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RELATED CORRESPONDENCE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

In the Matter of:	)	
	)	Docket No. 50-454 OL
COMMONWEALTH EDISON COMPANY	)	50-455 OL
	)	
(Byron Nuclear Power Station, Units 1 and 2)	)	

'84 AGO 16 P12:36

DIRECT TESTIMONY OF CHARLES C. STOKES
ON BYRON REINSPECTION PROGRAM

Q1: Please state your name and present employment.

A1: My name is Charles Cleveland Stokes. I am a nuclear engineering consultant. My partner and I do consulting work in the nuclear construction industry under the name P/S Associates.

Q2: Please describe P/S Associates.

A2: P/S Associates is a newly formed firm that offers consulting services to those differing entities involved in the nuclear construction industry. Our initial work has been on behalf of intervenors in NRC proceedings relating to the Diablo Canyon and Byron nuclear power plants.

Q3: Please describe your job responsibilities.

A3: I am a technical consultant in utility production facilities from the standpoints of design calculation, code and federal requirements, and quality assurance. I also design and review existing structures and mechanical components.

Q4: Please describe your education and professional background.

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A4: I am a member of the National Society of Professional Engineers and a registered Professional Engineer in Alabama, Florida and Georgia. I began my career as a co-op student employee with the Civil-Structural Department of Southern Company Services from November 1972 to June 1975. I was assigned various duties, primarily those of a draftsman and detailer for the Fossil-Hydro Concrete, Structural Steel, and Nuclear Concrete Departments. I worked on Farley Nuclear Plant Unit 1 and 2 miscellaneous outdoor concrete structures and HVAC, and Miller Steam Plant concrete base slabs. I detailed reinforcing and bolted connections.

I graduated in May 1975 with a BCE degree from Auburn University. I took a job as an Assistant Engineer with Southern Company Services Civil-Structural Department. In this position, I designed outdoor structures on Miller Steam Plant, and designed sub-structure concrete and checked super-structure calculations on Harris Dam. I held this position from June 1975 to July 1987, when I was promoted to an Engineer II. In this position, I was Civil Material Coordinator and designed miscellaneous items on Vogtle Nuclear Plant, performed 79-02 and 79-14 analysis on Farley Nuclear Plant Unit 1, and redesigned the precipitator structural steel on Miller Steam Plant, designed structural steel for coal conveyor on Schereer Fossil Plant, and wrote two specifications (one for modifications to the Reactor Heat Discharge System on the Hatch Nuclear Plant). I held this position from July 1978 to May 1980.

In May 1980, I graduated from the Birmingham School of Law with a Juris Doctorate degree. At this time, my resume was submitted to Bechtel Power Corporation's Civil Structural group in Gaithersburg, Maryland working on the Davis-Besse Nuclear Plant. I was accepted and worked from June 1980 until October 1980, performing 79-02 and 79-14 calculations. I then worked from October 1980 to May 1981 for the Nuclear Services Corporation, a division of Quadrex Corporation, on the Zimmer Nuclear Plant, during which time I was also assigned to the LaSalle Nuclear Plant. In June 1981 I began work for the Mechanical Engineering Department of the Lawrence Livermore National Laboratory as the stress analyst on the injector of the Advanced Test Accelerator (ATA). I also made some design changes to the Experimental Test Accelerator (ETA). I left the Livermore Laboratory in February 1982 to work for Reactor Controls, Inc., on Grand Gulf Unit 1 Control Rod Drive System.

In November, 1982, I was accepted by Bechtel to work on Diablo Canyon Nuclear Plant. I worked on Unit 1 until March 1983 when I was assigned to Unit 2. In October 1983, I was laid off two weeks after writing three Discrepancy Reports against both Units 1 and 2. (See my resume, Attachment 1.)

Q5: Are you familiar with the Byron Reinspection Program?

A5: Yes. I have spent more than 300 hours reading and reviewing Commonwealth Edison documents, NRC documents, Sargent & Lundy documents both at their offices and my office, weld

procedures of Hunter and Hatfield, and miscellaneous other documents on the Byron facility. I have reviewed documents reflecting many of the engineering evaluations performed by Sargent & Lundy. I have also reviewed the prefiled testimony of Edison witnesses Branch, McLaughlin and French, in addition to listening to portions of their testimony and the testimony of other Edison witnesses at the Atomic Safety and Licensing Board hearing on Byron in July of 1984. I have reviewed Edison's February 1984 Reinspection Report and the June 1984 supplement thereto, as well as engineering packages and other proprietary design documents obtained from Sargent & Lundy. In my review I have reworked a number of engineering calculations and have cross-referenced formulas used by Sargent & Lundy to perform their evaluations. I have spent some time reviewing parts of the Byron FSAR, various codes, NRC NUREGS, and researching formulas in the design criteria of Sargent & Lundy or which were referenced in Sargent & Lundy calculations on discrepancies found in the Reinspection Program.

Q6: What is the purpose of your testimony?

A6: My testimony addresses the engineering evaluations performed and the use of engineering judgment by Sargent & Lundy in its attempt to show that there are no safety-significant construction problems at Byron. The main purpose of my testimony is to suggest the need for an independent engineering analysis of the safety significance of the problems found in the Reinspection Program, as well as an independent

analysis and examination of certain hardware at Byron where evidence indicates possible safety problems.

My testimony is not intended to show conclusively that Byron is not safe to operate. A nuclear power plant is a large and complex facility requiring extensive time and resources for a conclusive engineering assessment; even an assessment limited to the Hatfield, Hunter and PTL discrepancies found in the Byron Reinspection Program, together with certain Systems Control Corporation discrepancies discussed in Edison's pre-filed testimony, would require far more time than I have had, as well as a range of engineering skills and experience including but not limited to those which I possess.

However, even in the limited time I have had to review Sargent & Lundy documents relating to the engineering evaluations, I have seen numerous indications of issues which, in my judgment, collectively require further exploration and resolution before there can be reasonable assurance that Byron is safe to operate.

For this reason, the purpose of my testimony is to suggest that there are enough signs of possible safety problems at Byron, and enough legitimate concern about Sargent & Lundy's engineering calculations and use of engineering judgment to require an independent engineering analysis of the discrepancies found in the Reinspection Program, and of certain other hardware issues, prior to a determination by this Board on whether there is reasonable assurance that Byron can be operated safely.

My testimony also includes instances in which I am unable, based on the limited review I have conducted, to agree with Sargent & Lundy's calculations or with its evaluation of safety significance. In addition I discuss miscellaneous hardware issues, which in my engineering judgment, based on the limited documents I have reviewed, do not appear to have been properly dispositioned. Lastly, I comment on those areas in which I perceive signs of possible problems, but as to which, due to time constraints and/or incomplete documentation, I have not yet reached any further conclusion other than my opinion that further review is warranted.

