7/16/81

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 1)	)	

RESPONSE TO NRC STAFF INTERROGATORIES AND REQUEST FOR PRODUCTION OF DOCUMENTS TO COALITION FOR THE ENVIRONMENT, ST. LOUIS REGION; MISSOURIANS FOR SAFE ENERGY; AND CRAWDAD ALLIANCE

Joint Intervenors submit the following Response to NRC Staff Interrogatories and Request for Production of Documents to Coalition for the Environment, St. Louis Region; Missourians for Safe Energy; and Crawdad Alliance. All documents identified, unless otherwise indicated, are in the possession and/or control of Kenneth M. Chackes, Attorney for Joint Intervenors and will be made available for inspection and/or copying upon reasonable request. The answers provided below contain all of the information presently available to Joint Intervenors. Additional information that would be responsive to these interrogatories is presently being sought via Joint Intervenors' discovery to Union Electric and the NRC Staff. For all questions not answered Joint Intervenors have no responsive information. Where "not determined" is provided in response to questions dealing with identification of our witnesses, Joint Intervenors mean that at present we do not plan to call any witnesses. If Joint Intervenors determine that witnesses will be called their identities will be immediately disclosed to the Applicant and NRC Staff. Joint Intervenors are unable to answer many of the questions pertaining to Contention No. 2 because of the unavailability of the technical specifications, and the FES and SER.

Q-1. (a) See Response to Applicant's Interrogatories Contention 2, A-1 through A-5, and G.

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(b) The time frame would be for the life of the plant.

(c) See Q-1(a) above.

(d) See Response to Applicant's Interrogatories Contention 2, A-11.

(e) The Missouri River Basin Commission and U.S. Army Corps of Engineers are presently studying the proposed water-using projection in the Missouri River basin.

(f) Unless release rates are managed as described in response to Q-1(c) concentrations of radionuclides in the river cross-section would not be uniform at less than infinite distances downstream. Unless the inhomogeneities are accurately known at specific finite distances downstream from the release and unless the residual times at each point in the cross section are known for the fish, sampled predictions of bioconcentration of radionuclides in fish cannot be made.

(g) See Q-1(a) above, and Response to Applicant's Interrogatories A-11 through 16.

(h) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) See Response to Applicant's General Interrogatories No. 1 and documents identified in Response to Applicant's Interrogatories Contention 2, A and G.

Q-2. (a) See Response to Applicant's Interrogatories Contention 2, A-1(5), A-8 through A-17, A-26 through A-31 and Response to Contention 2B.

(b) See Q-2(a) above.

(c) Not determined.

(d) The Joint Intervenors know of no drinking water intake within .05 miles of the Callaway plant discharge pipe as of this time.

(e) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) See Q-2(a) above and documents identified therein.

Q-4. (a) Joint Intervenors do not possess the resources to provide an answer to this interrogatory. A detailed explanation of the technical basis for this answer would involve a thorough review of the NRC and scientific literature on:

(1) The occurrence and magnitude of unscheduled radioactive releases.

(2) A comparison of the "predictions" versus "experiences" for all operating reactors in their lifetime.

(3) A review of criticism and failures of simplistic gaussian diffusion models which are considered good if they are within 50% to 200% of actual values. Refer to extensive air pollution literature in the Journal of the Air Pollution Control Association, Atmospheric Environment, Environmental Science and Technology, Bulletin of the American Meteorological Society, etc.

(b) Release rates which should be utilized for accurate prediction of radioactive material to the atmosphere should include the routine releases cited by applicant plus a worst case average release over the last 5 years of the sporadic, "non-routine" experienced releases cited in many NRC releases. A survey of the total radiation released from such "non-routine" releases should be completed for operating

PWR's using the same isotopes assumed for Applicant's routine releases. These results from the past 5 years should be scaled to an average annual isotopic release from non-routine releases and be added to Applicant's current prediction to improve these predictions. (Of course this approach assumes all "non-routine" releases are presently reported — something which has yet to be proven to the satisfaction of the Intervenors.) Their predictions would still not be accurate since they can only assume future conditions and not know them. Examples of sporadic releases and their usual unpredictability in time and magnitude appear in response to Q-4(d) below.

