UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:	)	
	)	Docket Nos. 50-454 OL
COMMONWEALTH EDISON COMPANY	)	50-455 OL
	)	
(Byron Nuclear Power Station,	)	
Units 1 and 2)	)	

AFFIDAVIT OF JAMES L. MILHOAN REGARDING MOTION TO REOPEN THE RECORD IN THE BRYON LICENSING PROCEEDING TO INCLUDE THE BYRON STATION DESIGN AS AN ISSUE

I, James L. Milhoan, having been duly sworn, state as follows:

1. My name is James L. Milhoan, and I am Chief, Licensing Section, Quality Assurance Branch, Division of Quality Assurance, Safeguards, and Inspection Programs, Office of Inspection and Enforcement, United States Nuclear Regulatory Commission. I am responsible for supervision of engineers whose jobs are to (1) review applicants' quality assurance programs for design, construction, and operation of nuclear power plants, (2) review and evaluate independent design review programs initiated by applicants and (3) inspect the quality of design activities, including examination of the as-built configuration. A statement of my professional qualifications is enclosed. (Attachment 1) In addition to the qualifications listed in Attachment 1, I also serve as a member of the ASME Boiler and Pressure Vessel Code Section III Subgroup on General Requirements and as a member of the Subcommittee on Design and Procurement of the ASME Committee on Nuclear Quality Assurance.

2. The purpose of this affidavit is to respond to the Intervenors' Motion to reopen the record in the Byron licensing proceeding to include the Byron station design as an issue.

3. The Intervenors' Motion (page 15) states that in their design review of a very limited portion of the plant, the IDR and IDI have revealed enough questionable design-related practices that a comprehensive design review is needed before there can be reasonable assurance that Byron can be operated safely. The Intervenors' Motion (page 1) also states that as a result of IDR findings, there is a likelihood that design deficiencies of safety significance exist throughout the Byron station.

4. In considering the overall results of the IDR[1/], the NRC integrated design inspection[2/] (IDI), and the Confirmatory Report on Jet Impingement Effects[3/], I believe it is reasonable to conclude that additional assurances of the design adequacy of Byron have been provided as follows: Detailed evaluations by either the staff or an independent contractor to the applicant have been performed on the auxiliary feedwater system, the component cooling water system, essential service water system, and the 125 VDC distribution system. These evaluations involved a substantial amount of engineering evaluation effort. The IDR alone, which was conducted by the independent contractor in response to the IDI findings, involved approximately 15,000

1/ IDR - Independent Design Review of Byron Station conducted by Bechtel Power Corporation, Final Report dated August 1984.

2/ IDI - NRC Integrated Design Inspection, Inspection Report (50-454/83-32) issued September 30, 1983.

3/ "Byron 1 Confirmation of Design Adequacy of Jet Impingement Effects", prepared by Sargent and Lundy for Commonwealth Edison, dated August 1984.

inspector-hours. My experience is that this is much more than is normally involved in an IDVP.[4/] The integrity of the design of the specific systems was established in that these evaluations did not identify any design errors which have to date required substantial design changes. The IDR had five observations in response to which the applicant made either a design change or imposed an additional administrative procedure. We have considered these five cases and determined that none of the resulting changes or the administrative procedure were essential to ensuring the plant could be operated safely or placed and maintained in a safe shutdown condition. Details of our review of these observations are forth in Paragraph 26, below. Nonetheless, as discussed in Paragraphs 20, 27 and 32 below, certain matters are under continuing review.