Q7: Why do you believe an "independent" engineering analysis is needed?

A7: A letter from Chairman Nunzio J. Palladino of the Nuclear Regulatory Commission to the Congress (Attachment D to the proposed prefiled testimony of intervenors' witness Dr. William H. Bleuel) specifies criteria for an independent design review of Diablo Canyon. In that letter, Mr. Palladino indicates that an independent design review must be done by an entity not previously involved in the activity under review. In my opinion, whenever the review is significantly judgmental, independence is needed.

Q8: Do you believe Sargent & Lundy's engineering evaluations were significantly judgmental?

A8: Yes. As illustrated in various instances cited throughout

the remainder of my testimony, an analysis of the findings of the reinspection program for safety and design significance, which is what Sargent & Lundy did, is comparable to a design analysis in the degree of judgment required.

Q9: In your opinion, why was not Sargent & Lundy's analysis "independent"?

A9: For a number of reasons. Sargent & Lundy has been and remains the architect-engineer on Byron from the plant's inception to the present date, in addition to being involved in a number of other Edison projects, such as the Braidwood plant. Sargent & Lundy thus has a direct economic stake in the outcome of the evaluations.

Moreover, my review of the firm's evaluations reveals repeated instances in which in my opinion, based on the limited documents I reviewed, it appears that Sargent & Lundy's judgments and evaluations fell short of the degree of objectivity and impartiality required of an independent review.

Sargent & Lundy has evaluated thousands of discrepant conditions and yet did not find one single item to be safety-significant.

As I reviewed a limited portion of Sargent & Lundy's calculations I found instances where the allowable stress appeared to exceed code requirements. I also found instances in which certain elements, minor by themselves, appear to have been omitted, non-conservatively, from calculations, including instances where if these factors had

been added, it appears that actual stress would have exceeded allowable stress. I also found instances where it appeared, on the basis of the calculation itself, that the equipment would fail. Yet the Sargent & Lundy evaluations with respect to each of those instances concluded that there was no safety significance.

Q10: What did your review of the design criteria (Sargent & Lundy's Structural Project Design Criteria BYRON AND BRAIDWOOD Nuclear Power Station Units 1 &2 (DC-ST-03-BY/BR) REV. 12) for Byron reveal?

A10: The design criteria were one the standards guiding calculations of design significance in the engineering evaluation of Reinspection Program discrepancies. In my review I found instances in which formulas appeared to be incorrect, and instances in which equations appeared to be missing elements. I also found instances of design assumptions that in my opinion are faulty and should not have been relied on in the design of the plant, being used as references in Sargent & Lundy evaluations of discrepancies found in the Reinspection Program. In sum, I found a number of problems in the design criteria for Byron.

Q11: Can you elaborate on the problems you found in the design criteria?

A11: Attachment 1 to my testimony is a non-exhaustive summary of what appear to be erroneous formulas, questionable design assumptions, and apparently faulty equations I found in my

brief review of the design criteria for Byron. Following that summary in Attachment 1 are copies of those sections of the design criteria that are listed in the summary.

Point 1 in the summary is section 12.2.4. In this formula the lambda symbol appears to be missing. This is important because these formulas should be correct. Although some engineers might know the correct formula without having to see it, others may not. If the wrong formula is utilized, faulty design may result. In 12.2.4 the formula is $P_{AE} = 1/2(\ )H^2 \overset{v}{o}\ k_{AE}$. The section of the design criteria where this apparent error is contained is otherwise so thorough that someone relying on it might not check the formula in another book. If the criteria had omitted the formula and simply referenced it, that would have caused to engineer to check the formula thereby finding the correct one. That would have been the better alternative. By making that section so thorough, then leaving out the lambda symbol, the likely result is that the formula may be used correctly.

This omission bears on the inferences about the plant which may be drawn from the Reinspection Program. This section of the design criteria relates to below-grade structural building outside walls - which were not included in the Reinspection Program and could not have been, because they are generally inaccessible and under the ground. Thus any error due to this omission would not have been detected in the program.

Q12: What are the other problems you find in the design criteria?

A12: The second point in my summary refers to Section 19.5.d. The equation appears to be missing a summation symbol before the b^2 (squared). Without the summation symbol it is not obvious that one should sum up the totals before b^2. It would make a significant difference in the calculation if such summation were not done. This equation relates to the summation of torsional stresses for the concrete turbine foundation - another example where any resulting error would not have been detected in the Reinspection Program, which did not tear up and reinspect concrete.

Also in Section 19.5.d the equation, I believe, should be the square root of F prime C. This is important because of the significant difference between the square root of the number and the number itself. Particularly, Section 19.5.d talks about allowable stresses. If the calculation is made without the square root, the result would be a higher allowable stress.

Section 32.3.2 also has an apparent error. It says :25 fy when it should be .25 fy. This section relates to buried piping. Again, any resulting error would not have been detected in the Reinspection Program, which treated buried piping as inaccessible.

Q13: Isn't that the kind of error an engineer would recognize right away?

A13: One would think so, but then why did no one correct the design criteria?

Q14: Are there other apparent errors you have found in the design criteria?

A14: Yes. Number 4 in my summary is Section 32.4.2., which also relates to buried piping. Spangler's equation appears to be listed as D.061, whereas it should be 0.061. Also, I believe that R^4 should be R^3 in the denominator. I do not know whether anyone took the D to mean diameter and used the diameter in the calculation, but this appears to be a problem. The difference between using R^4 as a denominator as opposed to R^3 is substantial. Although my review has been limited by time the apparent errors just discussed in addition to others listed on the summary in Attachment 1 suggest the presence of defects in some of the formulas Sargent & Lundy has employed at Byron.

Q15: When you stated that there are design assumptions that in your opinion appear to be faulty, to what assumptions did you refer?

A15: Section 34.2 of the design criteria states that embed plates are designed for 10 kips per foot tension load and 12 kips per foot shear load. I have attached Section 34.2 as Attachment 2.

Q16: What do you think the problem is with this section?

A16: This a major concern that I have from my discussions with Sargent & Lundy people and from what I saw in the field

while on the site visit. It appears that Sargent & Lundy has procedures to hang conduit, HVAC pipe supports, both small and large, off embed plates. If they hung everything off embed plates, as I understand Sargent & Lundy to say, there could be serious safety problems. I believe there are legitimate doubts as to whether the embed plates would survive a seismic event. These would affect, for example, Hatfield conduit supports, and Hunter pipe supports, which may be hung from embed plates, but again, the embed plates themselves were not reinspected in the Reinspection Program, and inadequacies in them thus may have gone undetected.

