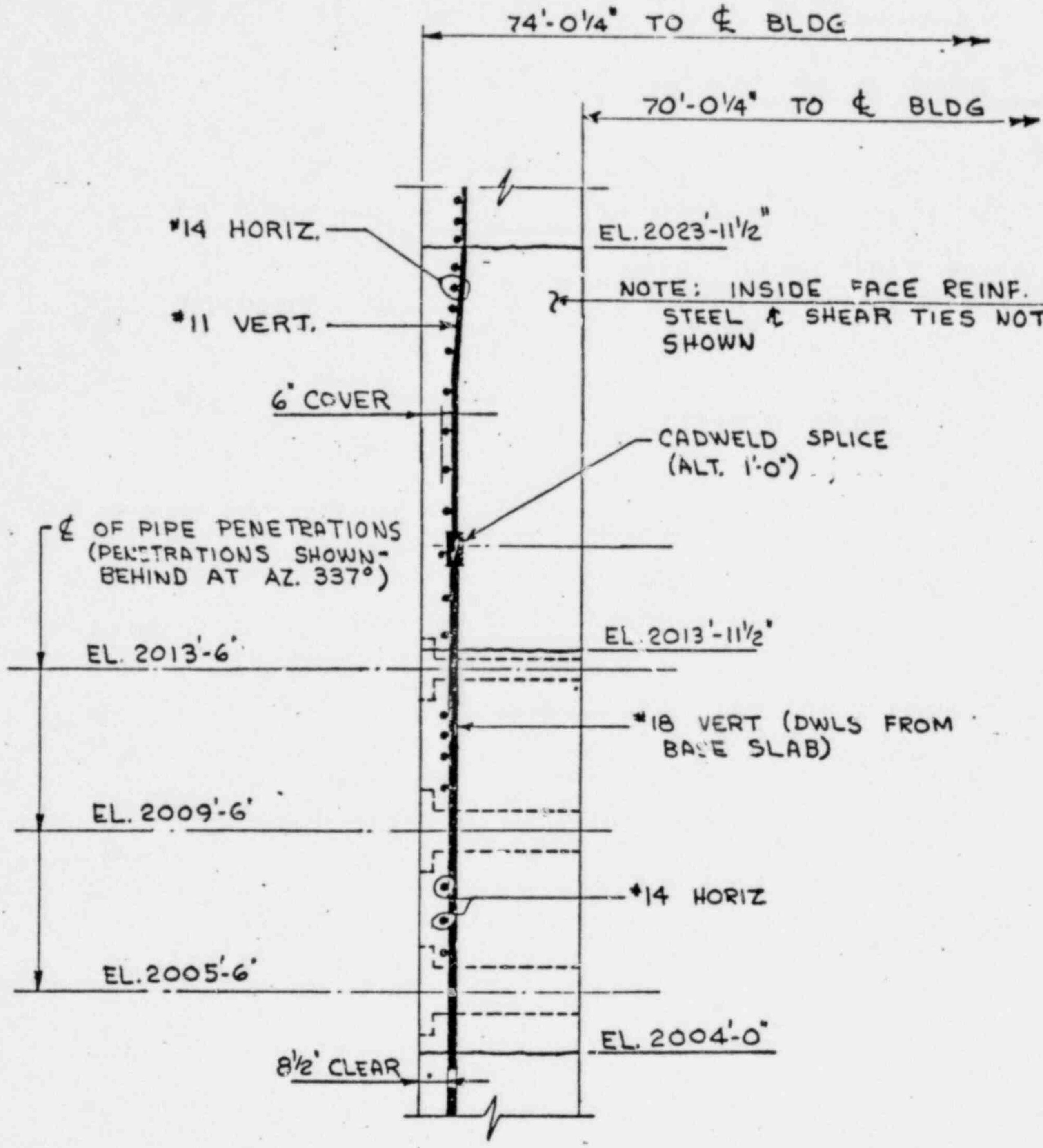
THOMAS J. ROWE

Mr. Rowe has over two years of soniscope field experience. He received his training under the supervision of Mr. Hedien and Mr. Vyskocil. He was the author of the WJE Soniscope Training Course and has conducted the course twice in recent months with the aid of Mr. Vyskocil, Mr. Hedien and Mr. James B. Ricks. Mr. Ricks was a principal developer of the original pulse-velocity instrument developed in the mid-1950's at the Portland Cement Association.