5. In August, 1982 the NRC staff undertook a number of initiatives to improve assurance of quality in design and construction of nuclear projects. One of these initiatives was to develop and implement an integrated design inspection (IDI) program to assess the quality of design activities, including examination of the as-built configuration. The objective was to expand the NRC examination of quality assurance into the design process. The approach is intended to provide a comprehensive examination of the design development for a selected system. A conclusion about the overall design process is reached based on the results of the inspection sample. The inspection is a multidisciplinary review involving mechanical systems

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[4/] IDVP - Independent Design Verification Program. On a case-by-case basis, the staff has requested that an applicant provide additional assurance that the design process used in constructing the plant has fully complied with NRC regulations. Many licensees have responded by initiating a design review through an independent third-party contractor. This has been termed an IDVP.

and components, electrical power, civil/structural, and instrumentation and control disciplines. The primary focus is on assessment of the implemented design process. The process is evaluated by examining actual design details. If errors are found in the design details, the design process is evaluated to see if the error resulted from an isolated mistake or if it reflects a more fundamental weakness in the design process.

6. The IDI for the Byron Station 1 was the second inspection conducted by the NRC in the IDI program. The inspection evaluated the design process based on examination of the auxiliary feedwater system. The IDI was conducted by the Office of Inspection and Enforcement on May 23-June 10, 1983, and June 20-June 30, 1983, at the Byron Station, Commonwealth Edison Company corporate office, Sargent and Lundy Engineers office, and Westinghouse Electric Corporation office. Results of the integrated design inspection are contained in an inspection report (50-454/83-32) issued September 30, 1983.

7. Although the inspection sampled a small part of the design effort, the inspection team reviewed hundreds of specific items. Approximately 7000 inspector-hours were expended by the staff and contractors to the staff in the conduct of the IDI and follow-up evaluations. The IDI team, while noting certain deficiencies, concluded that the design process in the areas of instrumentation and control, civil-structural, and electrical power was adequately controlled.

8. The IDI team was unable to conclude that the mechanical systems area was adequately controlled. Specifically, the IDI team identified deficiencies in the analyses related to postulated cracks and breaks in high-energy and moderate-energy lines and internal flooding. The IDI team recommended a comprehensive review, audit, and corrective action program to assure that the design work in this area was complete, adequate and controlled.

9. The IDI team also stated that its findings indicated a pattern of problems concerning the availability of valid, updated calculations to support the current design in the mechanical systems discipline. The IDI team was unable to conclude that valid, updated analyses were generally available. Accordingly, the IDI team recommended a systematic review and corrective action program to assure that the necessary calculations in the mechanical systems discipline were identified, performed, and updated as necessary to support the current design.

10. The concerns expressed by the IDI related mostly to the documented bases and calculations to support the current design rather than the design itself. Aside from the issue of meeting licensing commitments related to postulated breaks and cracks in high-energy and moderate-energy lines, when the IDI team examined the actual design in detail, no significant problems were found. This appeared to represent a general pattern that the IDI team attributed to the experience of Sargent & Lundy personnel.

11. In the mechanical components area, a concern was developed with respect to that portion of the balance-of-plant piping design work being performed by Westinghouse. The team examined a relatively small sample of this work

and found deficiencies. The sample was too small to permit conclusions; however, it did raise questions. Accordingly, the IDI team recommended further examination of the Westinghouse portion of the balance-of-plant piping design work to determine whether or not systematic weaknesses are indicated. In other respects, the IDI team was impressed with the work reviewed in the mechanical components area, which appeared to be generally correct, organized, and documented, indicating a controlled process.

12. Commonwealth Edison was requested to respond in writing to the inspection findings and unresolved items. It was also requested that the response include a description of Commonwealth Edison plans regarding programs recommended by the team for (1) high-energy and moderate-energy lines and internal flooding, (2) mechanical systems design calculations, and (3) Westinghouse balance-of-plant piping design. Finally, Commonwealth Edison was requested to include in the response its position, and the bases therefore, with respect to the necessity for conducting audits of design implementation in areas other than those audited by the NRC inspection so as to assure itself that deficiencies of similar importance either did not exist or were corrected.

13. On December 13, 1983, a public meeting was held in Chicago, Illinois, to review the major IDI findings. Subsequent to the December 13, 1983, public meeting, Commonwealth Edison on December 30, 1983, submitted its response to the September 30, 1983, IDI report.