Q17: Please explain why embed plates might not survive a seismic event.

A17: Ten kips is, I believe, too small for the design of the plates. For example, on the field trip I saw a 12 inch line that had a strut to the embed plate on the wall. If there is a large load on the strut, as I observed in the field (the strut appears to have 15 or 20 kips based on the size of it), and that load is applied to the plate, the bolt strength can be exceeded and the whole plate can pull off the wall. If that occurred throughout the plant problems could be widespread. If a large number of embed plates are questionable, the plant could not undergo a safe shutdown earthquake.

Q18: How do you know that every embed plate is designed to 10 and 12 kips?

A18: I do not; however, that is what this document appears to say. I did not find any calculations for embed plates, but this document appears to state that all embed plates are designed in this manner.

Q19: Do you have other concerns with the design criteria?

A19: Yes. In Section 37.2.1, relating to mechanical component supports such as Hunter pipe supports, there is a listing of design effects that are to be ignored when performing calculations, for example, torsional stresses, axial self weight and assumptions that all masses are lumped at the shear center. The stresses are small, but they are non-conservative. If these stresses were added to the calculations some of them, I believe, would fail. This same problem occurred at Diablo, namely, ignoring minor but non-conservative design effects. At Diablo the NRC required that all the minor stresses be included and all calculations where minor stresses were ignored be recalculated. I have included a summary of these procedures and the procedures themselves as Attachment 5 to my testimony.

Q20: Do you have other concerns with the design criteria for Byron?

A20: Yes. I listed the major concerns above, but I have many other concerns that I have not had an opportunity to review in depth. Attachment 6 to my testimony lists these potential problem areas I have encountered with the Byron design criteria and my initial concerns. It may be that there are answers to my concerns with the procedures listed

in Attachment 6, but my questions have not been resolved by my review of the documents thus far.

Q21: What other documents have you reviewed that you have concerns with?

A21: One of the documents is entitled Seismic Subsystem and Equipment Response Spectra Design Criteria Byron and Braidwood Nuclear Power Stations Units 1 and 2 DC-ST 04 BB REV.2 Copy 48. This document is attached to my testimony as Attachment 7. It relates to buildings and would affect calculations for each component in the building, including Hatfield, Hunter and PTL inspected components; Reinspection Program calculations relied on this document. In Section V.B. the document states, "The horizontal seismic model of the nuclear power plant complex involves many degrees of dynamic freedom; theoretically a response spectra could be generated for each degree of freedom in addition to the horizontal model, a separate vertical model was developed for the vertical direction of excitation, so additional degrees of freedom for which response spectra could be generated were introduced into the analysis." (Emphasis mine) It appears that only one model was done for both the horizontal EW and NS, even though the building cross sections were different and only one vertical model was made. For each building about its respective center of gravity six (6) seismic loadings should have been applied: accelerations along three axises and moments about those three axises (Fx, Fy, Fz, Mx, My and Mz). In the spectra, only

accelerations for EM, NS, and vertical are shown. At least one other spectra per building should have been made, one for the torsional acceleration to building components at their radial distances from the center of gravity. For many components, this torsional component is significant on loads, stresses, and ultimately to a conclusion as to safety of the plant.

Q22: Why is this important at Byron?

A22: At Byron it appears that the torsional component is being ignored. In every plant I've worked in, the torsional component has never been ignored. I might add that neglect of the torsional component is consistent with the FSAR, Section 3.7.2.11, where torsional effect was listed as insignificant. Nowhere else where I have worked has the torsional effect been considered insignificant. Now this was apparently detected by the NRC in 1982. The NRC apparently did not altogether approve of the practice, but did find it to be in compliance with the FSAR.

Q23: In your opinion, is it reasonable to assume that at Byron the torsional effect is insignificant?

A23: If all of the items in the building are within 10 feet of the center of gravity, then it probably is insignificant. But many of the buildings have items as far away as 50-60 feet from the center of gravity of that building. For example, the turbine building is a long rectangular building, in which the torsional effects are likely to be substantial.

Q24: Are there other areas you are concerned with?

A24: Yes. In an NRC letter dated January 30, 1984 to Commonwealth Edison, on page 11 there was a reference to an allegation received by the NRC regarding undersized welds where tube steel was used. That document is attached to my testimony as Attachment 8. I do not believe that allegation was substantiated at the time. Also attached to my testimony as Attachment 9 are the Sargent & Lundy calculations for the weld survey project. Page 10 of that Sargent & Lundy document, titled Flare Bevel Groove Welds, states, "Typical field measurements indicate that the actual radius is between T and 2.5T, where T is the tube wall thickness. Therefore, the design assumption of R = 2T and effective throat equal to 5/16 R per AWS is not applicable." This quote in my opinion appears to substantiate the NRC allegation contained in Attachment 8. The document itself (Attachment 9) raises further questions. It was prepared by one D.J. Sheahan. The document contains no apparent indication that it was ever checked or approved. It has no page numbers, no calculation number, and no book number. The only other thing that helps one trace it is on the fifth page where there appears the name "D. Patel - 28." I looked up this name on a list provided to intervenors in discovery and found that he or she is one of the structural leaders at Sargent & Lundy for the reinspection program.

In my opinion what Attachment 9 suggests is that a design assumption of R = 2T is not valid. However, I am

not certain what design assumption was in fact used. Again, this same problem was present at the Diablo plant. We used the design assumption of R = 2T and found it was wrong.

Attachments 8 and 9 thus potentially bring into question every weld to tube steel in the plant. It may be the case that large numbers of welds of this type are as much as 50% deficient. This problem would potentially affect, for example, Hatfield conduit and cable tray and Hunter pipe supports.

Q25: Do you have any concerns relating to welds reinspected in the Reinspection Progam and repaired prior to engineering evaluation?

A25: In my review, I came across a document labeled Reinspection Program entitled <u>Daily Inspection Report Hunter Corporation Inspector Mark. M. Tabbert 8/16/83</u>. This is a list of 118 ASME, AISC deficient welds, mainly on a feedwater system. This document is attached to my testimony as Attachment 10. I have compared the list of 118 welds with the listing in the ASME category that was reviewed by Sargent & Lundy and it appears from the documentation available to me that none of the 118 were in the group reviewed by Sargent & Lundy. It thus appears that these 118 welds may have been fixed before any review was done. The list of the 118 welds describes the deficiencies, and in my opinion, if there had been an engineering evaluation performed on these weld deficiencies some would likely have been found to be

safety-significant.