PLAN-CONTAINMENT WALL AZ 340°
(3RD LIFT)

FIGURE 72A SHEET 1 OF 2




SECTION-CONTAINMENT WALL AZ 340°
(3RD LIFT)

FIGURE 72A SHEET 2 OF 2

TABLE 95(d)

Safety-Related Preassembled Piping
Formations Installed Prior To November 2, 1979

| No. | Location | Date Issued To Field | Service |
|---|---|---|---|
| A09 | Aux. Bldg. 1974 El. Area 1 | 3/27/78 | (BG) Chemical & Volume Control |
| A12 | Aux. Bldg. 1974 El. Area 1 | 3/27/78 | (BG) Chemical & Volume Control |
| A15 | Aux. Bldg. 2000 El. Area 1 | 9/19/78 | (BG) Chemical & Volume Control |
| A16 | Aux. Bldg. 1974 El. Area 2 | 4/11/78 | (EM) High Pressure Cooling Injection |
| A17 | Aux. Bldg. 1974 El. Area 1 | 4/11/78 | (EM) High Pressure Cooling Injection |
| A31 | Aux. Bldg. 1974 El. Area 3 | 9/28/78 | (BG) Chemical & Volume Control |
| A32 | Aux. Bldg. 1974 El. Area 3 | 5/09/78 | (BG) Chemical & Volume Control |
| A33 | Aux. Bldg. 2000 El. Area 2 | 9/21/78 | (BG) Chemical & Volume Control |
| A34 | Aux. Bldg. 1974 El. Area 2 | 5/12/78 | (BG) Chemical & Volume Control |
| A35 | Aux. Bldg. 1974 El. Area 1 | 5/12/78 | (BG) Chemical & Volume Control |
| A39 | Aux. Bldg. 1974 El. Area 2 | 6/15/78 | (BG) Chemical & Volume Control |
| A43 | Aux. Bldg. 2000 El. Area 5 | 12/06/78 | (AL) Auxiliary Feedwater |
| A44 | Aux. Bldg. 2000 El. Area 5 | 4/06/79 | (AL) Auxiliary Feedwater |
| A45 | Aux. Bldg. 2000 El. Area 5 | 12/06/78 | (AL) Auxiliary Feedwater |
| A46 | Aux. Bldg. 2000 El. Area 5 | 2/23/78 | (AL) Auxiliary Feedwater |
| A47 | Aux. Bldg. 2000 El. Area 5 | 9/06/78 | (AL) Auxiliary Feedwater |
| A48 | Aux. Bldg. 2000 El. Area 5 | 2/23/79 | (AL) Auxiliary Feedwater |
| A49 | Aux. Bldg. 2000 El. Area 5 | 9/06/78 | (AL) Auxiliary Feedwater |
| A51 | Aux. Bldg. 2000 El. Area 5 | 9/06/78 | (AL) Auxiliary Feedwater |
| A52 | Aux. Bldg. 2000 El. Area 5 | 4/06/78 | (AL) Auxiliary Feedwater |

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| UNION ELECTRIC COMPANY | ) | Docket No. STN 50-483 OL |
| | ) | |
| (Callaway Plant, Unit I) | ) | |

AFFIDAVIT OF ROBERT J. SCHUKAI

City of Saint Louis)
: SS
State of Missouri )

Robert J. Schukai, being duly sworn according to law, deposes and says that he is General Manager-Engineering of Union Electric Company; that the answers contained in "Applicant's Answers to Interrogatories of Joint Intervenors (First Set)" are true and correct to the best of his information, knowledge and belief; and that the sources of his information are the officers, employees, agents and contractors of Union Electric Company.

Robert J. Schukai

Robert J. Schukai

Sworn to and subscribed before me this 10th day of July, 1981.

Charles A. Bremer

Notary Public

My commission expires 5/8/82.

CHARLES A. BREMER
NOTARY PUBLIC, STATE OF MISSOURI
MY COMMISSION EXPIRES 5/8/82
ST. LOUIS COUNTY