(c) Meteorological considerations at Callaway which will make U.E. "unable to predict accurately the dispersion of radioactive materials" are any non-gaussian well-mixed conditions. Examples would be winter stable conditions with a low-lying inversion layer coupled with marked wind shear across the inversion layer. During convective erosion of such a layer no accurate prediction could be made with U.E.'s current level of modeling sophistication. As a graphic example of a condition similar to those mentioned above refer to photograph attached as Exhibit A.

(d) Applicant can only make predictions based on ideal, engineering design assumptions of releases. To "accurately predict the amount or discharge rate. . . into the atmosphere," any deviation from normal must be programmed in advance into Applicant's assumptions. To support the Applicant's inability to "accurately predict" radioactive discharges to the atmosphere, we provide examples from the NRC that show the sporadic nature of many releases. See Response to Applicant's Interrogatories. . . Contention 2.

As can be seen from the cited detailed experiences at Westinghouse plants, summarized from the NRC's "Licensee Event Report (LER) Output on PWR Events Involving Released Activity from 1969 to the Present" (through March 10, 1981), sporadic releases can be of significant magnitude compared to routine releases and

hence make it impossible for Applicant to "accurately predict" atmospheric radioactive discharge.

(e) Meteorological considerations used by U.E. are mostly annual average data. We see no indication that U.E. has considered more than traditional and often inadequate gaussian dispersion conditions. Their main claim to fame is use of the "straight line method" (FSAR, Callaway Site Addendum, Section 11.3.3.4.1) or, more accurately, "monotonically decreasing" concentraton approximation (NRC Regulatory Guide 1.111, p. 6) They do not consider that measured plumes often are not gaussian, that models of complex sources (such as the BLP model for complex sources in the alumininum industry) can readily show plumes impacting for short distances before becoming elevated above ground level and hence free of ground removal mechanisms, that measured plume dispersion vertically and horizontally often does not follow Pasquill-Gifford stability curvese.

These are normal meteorological occurrences documented in the literature. Since U.E.'s models are too simplistic to take into account these meteorological conditions, their predictability is highly suspect. If the meteorological conditons can cause plume lifting as in the case of the aluminum industry (where tracer studies confirmed the BLP model as reported at the Air Pollution Control Association (APCA) Specialty Conference on Modeling Dispersion from Complex Sources, held in St. Louis on April 7-9, 1981), then the fallout rate or removal rate will be inaccurate. Hence more radioactive material will be transported downwind than the Applicant predicts.

(f) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) See Q-4 above, and Response to Applicant's General Interrogatories No. 1 and Contention 2-A and 2-D documents identified therein.

Q-5. (a) The statement made by the Intervenors (Contention 2E) is not an assertion. It is a fact that when radioactive levels drop below the level of detection of any monitoring equipment, the equipment can no longer detect the radioactivity. Furthermore, no continuous monitoring of low-level beta radiation is planned. A number of radionuclides emit no high-energy gamma (e.g., H-3) and thus cannot be detected by continuous gamma monitoring. Batch release sampling as required by Reg. Guide 1.21 composited and analyzed on a monthly basis will be inadequate to detect "spikes" of radioactivity exceeding 40 CFR 190 or even 10 CFR 20 limits.

(b) U.E. has given the level of detection of the commercial monitoring equipment, in SNUPPS Table 11.5-2 for liquid wastes. 1×10^{-6} μCi/cc. (1000 pCi/l). As stated in response to Q-5(a), no continuous monitoring device will be used for low-level beta.

(c) Information available to the Intervenors at present is inadequate to make such calculations. Furthermore, since no continuous monitoring of low-level beta is proposed, there is no minimum level of detection for a number of radionuclides without high energy gamma emissions. Therefore, any dose level is possible from these radionuclides.

(d) For low-level beta emissions any release rate is possible since grab samples might not be analyzed for periods as long as one month after release.

(e) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) See documents cited in Q-5(a)-(d) above.