Q26: What is the next area of your concern?

A26: Calculation book 19.1.2, the Design Procedures and Assumptions for Evaluation of As Built Welds with AWS Inspection Discrepancies, p. 14 No. 7 is attached to my testimony as Attachment 11. The document states that "Convexity is only considered a defect on welds with fatigue load application and does not effect welds at Byron/Braidwood stations." This quote is consistent with the testimony of John McLaughlin, who testified that pipe supports were not subject to fatigue analysis.

In my opinion, pipe supports are subject to fatigue and thus convexity should have been considered a defect. Donald L. Leone, in his testimony at page A.20 (fourth line) (later adopted by Mr. Branch) stated that, "weld discrepancies involving ASME Class I piping were evaluated against the fatigue analysis for the piping system."

This may evidence a lack of communication between the structural group and the mechanical group. If mechanical designs account for fatigue in the piping system, then the structural group should take that into account when designing those respective pipe supports. Pipe supports are subject to load reversal many times. In the NRC document Review of U.S. Nuclear Regulatory Commission 1983 Annual Report (Attachment 12), on page 17, an article entitled Water Hammers speaks of "water hammer", a condition that causes fatigue loading and states, "The frequency of occur-

rence is low and damage has generally been limited to piping supports." As the NRC article points out, pipe supports are thus subject to fatigue loading. I disagree with Sargent & Lundy's apparent position that fatigue loading does not affect AWS welds at the Byron plant.

Q27: Do you have other concerns affecting the Reinspection Program?

A27: Yes. Continuing in Sargent & Lundy Calculation book 19.1.2; Design Procedures and Assumptions, page 20, No. 5. (attached as Attachment 13) it states, list D1.1-83 as the structural welding code. In my review of the FSAR, the 1983 code was not listed. This would relate to all as built welds with AWS code discrepancies, whether Hatfield or Hunter.

Q28: Why is this a problem?

A28: There are potential problems when one installs or builds under one code and then attempts to pass the weld under a later code. For example, under the 1983 code it may be easier to pass a weld by calculation because the installation criteria in the new code are more demanding. In contrast, welds may be more difficult to justify by calculation under the earlier code because the installation criteria were then less demanding. Therefore, if one builds under an earlier code (with less demanding installation criteria) and attempts to pass the weld under a later code (requiring less demanding calculations) one may end up

passing a weld that may have failed under either code.

As a standard practice if one installs a weld under one code, one should qualify that weld under the same code.

Since the 1983 code is not referenced in the Byron FSAR, it appears that the weld was not installed per the 1983 code and therefore should not be qualified under that code.

Q29: Do you have additional concerns about the Sargent & Lundy calculations relating to the Byron Reinspection Program?

A29: Yes. I have questions on calculations and assumptions found in Calculation book 19.1.2, Sections 2.1 (p. 5) and 4.1 (pp. 7-11), Section 19 (pp. 1-5), Section 21 (pp. 77, 78, 78A), Section 21 (p. 97A), Section 21 (p. 109) and Section 21 (p. 113).

Q30: What is your concern with Section 2.1 and Section 4.1?

A30: This relates to a PTL inspected weld in the Reinspection Program. On weld No. 140, Beam No. 33601-L on page 8, the weld appears to be shown to be overstressed 1.18 to 1.1. This is true even though Sargent & Lundy used a 10% overstress factor for as built conditions. On page 11 the engineer writes, "Ry is assumed to be taken by check plates." On the same page actual stress divided by allowable is shown to equal .996, less than 1.0. It appears that this joint would fail if it were recalculated including Ry. (Reaction R in the vertical (y) direction.)

Q31: What is your concern with Section 21 pages 77, 78 and 78A?

A31: This, too, relates to a PTL reinspected weld. These pages concern a combination bolted and welded connection. The engineer who performed this calculation makes an assumption that the bolts take part of the load and that there is a minimum pretension in the bolts. My concern is that this assumption is, to my knowledge, not consistent with industry practice. When one designs a welded and bolted structural plate the weld is designed for 100% of the shear load. In the case of pull out or tension load the anchor bolts are designed for those loads. One cannot reasonably assume that a certain amount of load goes to the bolted connection. In this case, looking at the drawing of the weld and bolts, the bolts would not catch very much load until the weld fails. Even if there is bolt pretension initally, the bolts will relax over time. For the engineer to make the assumption that the bolts take the load, he or she would have to show that those bolts could take the load for the life of the plant. This assumption is not conservative, nor is it proven in the calculation. However, it appears that without this assumption, the weld would not have passed the code.

Q32: What is your concern with Section 21, page 97A?

A32: The calculation in this section, relating to a PTL reinspectd weld, shows "I = 2.13 greater than 1.0 n.q." If this means, as it would appear, that the interaction value is equal to 2.13, which is greater than the code allowable

of 1.00, then this equipment does not qualify under the code. This is among the worst cases of apparent failure I have come across in my review. The document goes on to say, "Therefore, an Rz must be added per Phase III modifications." It is not apparent what is a Phase III modification. If this did fail under the Reinspection Program, why was it not recorded as a Reinspection Program failure? A failure of this apparent dimension would, in my opinion, likely be safety-significant.

Q33: What is your concern with Section 21, page 109?

A33: This, again, concerns a PTL reinspected weld. The responsible engineer states, "This portion of the load will be taken from load "D" and it will be distributed to weld "A" and "B". In my opinion this should not merely be assumed to happen; it should be proved. There are reliable ways to distribute loads based on fixity or on the strength of the section. The fact that the engineer makes this assumption does not necessarily mean that the load would in fact react in that manner. I do not see any apparent foundation for this assumption. A calculation should have been performed to justify this questionable assumption. The only apparent way this assumption could be valid is if one assumes "D" failed. That would ensure "A" and "B" would assume the load, but that in itself fails "D".

Q34: Do you have concerns relating to Systems Control Corporation ("SCC")?

A34: Yes. I have a series of questions regarding SCC supplied control boards. Commonwealth Edison's NCR 695 Attachment A (my Attachment 14) shows that three main control board sections (1PM02J, 1PM02J, AND 1PM05J) has been repaired with "Bondo," an auto body type repair compound, and by tack welds, rather than by the full penetration weld specified in the design. An April 28, 1982 SCC letter to Sargent & Lundy (Attachment 15) revealed that SCC had used such auto body filler in many panel face repair applications. In the third paragraph, SCC states, "We can only conclude that the area of the board containing the cracks may have been subjected to abnormal thermal or structural stresses."