14. In a March 23, 1984, letter, the NRC staff informed Commonwealth Edison of the results of its review of responses contained in the Commonwealth

Edison December 30, 1983, letter. While Commonwealth Edison disagreed with some of the particular IDI findings, it took corrective action to resolve the particular findings in any case. The staff did not reply to these Commonwealth Edison statements since corrective action was being taken. The NRC staff noted where additional information was required or where additional review of responses was being conducted. In addition, the staff informed Commonwealth Edison that reinspections related to the Byron IDI in the areas of internal flooding, postulated breaks and cracks in high-energy and moderate-energy lines, and electrical power would be conducted. The NRC also noted that the Commonwealth Edison position with respect to the necessity for conducting audits of design implementation in areas other than those covered by the IDI was still being reviewed.

15. As described below, as a result of various reinspections and NRC staff review of additional information submitted by Commonwealth Edison, IDI items with the exception of high energy line break analysis and auxiliary building flooding have been satisfactorily resolved. The IDI finding regarding high-energy line break analysis requires further staff action as discussed in Paragraph 20 below. The IDI finding regarding auxiliary building flooding appears adequately resolved subject to staff review of documentation as discussed in Paragraph 27 below.

16. With respect to the IDI comments concerning portions of the balance-of-plant piping design work being performed by Westinghouse, Commonwealth Edison initiated additional comprehensive reviews of the Westinghouse work and it was concluded that there were no systematic deficiencies in the

control of this work and no hardware modifications were necessary. The IDI team considered this action an acceptable resolution of its finding.

17. The IDI team was unable to conclude that valid, updated calculations were generally available to support the current mechanical systems design. To address this concern, Commonwealth Edison, in its letter of December 30, 1983, stated that all safety-related calculations in the Sargent and Lundy Project Management Division (PMD) calculation books were being reviewed to verify if they were technically adequate to support the current Byron/Braidwood design and to determine if the format conforms to the applicable version of the Sargent and Lundy procedure at the time the calculations were performed. In a June 19, 1984 letter to Mr. R. C. DeYoung, Director of the Office of Inspection and Enforcement, Commonwealth Edison provided the results of its review and concluded that the current mechanical systems design was adequately supported by calculations. Commonwealth Edison stated that a total of 112 calculations were reviewed and that 39 required no changes to the original calculations. Seventy three calculations were revised to incorporate updated information or the documentation was improved. However, Commonwealth Edison also stated that the existing calculations were technically adequate and supported the original design and that revisions to the calculation format, list of references, updated information, or other related areas were made in order to improve the documentation aspects of the calculations and in no instance did these changes result in a design change or hardware change. Commonwealth Edison also stated that its review did not include any specific provisions to determine that all necessary calculations by Byron PMD engineers had been identified and performed but that two considerations should resolve the particular issue:

a. The Mechanical Project Management Engineers initiated a survey to confirm that the necessary PMD calculations had been performed. Two additional calculations, related to sizing of the diesel oil day tank and the diesel oil storage tank, resulted from this review.

b. In order to provide additional assurance that Sargent and Lundy had adequately addressed the issue, the services of Bechtel Power Corporation were retained to perform an IDR, which would include a review of the design adequacy and the design process of the selected sample systems to ensure that the output documents meet the licensing commitments and safety-related design requirements.

The IDI team considered this an acceptable resolution of its finding subject to review of the Bechtel IDR, which is discussed separately below.

18. The IDI report identified deficiencies in the analysis related to postulated breaks and cracks in high and moderate energy lines and internal flooding. The IDI team performed a reinspection of this area between March 26 and April 10, 1984. In a May 2, 1984 inspection report the staff informed Commonwealth Edison that the results of the reinspection of Sargent and Lundy's analyses of postulated failures to high and moderate energy piping indicated the analyses were not complete enough to ensure that the design was adequate and that additional work was required. Open items from the reinspections were identified in May 2, 1984 and April 9, 1984 inspection reports. The most significant items involved components and equipment not addressed by Sargent and Lundy's jet impingement analysis.