Q-6. (a) According to SNUPPS FSAR Table 12.2-7, Revision 3 of April 1981, only four isotopes are listed in the spent fuel pool water inventory (cobalt-58 and 60 and cesium-134 and 137). "Other isotopes will be present in much lower concentrations." This is a totally inadequate survey of probable fission and activation products that will be present as the result of leakage from fuel rods with cladding irradiated and embrittled for at least three years at the time of initial storage (including some with pinholes, fissures, faulty welds, etc.), plus incremental leakage as the rods continue deteriorating during prolonged storage. The following "principal fission products released to fuel pool waters" were cited by A.G. Johnson, Jr., of Battelle Pacific Northwest Laboratories: iodine-131, tritium, strontium-90, cerium-144, rubidium and rhodium-106, zirconium and niobium-95, xenon-133 and krypton-85, as well as the cesium isotopes mentioned by the Applicant. ("Impacts of Reactor-Induced Defects on Spent Fuel Storage," from Storage of Spent Fuel Elements, proceedings of NEA Seminar in Madrid, June 1978, p. 239).

Furthermore, according to Regulatory Guide 1.112 and SNUPPS FSAR Appendix 11.1A, the "Parameters for Calculation of Source Terms for Expected Radioactive Concentration and Releases" do not include radioactive effluent from the spent fuel pool, even though leakage of as much as a gallon a minute could continue from the pool for an hour before a monitor would respond. (SNUPPS FSAR, p. 9.3-17 and 9.1-16, 17).

(b) Although some parameters are given for the spent fuel pool maintenance and cleanup system in Section 9.1 of SNUPPS FSAR, the Intervenors cannot estimate the release rates of radionuclides from the spent fuel pool (gaseous and liquid) without knowing in detail the following: (1) the effect of the water chemistry on the cladding; (2) the number of stored fuel assemblies; (3) the duration of storage in the pool; (4) the spacing of the fuel assemblies; (5) the average condition of the cladding initially and over time; (6) the condition of the racks; (7) burnup time of the rods in the reactor vessel; and (8) the rate of buildup of tritium in the spent fuel pool water caused by diffusion through the cladding and from the addition of boron. Data published by the NRC each month in the "Operating Units Status Report," on the Status of Spent Fuel Storage Capability, indicate that increasing numbers of licensees are being granted permission to expand the storage capacity of the pools through compaction (e.g., NUREG-0020, Vol. 4, No. 12). Because no commercial reprocessing plant will be capable of reprocessing the fuel rods irradiated at the Callaway Plant within the requisite number of years specified by the operating license in question, the omission of the spent fuel pool from the plant's radwaste effluent parameters makes the annual emission projections all the more unreliable.

(c) (1) "Steam generator tube deterioration" refers to a decrease in the integrity of the steam generator tubes which contain the highly radioactive water from the reactor's primary system, and which isolate the radioactive water from the secondary system. The deterioration may come in the form of corrosion resulting in steam generator tube wall thinning. A major corrosion-related phenomenon occurs as a result of a build-up of support plate corrosion products in the annulus between the tubes and the support plates. This build-up eventually causes a diametral reduction of tubes, called "denting", and

deformation of the tube support plates. This phenomenon can lead to other problems, including stress corrosion cracking, leaks at the tube/support plate intersection, and U-bend section cracking of tubes which can be highly stressed because of support plate deformation. The defects in tube integrity can cause leakages around the dented areas resulting in highly radioactive primary water leaking into the secondary water and subsequently being discharged into the Missouri River. It is not possible to quantify the amount of such releases without having actual plant operating data.

References:

Identification of Unresolved Safety Issues Relating to Nuclear Power Plants Report to Congress, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NUREG-0510), January 1979;

Nero, Anthony V., Jr., A Guidebook to Nuclear Reactors, University of California Press, 1979.

(2) "Decontamination procedures" refer to the need now recognized by nuclear power plant operators to remove the radioactive corrosion products (metal oxides) which accumulate along the piping and other component parts of the primary coolant system (known as "crud") and the secondary coolant system (known as "green grunge"). The corrosion products cause the radiation fields within certain areas of nuclear power plants to become too hot for workers and inspectors to perform necessary tasks. ("Approved Task Action Plans for Category A Generic Activities," NUREG-0371, Appendix E, pp. 164-174;

"Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," NUREG-05'0, pp. 20, 21, 42-44; "Primary System Shutdown Radiation Levels at Nuclear Power Generating Stations," Sawochka, S.G., et al., for Electric Power Research Institute, EPRI 404-2, December 1975). In order to clean out the corrosion products, thereby reducing the radiation fields and improving the efficiency of the plant, the use of chemical decontaminants (solvents) on a periodic or ongoing basis is being considered. ("Final EIS related to Primary Cooling System Chemical Decontamination at Dresden Nuclear Power Station," NUREG-0686, October 1980).