An April 30, 1982 Sargent & Lundy interoffice memorandum from J.A. Schwin to B.G. Freece in response (NCR number F-695; my Attachment 16) has a note at the bottom which states, "The use of body filler material (Bondo, etc.) is a standard practice of control board manufacturers in repairing blemishes to their boards."

The documents I have reviewed do not disclose the final resolution of this problem, but the documents do raise serious questions. The drawing called for full penetration welds, not tack welds and bondo. The question then arises -- What is the function of bondo on these control boards? Is it for strength? Or is it a sealant? These main control boards are Class I safety equipment. Is it possible that the bondo could again crack and a particle become

lodged in the contact switches? I would also question whether procedures are in place for the design, installation and qualification of bondo in Class I safety related controls.

In sum, I would state that if the use of bondo is as a sealant, high temperature silicon would be better. If the use is for strength, welding would be better, even taking into account the potential for warping the panel.

Q35. Do you have other concerns with SCC control board panels?

A35. Yes. There is another issue rela' d to SCC control boards which arises from my review of Edison's NCR F-544 (Attachment 17). NCR F-544 indicates that main control board panels OPM01J, OPM02J, 1PM01J, 2PM01J, 1PM04J, 1PM11J, and 2PM11J did not meet the AWS D1.1 code criteria for welds. As I understand AWS D1.1, it is not a stringent requirement, but is among the easier AWS code requirements to meet. In an effort to correct the situation NCR F-544 indicates that SCC was allowed to write its own acceptance criteria. This is an issue which would appear to deserve investigation.

Q36: Were there other Sargent & Lundy calculations for the Reinspection Program that caused concern?

A36: A review of Sargent & Lundy Calculation book no. <u>BRP-1 for Hunter Subjective Welding</u> also raised concern on my part. This calculation book reviewed 60 AWS type discrepancies and 49 ASME weld discrepancies, of which only two were on

the feedwater system and five on main steam. (This contrasts with the 118 ASME welds referred to earlier in my testimony which may have been fixed without a Sargent & Lundy review.)

Specifically, ASME welds Nos. 62 (S-CC-100-11A) and 63 (S-CC-100-33) (Attachment 17) were accepted despite the fact that the accuracy of the gauges supplied for measuring the welds was only 1/64 of an inch, whereas ASME requires machine shop type accuracy to the thousandths. The information I have reviewed suggests that this is an impermissible practice.

Q37: Do you have additional concerns over documents you reviewed at Sargent & Lundy?

A37: Yes. I have not had time to discuss them all in this testimony, but I do have time for one more. I reviewed a drawing in Sargent & Lundy's home office, Review of Category I Conduit Supports Typical Support Types and Load Tables DWG. 6E-3393B. This document depicted support type CF and MCF (floor to ceiling conduit support) and type CC and CP maximum load table. When I reviewed the tables I was immediately concerned. In the plant visit I observed cantilevers coming off the ceiling for 18 feet using unistrut. I found this alarming. The CF and MCF uses approximately 2 inch to 4 inch tubing. I saw conduit supports in the field that did not meet design requirements. It was floor to ceiling and was welded at the top and at the bottom. According to the drawing it should have had a slip or pin

connection at the top and no weld. Beyond that, it appears that the KL/R, which is the slenderness ratio, which is a factor that comes out of the UBC, AISC and a number of other documents including the unistrut catalog, which the actual supports came out of, states that the limit for KL/R is 200 feet. I reviewed many designs that exceeded the 200 foot factor. One I noted was 300 feet.

It appears there is a problem with the K factor used in the equation. The unistrut catalog uses .8 as a built in factor within that document. A pin connector uses a factor of 1.2, and a cantilever uses a factor of 2. When one uses any of those other factors instead of the .8 for unistrut, these supports are substantially outside of the length requirement. The unistrut book says that when one exceeds 200 for KL/R one no longer has a yield stress failure, but a buckling failure. I have viewed this problem both in the tables at Sargent & Lundy and in the field at Byron.

Attachment 1

RESUME

CHARLES C. STOKES P. E. SS# 416-72-9963
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COTTONWOOD,AL. 36320
TEL. (205) 677-5078

PERSONAL DATA:
Date of Birth - 03/12/51 - SINGLE - U.S. CITIZEN - EXCELLENT HEALTH

PROFESSIONAL EXPERIENCE:

FIELD ENGINEER - (NOV. 8, 1982 TO OCT. 14, 1983)
Accepted assignment to Pacific Gas and Electric Companies DIABLO CANYON NUCLEAR PROJECT UNITS 1 & 2. Placed in on-site engineering group. Performed pipe stress and pipe support design calculations. Wrote paper on how to design and represent flare, flare-bevel, skewed welds and other partial and full penetration welds on drawings to comply with AISC and AWS prequalified welds for structural and tube steel. Was assigned to Pipe Support Design Tolerance Clarification Group to authorize changes required for installation of supports and was responsible for snubber substitution on both units.

PIPE STRESS/SUPPORT ENGINEER - (2/82 TO 5/82)
Field consultant on Mississippi Power & Light's GRAND GULF 1 for RCI Inc. Assigned to Control Rod Drive System to assist ECHO pipe stress group and RCI hanger group in resolving interference problems by suggesting alternate design. Responsible for ECN's of as-builts and alternate designs and supervising drafting. Assisted QC and Construction personal in interpretation of drawings. BWR Plant and class 1 pipe.

MECHANICAL ENGINEER - (6/81 TO 2/82)
Assigned to the Mechanical Engineering Dept. of the Lawerence Livermore National Laboratory as a stress analyst on the injector of the Advanced Test Accelerator (ATA). Performed calculations on the injector housing, epoxy insulators, accelerator cells, cathode, anode, support structure and handling fixtures for fabrication and installation. System involved vacuum-oil interfaces and extremely strong magnetic and radiation fields. Injector constructed of aluminum and stainless steel with insulators of a special fill-epoxy compound. Also made design changes to epoxy insulators on Experimental Test Accelerator (ETA).

PIPE STRESS/SUPPORT ENGINEER - (10/80 TO 5/81)
Contracted to Nuclear Services Corporation, a division of Quadrex Corp. in San Jose California. Performed pipe stress calculations and design of safety related small bore piping supports. SAGS program was used in analysis of complex supports. Was assigned to ZIMMER NUCLEAR PLANT as a member of special pipe stress and hanger analysis group. Class I, II, III pipe.