(d) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) See Q-6(a)-(c).

Q-7. (a) (1) See NRC Report No. 50-483/80-14: Page 5: Visual inspection of plates installed prior to June 9, 1977: manually embedded plates supporting structural steel framing substantially loaded by floor slab dead load with no sign of distress. No live load tests were done.

(2) Page 5: "machine welded embedded plates, some of which were loaded with support attachments and others not yet loaded, were observed to be fully intact with no sign of distress." The inspection and testing were inadequate because

some of the plates weren't loaded at all, others with only dead loads.

(3) Pages 6-7: "six plates were loaded to allowable design load without plate failure or plate deflection of more than 1/4 inch. . ." Six plates is a rather small sample. We question whether this constitutes adequate testing; the lack of live load testing; and the lack of testing to determine whether plates will hold up under additional stress such as earthquakes.

(4) Pages 7-8: Eighty plates inspected during July and August, 1977 did not meet the specification requirements of C-131, Revision 9. However, UE indicated that the welds inspected had an undersize not exceeding 1/8". That is, UE is making further revisions and justifying it by calculating that 1/8" undersize is acceptable.

(5) Pages 8-9: Forty-seven plates originally rejected by Daniel reinspected by a team of UE, Bechtel, and Daniel inspectors - 39 did not meet the requirements of Specification C-131, Rev. 9 or AWS including the 4 exceptions. These plates did indicate poor workmanship characteristics. However, they were considered adequate because none exceed 1/8" undersize or 1/16" undercut. Tests were performed to demonstrate that they provide adequate structural integrity: six rods bend tested and six rods tension tested. We believevve this is insufficient testing to insure that welds that did not meet specifications are adequate.

(b) See Q-7(a) above.

(c) See Q-7(a) above.

(d) Most inspection and test procedures seem to have been aimed at plates not installed (see NRC Report No. 50-483/80-14, page 5 ". . . the licensee initiated a 100% reinspection of plates not installed as of that date. . ."). Testing performed on the installed plates is not as extensive as that on those not installed. They were submitted to visual inspection (page 5 - "During the NRC inspection on June 10-12, 1980, a visual inspection of embedded plates installed in concrete prior to June 9, 1977 was performed), and to minimal testing (page 6 "Six plates were laoded to allowable design load. . ."). This does not provide assurance that none of the installed plates have defective welding.

(e) NRC Report No. 50-483/80-14, page 7 describes Revision 9 of Specification C-131; ULNRC 349, question 2, asks why Revision 9 is acceptable; UE responds in ULNRC-354. The NRC does not show any further objections to these deviations, so we assume that they give their approval.

(f) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) See Q-7(a)-(e) above.

Q-8. (a) See Response to Applicant's Interrogatories . . . Contention No. 1, Nos. 1B-6 and 1B-7.

(b) See Q-8(a) above.

(c) See Response to Applicant's Interrogatories. . . Contention No. 1, No. 1B-7.

(d) NRC Report No. 50-483/78-03, page 3: "NCR 2-2081-C-A was written for the crack in question along with other cracks in the control building walls. This NCR was superceded by NCR 2-2173-C-A after all the cracks documented in the

former NCR had been evaluated. Evaluation of the crack in question was to 'use as is', as it was an acceptable crack caused by normal concrete shrinkage. This item is closed."

(1) We question the fact that the first NCR was "superceded by the second.

(2) Even though this crack may be due to normal causes, this does not necessarily mean that it is acceptable.

(e) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) NRC Report No. 50-483/77-06.

NRC Report No. 50-483/78-01.

NRC Report No. 50-483/78-03.

Letters: Kay Drey to Ernst Volgenau, April 20, 1978; Volgenau to Drey, May 21, 1978.

Q-9. (a) NRC Report No. 50-483/77-07, page 13 states: "During a visit to honeycomb areas of the tendon gal'ery. . ., NRC inspector observed repair activity in prog·ess even though the NCR covering the repair work was unapproved by the A-E. . . The dry-pack grout was not being tested as required by Specification C-103 because the current revision of Speci icr ıon C-191, prescribing the necessary tests to be performed, failed to includ ·ch a test." Thus, some repairs were performed using untested materials. Se also resp ıse to Q-9(b) below.