PIPE SUPPORT ENGINEER - (6/80 TO 10/80)
Assigned to Bechtel Power Corporation's Civil Structural group in Gaithersburg, Md. working on the DAVIS-BESSE PROJECT. Checked and made base plate and anchor bolt stress calculations and modifications for anchors and pipe hangers. ANSYS finite element program utilized to account for plate flexibility and bolt elongation. Strudl was used for analysis of complex frames. Other in house programs were also used.

PROJECT/DESIGN ENGINEER - (7/75 TO 5/80)
Southern Company Services Inc., Birmingham, Alabama.
Wrote two specifications concerning modifications to Georgia Power's HATCH NUCLEAR PLANT.

continued:

The main item modified was the Reactor Heat Discharge System in the Torus.

Designed the structural steel truss for Georgia Power's SCHEREER PLANT coal conveyor system Unit No. 2, including details and bents.

Redesigned the precipitator structural steel on Alabama Power's MILLER STEAM PLANT to add precipitator roof enclosure. Elastic analysis performed to allow for thermal growth and to resist wind forces. STRUDL analysis, code check and design was used.

Acted as a nuclear pipe support stress analysis, designer and checker on Alabama Power's FARLEY NUCLEAR PLANT. Performed stiffness calculations and checks by hand and computer. STRUDL was used for analysis of complex structures. Also worked in the field supplying support information to office personal. Work performed in accordance with NRC 79-02 and 79-14. PWR class I, II, III pipe.

Served as civil material coordinator on Georgia Power's VOGTLE NUCLEAR PLANT. Was responsible for civil quantity take-offs for project construction scheduling, financing and material purchases. Computer storage and retrieval of information was used.

Did ANSYS finite element analysis of powerhouse substructure on Alabama Power's HARRIS DAM. Supervised drafting, checked drawings and checked calculations on superstructure concrete.

Designed outdoor structures on Alabama Power's MILLER STEAM PLANT. These included railroad, truck and ash pipe bridges, ash trench system and off-site make-up water system. Responsible for checking calculations, supervising drafting and coordinating field and inter-office disciplines.

PROFESSIONAL LICENSES AND AFFILIATIONS:
Registered Professional Engineer - State of Alabama - (12786)
Registered Professional Engineer - State of Florida - (29985)
Registered Professional Engineer - State of Georgia - (12340)

EDUCATION:
Birmingham School of Law, Birmingham, Al., Juris Doctorate degree, May 1980.
Auburn University, Auburn, Al., BCE degree, May 1975.
Massey Institute of Technology, Jacksonville Fl., correspondence accounting.

THE FACTS STATED ABOVE ARE TRUE AND ACCURATE

Charles C. Stokes P.E.

Attachment 3

RELATED CORRESPONDENCE

S & L DOCUMENTS REVIEWED

STRUCTURAL PROJECT DESIGN CRITERIA BYRON AND BRAIDWOOD NUCLEAR POWER STATION UNITS 1 & 2 (DC-ST-03-BY/BR) REV. 12

'84 AGO 16 P12:37

Sect.1.1, para. 2 No exceptions to the Final Safety Analysis Report and Enviromental Report is permitted.

Sect.7.4.1.b Interior walls 12", conc. slabs 12", on metal deck-category I floors 8", roof 14", control room ceiling 4" and category II slabs 6" (? anchor bolt problems)

Sect.7.4.2.b exterior walls below grade 15" min. thick. and above grade 24" min. thick.(? reversed)

Sect.8.1.a ACI 318-71 (? FSAR REQUIREMENTS) (? USED IN DESIGN)

Sect.8.1.b ACI 322-72 (? FSAR REQUIREMENTS) (? USED IN DESIGN)

Sect.8.1.c AISC-69(ELASTIC DESIGN) (? PLASTIC DESIGN) (? USED IN DESIGN)

Sect.8.1.d UBC-73(SEISMIC ANALYSIS CATEGORY II STRUCTURES) (? FSAR REQUIREMENTS) (? USED IN DESIGN)

Sect.8.1.e AISI-68(DESIGN COLD FORMED STEEL STRUCTURAL MEMBERS) (? FSAR REQUIREMENTS) (? USED IN DESIGN)

Sect.8.1.f 73 ASME Sect. III Div. 2, proposed standard code for concrete reactor vessels and containments. (? FSAR REQUIREMENTS) (? USED IN DESIGN)

Sect.9.5 CATEGORY II DEFLECTION WAIVED (? WITHOUT SOME LIMIT IMPOSIBLE TO DETERMINE WHEN CAT. II EFFECTS CAT. I)

TABLE 9.4-1 NOTE 5 1.67 AISC <= .95 Fy (? FSAR REQUIREMENTS)

TABLE 9.4-1 DESIGN STRESSES 1.75 AISC ? Fy (? FSAR REQUIREMENTS)

Sect.10.2.1.1.3.4 In all cases, structural members will be checked for the loads obtained from the pipe and cable pan hanger drawings. (? table diff.)

Sect.10.2.2.1.1 33 hz or less or increase acceleration 50 %

Sect.10.2.2.2.1 LEEWARD PRESSURE IS SUCTION NOT APPARENT IN TABLE

Sect.10.2.2.2.2 LEEWARD PRESSURE IS SUCTION NOT APPARENT IN TABLE

Sect.10.2.2.2.3 LEEWARD PRESSURE IS SUCTION NOT APPARENT IN TABLE

Sect.1Ø.2.3.3.1 FOLLOWING PARA. EXTREME ENVIROMENTAL (1.67 AISC allow. ? .95Fy) (? FSAR REQUIREMENTS)

Sect.1Ø.2.3.4 a. 1.6 AISC allow. .95Fy (? FSAR REQUIREMENTS) (? USED IN DESIGN) see other sections.

Table 1Ø.3-1 DESIGN STRESSES COLUME 1.6 AISC. allow. .95 Fy (<= left out)

Sect.12.2.4 FORMULA Pae=1/2 ? HXH Kae (missing lambda symbol)

Sect.18.1.1 ALL DESIGN ASSUMPTIONS, METHODS, REFERENCES AND MATERIALS SHALL BE DEFINED FOR EACH AREA OF DESIGN USING STANDARD CALCULATIONAL SUMMARY SHEETS.

Sect.19.5.d EQUATION MISSING SUMMATION SYMBOL BEFORE THE b SQUARED

Sect.19.5.d EQUATION SHOULD BE SQUARE ROOT OF F'c

Sect.2Ø.3.1.d MAX. WT. OF CONDUIT & CABLE DIFFERS FROM NEC 71 VALUES IN UNISTRUT CAT.