(b) NRC Report No. 50-483/77-07, pages 12-13, states: "Based on 25% sample, the results of this soniso investigation indicate that intern l honeycomb probably does not occur in the base slab, except at those 19 areas where honeycomb

was visible." We question whether a 25% sample is really adequate assurance that no more voids exist and whether "probably" is a really satisfactory answer.

(c) Yes. See Q-9(a) and (b).

(d) Regarding NRC Report No. 50-483/80-30:

(1) We question why it took from August to November to chip away the concrete on 4 areas of imperfection.

(2) We question why the engineers in QC and construction engineering did not inform their superiors of the inital NCR concerning this matter - especially after it was later judged to be "potentially reportable per 10 CFR 50.55(e)."

(3) ". . . licensee personnel attributed the occurrance of the imperfections to the complex nature of those portions of the dome slab where the imperfections had occurred. . . the RIII inspector concurred with the licensee's conclusions. . ." This judgment is questionable, especially after the subsequent appearance of three additional void areas for which "there appeared no plausible explanation."

(e) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) (1) NRC Report No. 50-483/80-30.

(2) NRC Report No. 50-483/80-31.

(3) ULNRC-406.

(4) NRC Report No. 50-483/77-04.

(5) NRC Report No. 50-483/77-06.

(6) CPPR-139.

(7) NRC Report No. 50-483/77-07.

(8) NRC Report No. 50-483/77-09.

(9) NRC Report No. 50-483/78-01.

(10) NRC Report No. 50-483/78-02.

(11) NRC Report No. 50-483/80-16.

(12) Letter James Keppler to Kay Drey April 4, 1980.

Q-10. (a) The minutes of the January 23, 1978 meeting (enclosure with 78-01), pp. 7-8 state that the minimum requirement represents "the absolute minimum cover to assure corrosion control," and the maximum requirement represents "the absolute maximum depth in order to provide surface crack control for the concrete containment. . ." If these requirements are not adhered to (as they aren't in all cases below the 6th lift), this means that corrosion and cracking will not be adequately controlled. This could impair the function of the concrete containment.

(b) Base slab of Diesel Generator Building has reinforcing steel 3 1/2" below the surface, contrary to the 1" specification.

(c) The requirements referred to are the minimum concrete coverage of 2" for #6 through #18 bars (Section CC-3533.1 of Appendix C to BC-TOP-5) and the maximum of t/5 (9.6" in this case) to reinforcing steel that is considered face reinforcement (Section CC-3534 of Appendix C to BC-TOP-5). See minutes of January 23, 1978 meeting, pp. 7-8.

(d) Yes.

(e) Regarding No. 77-11, p. 10: "It was indicated that this minimum could be reduced by one-third to give an absolute minimum concrete cover of one and one-third inches." The report does not state by whom was it "indicated." NRC inspectors did not accept this interpretation yet the NRC indicated that "it would be acceptable if the cover requirements were fully met in the area of the 6th lift, utilizing the 5th lift as a transition area." (p. 11). Also, the NRC apparently gave in to U.E.'s

objection to the requested chipping out of two areas where cover problems could be located (p. 10). Why did U.E. object to this request?

Regarding No. 78-01: The NRC states it will impose requirements starting at the 6th lift. At this point, work was still going on on the 4th lift. Problems surely could have been cleared up by the 5th lift. At pages 10-11: (item 12 on p. 10): concrete cover was 12" to a #11 bar; page 11 states that this item "fell into a category in which the licensee felt that they could allow the 1 1/2" placement tolerance on the 9.6" maximum cover. . . which would allow a maximum cover of 11.1 inches." This measurement exceeds requirements even with placement tolerance.

(f) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) (1) Letter Kay Drey to Thomas Eagleton and John Danforth, December 6, 1977.

(2) NRC Report No. 50-483/77-11.

(3) Letter Kay Drey to James Keppler January 12, 1978.

(4) NRC Report No. 50-483/77-10.

(5) NRC Report No. 50-483/78-01 and enclosure: minutes of January 23, 1978 meeting.

(6) SNUPPS letter February 13, 1978 (Petrick to Case).

(7) Letter Kay Drey to Ernst Volgenau April 20, 1978.