FIGURE 21.8-3 ?NF TO WELD

FIGURE 21.8-4 ?NF TO WELD

FIGURE 21.8-5 ?NF TO WELD

Sect.32.3.1 ? EQUATION NOT ABLE TO VERIFY (REF. STEEL PLATE ENG. DATA-VOL.3 WELDED STEEL PIPE AISI)

Sect.32.3.2 WALL THICKNESS SHOULD BE CHECKED FOR INTERNAL PRESSURE AND EXTERNAL LOAD BEFORE INTERNAL PRESSURE IS APPLIED, NOR HAS A MINIMUM THICKNESS BEEN CHECKED FOR SAFE HANDLING

Sect.32.3.2 :25 fy SHOULD BE .25 fy

Sect.32.4.2 SPANGLER'S EQUATION D.Ø61 SHOULD BE Ø.Ø61 AND R TO THE FORTH SHOULD BE R TO THE THIRD IN DENOMINATOR

Sect.34.2 EMBED PLATES DESIGNED FOR 1Ø KIPS PER FOOT TENSION LOAD AND 12 KIPS PER FOOT SHEAR LOAD (? PLATE SAFETY FACTOR WITH CRITERIA THAT ALMOST EVERY THING IS HUNG FROM THEM)

Sect.35.3.1 STRESS LIMITED TO 1.ØFy FOR LOADING AND Fy/sq. root of 3 FOR SHEAR (? .95Fy for tension loading)

Sect.36. *************

Sect.37.1.2 (? NO LIMIT OF DEFLECTION ON NON-SAFETY HANGERS IN SAFETY RELATED AREAS) WHAT CLEARANCE CRITERIA WILL BE USED TO ENSURE THAT NON-SAFETY DOESN'T DAMAGE SAFETY?

Attachment 5

Sect.37.2 NO DEFINITIVE STATEMENT THAT TORSIONAL STRESSES SHOULD BE CHECKED

Sect.37.2.1.f DEFLECTION AND ROTATION OF PRIMARY STRUCTURAL STEEL IGNORED IN DEFLECTION CHECK (? MEMBERS WITH PINNED ENDS)

Sect.37.2.1.g.1.B. IGNORE AXIAL SELF WEIGHT (? MAGNITUDE OF LOAD AFFECTING MEMBERS AND CONNECTIONS)

Sect.37.2.1.g.1.C. TORSION ANALYSIS NOT REQUIRED (? MAGNITUDE OF LOAD AFFECTING MEMBERS AND CONNECTIONS)

Sect.37.2.1.g.2.B. AXIAL SELF WEIGHT MAY BE IGNORED (? MAGNITUDE OF LOAD AFFECTING MEMBERS AND CONNECTIONS)

Sect.37.2.1.g.2.C. TORSION INCLUDED HERE ? LOGIC

Sect.37.2.1.g.3.A. ASSUME ALL MASSES LUMPED AT THE SHEAR CENTER

Sect.37.2.1.g.3.B. AXIAL SELF WEIGHT MAY BE IGNORED

Sect.37.2.1.g.3.C. TORSIONAL ANALYSIS IS NOT REQUIRED

Sect.37.2.1.g.4.A. ASSUME ALL MASSES LUMPED AT SHEAR CENTER

Sect.37.2.1.g.4.B. AXIAL SELF WEIGHT MAY BE IGNORED

Sect.37.2.1.g.4.C. TORSIONAL ANALYSIS NOT REQUIRED

Sect.37.2.1.g.5. EXACT ANALYSIS MUST BE PERFORMED FOR LOADS GREATER THAN 20 KIPS

Sect.37.2.1.g.5.A. ASSUME ALL MASSES LUMPED AT SHEAR CENTER

Sect.37.2.1.g.5.B. AXIAL SELF WEIGHT MAY BE IGNORED

Sect.37.2.1.g.5.C. TORSIONAL ANALYSIS NOT REQUIRED

Sect.37.2.1.g.6.A. ASSUME ALL MASES LUMPED AT SHEAR CENTER

Sect.37.2.1.g.6.B. AXIAL SELF WEIGHT MAY BE IGNORED

Sect.37.2.1.g.6.C. TORSIONAL ANALYSIS NOT REQUIRED

Sect.37.2.1.g.7.A. ASSUME ALL MASSES LUMPED AT SHEAR CENTER

Attachment 9

FLARE BEVEL GROOVE WELDS

FOR FLARE BEVEL GROOVE WELDS, THE EFFECTIVE THROAT MUST BE SPECIFIED FOR TUBE WELDS. THE MINIMUM RADIUS OF BEND IS NOT SPECIFIED IN ASTM A500 OR ASTM A501. THE MAXIMUM RADIUS IS SPECIFIED AS THREE TIMES THE TUBE WALL THICKNESS. TYPICAL FIELD MEASUREMENTS INDICATE THAT THE ACTUAL RADIUS IS BETWEEN T AND 2.5T, WHERE T IS THE TUBE WALL THICKNESS. THEREFORE, THE DESIGN ASSUMPTION OF R = 2T AND EFFECTIVE THROAT EQUAL TO 5/16R PER AWS D1.1 IS NOT APPLICABLE.

NO.	INPUT DOCUMENT: DESCRIPTION	SOURCE	IDENT.	REVISION NUMBER OR DATE ON LINE INDICATED: DATE	S	DATE	S	DATE	S	DATE	S	USED IN DESIGN OF	CNT	C/N
1.	BY/BR Design Criteria	S&L	DC-ST-03 BB	1982										
2.	Spectra Structural Design Standards	S&L	DC-ST-04 BB											
3.	S&L Structural Design Standards	S&L	SDS E1.0, 7.0, 30.0, 31.0, 33.0 5.0	1 1 0 1 1 0										
4.	Preparation, Review & Approval of Structural Dept. Design Calculations	S&L	SAS 22	2										
5.	Structural Welding Code	AWS	D1.1-83	1983										
6.	Calculation Books	S&L	12.2.8 to 12.2.62 18.1.1 to 18.1.50 18.2.1.2 18.2.1.3									Weld Assessment		

CNT - Continued C/N - Comment Number S - Status

SARGENT & LUNDY ENGINEERS

DESIGN CONTROL SUMMARY
DESIGN INPUT DOCUMENTS
PROJECT NO. 4391/92 UNIT NO. 1&2
PAGE 20 OF CALCULATIONS NO. 19.1.2 REV. 0 DATE

Attachment 13

RELATED CORRESPONDENCE

DOCKETED USNRC

'84 AUG 16 P12:36

OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

Attachment 14

RELATED CORRESPONDENCE

FORM QP 15-1.1
8-25-81 (Rev.6)