(8) Letter from Olin Parr (NRC) August 11, 1978.

Q-11. (a) The ASME Code provides assurance of structural integrity and quality commensurate with the relative importance assigned to individual items of a nuclear power plant. The ASME Code is one of the primary instruments used to fulfill the

requirements of 10 CFR Part 50, Appendix B and is frequently cited in the Final Safety Analysis Report submitted by Applicant Union Electric. If safety-related piping were not in conformity to ASME Codes, failures in critical systems could occur and could compromise the safe operation of the Callaway Plant.

For example, there were substantial noncompliances to specifications and ASME Code requirements in pipe formations fabricated by Gulf & Western, Paola, Kansas. In some cases, these formations were installed at the Callaway Plant with visible weld discrepancies involving incomplete fusion, surface porosity and improper weld profiles. Concern regarding the visible weld discrepancies was not shown at the Callaway Plant until Wolf Creek indicated concern. The weld defects could have resulted in failures, degrading the functionability of critical systems to the extent that safe shut down capability is compromised.

(b) The evaluation and acceptance of SA-312 pipe manufactured by Youngstown Welding and Engineering, prior to mid-November, 1979, was apparently based on a Bechtel report, dated June 1979, and titled, "Report on Investigation of Weld Imperfections in ASME SA-312 Double Welded Austenitic Stainless Steel Pipe for Compliance with NRC I & E Bulletin 79-03." Many of the "actions to be taken" in 79-03A seem to be adopted from the Bechtel report. However, the report is not consistent with the ASME Code and in some cases can be considered perverse to the Code.

The following are some examples which illuminate the point:

(1) The report does not substantiate the statement, on page 2 of the report, that, "All of the mechanical property requirements of ASME SA-312 were met with CLP up to 26 percent." ASME Section II, SA-312 paragraph 10 lists three required mechanical tests to determine acceptable mechanical properties. They are a transverse or longitudinal tension test, a flattening test, and a hydrostatic test. The Bechtel report gives extensive

test data for tension and hydrostatic tests, but omits the flattening test required by SA-312, paragraph 10.2. If pipe with CLP, cannot meet the material sepcifi ations it is not suited for use in Class 2 or Class 3 systems. (See ASME Section III paragraphs NC 2551 (a) and ND 2551).

(2) The Bechtel report recommends an etch test to determine the extent of CLP defects, but does not cite an established code or standard for conducting the test. An etch test is not recognized in the ASME Code, Section III or Section V, Nondestructive Examination, sometimes referenced in Section III. The liquid penetrant method of examination could detect CLP and has standards of acceptance for all classes of materials in Section III. Without prescribed procedures or instructions the etch test cannot reliably assure the quality of the pipe being examined.

(3) The report recommends that, "Piping systems subject to design hoop stress of less than 85% of allowable stress need not be examined." In a SNUPPS letter to Region I, dated October 5, 1979 the "85% allowable stress" criterion is explained with the following statement, "All piping systems containing Youngstown materials are being analyzed and, as suggested by Bechtel, those systems subject to design hoop stresses of less than 85% of ASME Code allowables will be installed without restriction. This 85% value is cited in the enclosed Bechtel report and is based upon ASME Section III, Division 1 allowable stress values for austenitic stainless steels. Specifically, Note 3 to Table I-7.2 of Appendix I of the Code, which is applicable

to all grades of stainless steel within the AS-312 material specification, provides a series of "efffficiency factors" for longitudinally welded pipe. The efficiency factors are percentages of the stress values and depend upon the type of longitudinal weld joint. For welds without filler metal pipe of the SA-312 type, the efficiency factor is 85% when no volumetric examination of the longitudinal weld is performed."

Table I-7.2 of Appendix I (cited above) is for determining allowable stress values for Class 2 and Class 3 piping. However, the Section III standards for defect determination cited in the report are from Subsection NB - Class 1 Components. The report recommendation does not differentiate between classes and presumably is to be used for all classes of material. This is not consistent with Section III, paragraph NA-2142.1 which states, "Design loading for Class 1 components and support shall be as defined in NB-3112 and NF-3112."