Commonwealth Edison

NONCONFORMANCE REPORT FOR CONSTRUCTION AND TEST

NCR NO. 695

SHEET 1 OF 2

IDENTIFICATION AND DESCRIPTION — PART 1

1. DESCRIPTION OF ITEM: MAIN CONTROL BOARDS 1PM02J, 1PM05J

(EQUIP. NO.): 1PM02J, 1PM05J

3. SYSTEM AND UNIT: VARIOUS, UNIT 1

MANUFACTURER/SUPPLIER: C.D. SYSTEMS CONTROL

DOCKETED USNRC

'84 AUG 16 P12:36

OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

8. CATEGORY: DEFECT / FAILURE / DAMAGE / DWG NONCONFORMANCE / SPEC NONCONFORMANCE / UNSAT CONDITION / DOCUMENTATION

SUPPLIER INSPECTION / CONSTRUCTION / CONTRACTOR / TEST

10. DESCRIPTION OF NONCONFORMANCE: Manufacturer's Filler Board front not welded with full penetration welds. Tack welded / Body filler used at located shown on attachment A. Appear to be cracking with age

P.O. NO. / HOLD TAG NO.: VARIOUS

WORK REQUEST NO.: N/A

14. SIGNATURES

		DEPARTMENT	DATE
NONCONF. OBSERVED BY	[signature]	PCD	2/23/82
NONCONF. VERIFIED BY	[signature]	PCD	2/23/82
	[signature]	QA	3/2/82
	[signature]	PCD	2/23/82

10a. WORK LIMITATIONS: X WORK CAN PROCEED ☐ WORK CANNOT PROCEED X OTHER LIMITATIONS

EXPLAIN WORK LIMITATION: When cracks (Tack weld + Body filler found) repairs to be made only to No. 16.

EVALUATION AND DISPOSITION — PART 2

11. CAUSE OF NONCONFORMANCE: UNKNOWN

10CFR50.55e Report Required Yes ☐ No ☐

10CFR21 Notification Yes ☐ No ☐

Name Date Time

NRC

Designated CECo Officer

ECN Required Yes ☐ No ☐

16. ACTION REQUIRED TO CORRECT THE NONCONFORMANCE: Re-weld with full penetration per Hatfield Electric's Procedure #13AA detail B-U3b under section II groove weld.

17. CORRECTIVE ACTION REQUIRED TO PREVENT RECURRENCE OF NONCONFORMANCE (WRITE N/A IF NOT KNOWN)

18. REVIEW AND APPROVAL

REVIEWED BY ______ DATE ______ APPROVED BY ______ ENGINEERING DEPARTMENT HEAD DATE

DISPOSITION — PART 3

19. DESCRIPTION OF ACTION TAKEN TO CORRECT THE NONCONFORMANCE

FORM ______ USED

20. DESCRIPTION OF CORRECTIVE ACTION INITIATED TO PREVENT RECURRENCE

21. DISPOSITION COMPLETED BY: ______ NAME DEPARTMENT DATE

22. DISPOSITION COMPLETION APPROVED BY

CLOSURE — PART 4

23. CORRECTIVE ACTION/QC REVIEW

☐ C.A. NOT REQUIRED ☐ C.A. ADEQUATE ☐ FURTHER ... REQUIRED

Attachment A to NCR No ____

Control Panel	DWG/Location	deficency
1PM02J	6E-1-4045B (1)(2) JF36 & CF36 (Cracks At Both Ends)	2 cracks total
1PM02J	6E-1-4045A (1)(2) VF15 & SF11 (Tack weld & Body Filler)	2 cracks total
1PM05J	6E-1-4051B Rev A (1) LI-462 & TI-453 (2) PI-458 & TI-450	2 cracks total

Attachment 17

Commonwealth Edison

RELATED CORRESPONDENCE

FORM QP 15-1.1
12-15-73 (Rev.2)

NONCONFORMANCE REPORT FOR CONSTRUCTION AND TEST

NCR NO F544

SHEET 1 OF 1

PART 1

3. DESCRIPTION OF MATERIAL, PARTS OR COMPONENTS: U-1 & U-2 MAIN Control Boards

OPM01J, 1PM01J, 1PM02J, 1PM05J, OPM02J, 2PM01J, 2PM05J, 2PM06J

5. SYSTEM AND UNIT: S&L SPEC 2788 MISC Systems

6. MANUFACTURER/SUPPLIER: Systems Control

8. CATEGORY: ☐ DEFECT ☐ FAILURE ☐ DAMAGE ☐ DIG NONCONFORMANCE ☐ SPEC NONCONFORMANCE ☒ UNSAT CONDITION ☐ DOCUMENTATION ☐

9. OBSERVED DURING: NRC INSPECTION ☐ SUPPLIER INSPECTION ☐ CONSTRUCTION ☐ CONTRACTOR

USNRC '84 AUG 16 P12:36 OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

12. DESCRIPTION OF NONCONFORMANCE: Welding of structural members of control boards do not meet (AWS D1.1) criteria

[Should say a lack of qualification of welders.]

11. P.O. NO. & P.O. ITEM NO.: 207534

12. HOLD TAG NO.: None Required

13. WORK REQUEST NO.: N/A

14. SIGNATURES

	NAME	DEPARTMENT	DATE
NONCONF. OBSERVED BY	[signature]	QA	8-8-80
NONCONF. VERIFIED BY	[signature]	PTL	8-14-80
QA SUPERVISOR COORDINATOR	[signature]	Q.A.	8-14-80
PROJ. OR SITE PROJECT ENGINEER	[signature]	SCD	8/14/80

PART 2

15. CAUSE OF NONCONFORMANCE

10CFR50.55e Report Required Yes ☐ No ☐

16. ACTION REQUIRED TO CORRECT THE NONCONFORMANCE

17. CORRECTIVE ACTION REQUIRED TO PREVENT RECURRENCE OF NONCONFORMANCE (WRITE N/A IF NOT KNOWN)

18. REVIEW AND APPROVAL

REVIEWED BY APPROVED BY

DATE

PART 3

19. DESCRIPTION OF ACTION TAKEN TO CORRECT THE NONCONFORMANCE

20. DESCRIPTION OF CORRECTIVE ACTION INITIATED TO PREVENT RECURRENCE

21. DISPOSITION COMPLETED BY 8008??0??7

22. DISPOSITION COMPLETION APPROVED BY

PART 4

☐ C.A. NOT REQUIRED ☐ C.A. ADEQUATE ☐ ADDITIONAL C.A. REQUIRED