Note 3 does not substantiate the statement in the letter that, "the efficiency factor is 85% when no volumetric examination of the longitudinal weld is performed." Note 3 provides a weld efficiency factor of 1.00 for Class 2, SA-312 piping examined by the ultrasonic or eddy current method. The other two methods of examination allowed in NC 2551 (b) would have an efficiency factor of .85. Note 3 does not mention volumetric examination and even requires a .85 efficiency factor for Class 2, SA-312 piping volumetrically examined by the radiographic method. Note 3 does not resolve

the examination problems associated with the requirements of NC 2551 (b) and related sections of NB 2550 where radiographic and ultrasonic techniques are not able to detect, amounts of CLP, significant to the safe operation of a nuclear plant. The "85% allowable stress" criterrion is a substitute assurance of structural integrity and quality that is not consistent with Section III and may be considered perverse to Section III.

(c) The following:

(1) The problems presented in IE Bulletin 79-03 and the June 1979, Bechtel report, relating to the inability of radiographic and ultrasonic techniques to detect CLP, are generic to the fusion welding of double welded butt joints. The only significant diffference between detection and non-detection of CLP defects is the existence of a discrete gap between the unfused faces of the weld joint. This difference is not a function of the SA-312 material specification but is associated with the welding process procedure. The Bechtel report and 79-03A limited the problem's investigation to SA-312 material, when it affects all material specifications examined in accordance with NB-2550 or NC-2550.

(2) The evaluation and acceptance of SA-312 piping with clouded structural integrity and quality as explained in the answer to question Q-II(b) is unacceptable.

(3) Many unaccepetable portions of NRC Report No. 50-483/80-10 have been clarified in NRC Report No. 50-483/81-04 and are now acceptable. However, the cause of the welding defects reported in Report No. 50-483/81-04 has not been determined.

If the weld defects were caused by the melting through of the inside pass by a pass made from the outside and the weld puddle became contaminated by the air inside the pipe, the mechanical properties of the weld metal may be adversely affected. Also, it has not been determined if this is a single isolated incident or a common event for the manufacturer involved, casting doubt on the quality of other SA-358 piping.

(d) (a) See Objection No. 1.

(b) See Objection No. 1.

(c) Not determined.

(d) No answer required.

(e) No answer required.

(f) See documents identified in Q-11(a), (b) and (c).

Q-12 (a) Improper inspection technique refers to the radiographic technique used by Gulf & Western to examine 32 piping ɔrmations. Noncompliances to specifications and ASME ode requirements involved improper film density, multiple penetrameter images, incorrect penetrameter placements and surface indications inhibiting radiographic interpretation. (See Bechtel's "Final Report on Gulf & Western Preassembled Formations for Callaway Plant No. 1 (Union Electric) and Wolf Creek (Kansas Gas and Electric)).

(b) 32 piping formations are involved. When the radiographs of 3 formations were reviewed, 22 of 52 welds indicated rejectable defects including incomplete penetration, incomplete fusion and slagg inclusions. Similar weld quality is indicated in the other formations. The full number of defective welds has not been presented in public documents. (See ibid.).

(c) (a) See Objection No. 1.

(b) See Objection No. 1.

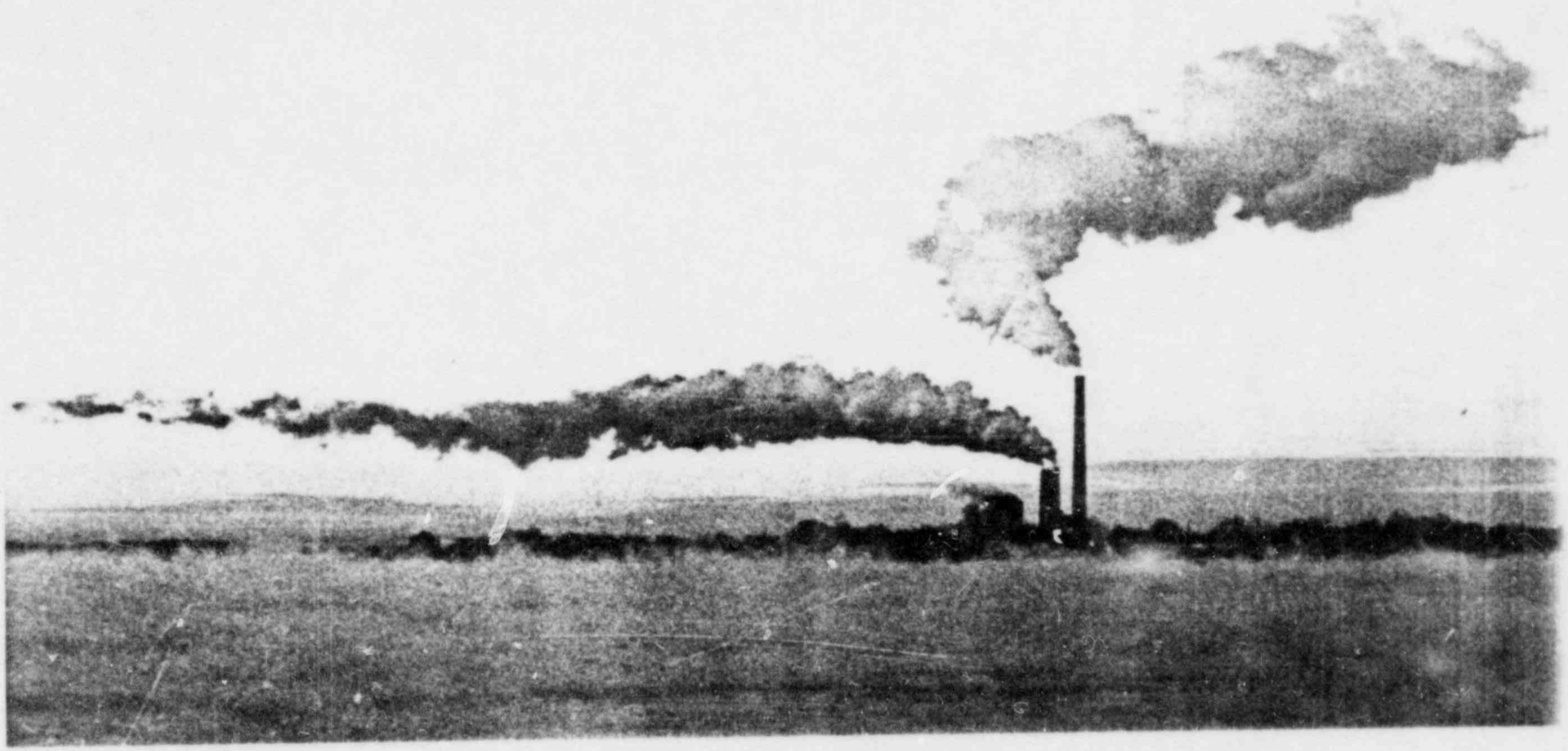
(c) Not determined.

(d) No Answer Required.

(e) No Answer Required.

(f) See documents identified in Q-12(a) and (b).

Q-13. See Objection No. 1.



Salem Power Plant: 1971 or 1972

Reproductions must bear the legend:

[Photo by RALPH Turcotte
Beverly (Mass.) Times]

EXHIBIT A
(No. Q-4(c))

Kenneth M. Chackes, attorney for Joint Intervenors Coalition for the Environment, St. Louis Region; Missourians for Safe Energy; and Crawdad Alliance, and authorized as their agent for the purpose of answering the above interrogatories, hereby states to the best of his knowledge, information and belief that the responses provided above are true and contain such information as is presently available to Joint Intervenors.

Kenneth M. Chackes

Kenneth M.Chackes

Subscribed and sworn to before me this 16 day of July, 1981.

Mary Anne Sldey

Notary Public

My Commission Expires: 5/18/82

CHACKES AND HOARE

Kenneth M. Chackes

Kenneth M. Chackes
Attorneys for Joint Intervenors
314 North Broadway
St. Louis, MO 63102
314/241-7961

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 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the Response to NRC Staff Interrogatories and Request for Production of Documents to Coalition for the Environment, St. Louis Region; Missourians for Safe Energy; and Crawdad Alliance have been served on the following by deposit in the United States mail this 17 day of July, 1981.

James P. Gleason, Esq., Chairman
Atomic Safety and Licensing Board
513 Gilmoure Drive
Silver Spring, MD 20901

Mr. Glenn O. Bright
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Jerry R. Kline
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Thomas A. Baxter, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N.W.
Washington, D.C. 20036

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Roy P. Lessy, Jr., Esq.
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Kenneth M. Chackes

Kenneth M. Chackes
CHACKES AND HOARE