19. On August 16, 1984 Commonwealth Edison submitted a confirmatory report ("Byron 1 Confirmation of Design Adequacy of Jet Impingement Effects") addressing the IDI concerns on jet impingement effects and a letter addressing the reinspection open items. Cognizant Office of Nuclear Reactor Regulation review branches were requested to review the August 16, 1984 Commonwealth Edison submittal. The Auxiliary Systems Branch (ASB) reported its review findings in a September 11, 1984 memorandum (Attachment 2). The Mechanical Engineering Branch (MEB) documented its review on a memorandum of October 11, 1984 (Attachment 3)

20. Reviews by the IDI team, ASB and MEB have concluded that Commonwealth Edison responses were acceptable with respect to analyses of the effects due to failures of high and moderate energy piping subject to the following. MEB has a follow-up item with respect to Sargent & Lundy's use of NUREG/CR-2913 dated January 1983, which was used as a basis for defining jet impingement zones of influence in the confirmatory report. The staff has made a preliminary review of this report and finds its general methodology and analytical approach acceptable. MEB recommends approval of operation up to 5% power pending the applicant's identification of the systems and locations where the methodology of the NUREG report was applied and its demonstration that it meets the commitment in FSAR Section 3.6.2, to comply with Standard Review Plan, Section 3.6.2, Paragraphs III.2c and III.3. MEB's recommendation is based on the fact that the probability of a pipe break is extremely low during the approximately 2 month anticipated period of low power testing and the consequences of a break are likely to be less severe should they occur during low power testing than at higher power levels in view of the low levels of decay heat and the

small fission-product inventory. Additionally, should any additional steps be required as a result of the further review to provide protection against postulated breaks those steps could be undertaken following low power testing.

21. The Bechtel IDR report confirmed the IDI concerns and stated that it could not be established that the design process for high energy line break analysis was carried out in sufficient depth to assure meeting the design objective. In response to staff questions at a September 14, 1984, public meeting, Bechtel agreed to document its review of the confirmatory report and provide a description or final statement of how its previous finding could be closed out. In a Commonwealth Edison letter of October 1, 1984 to Mr. R. C. DeYoung, the staff was informed that Bechtel had reviewed the confirmatory report and that the report satisfies Bechtel's concern for design process previously identified (Bechtel Observation Report 8.47).

22. Subsequent to the March 23, 1984, NRC letter (see paragraph 14), Commonwealth Edison took action to have Bechtel Power Corporation conduct a review of Sargent & Lundy covering three systems at the Byron station. In a June 5, 1984, letter to Commonwealth Edison, the NRC staff approved the program plan for a review by Bechtel of three systems for the Byron station. The letter stated that the Bechtel review was a satisfactory method for resolving the IDI finding concerning the necessity for conducting additional audits of design implementation, subject to factoring NRC staff comments into the program plan. In a July 6, 1984, letter, Commonwealth Edison provided acceptable responses to the NRC staff comments on the program plan.

23. The IDR is similar to independent design reviews conducted on behalf of other applicants in cases where the NRC does not perform an integrated design inspection. On a case-by-case basis, the staff has requested that an applicant for an operating license provide additional assurance that the design process used in constructing the plant has fully complied with NRC regulations and licensing commitments. Many licensees have responded by initiating a design review through an independent third-party contractor. This has been termed the Independent Design Verification Program (IDVP). An IDVP is intended to provide close examination of the design process and its implementation for a limited sample of structures, systems, or components, for a particular nuclear power facility. It is a multi-disciplined review that addresses mechanical, electrical, structural, and instrumentation and control disciplines. The review has generally encompassed the entire design process, including formulation of the principal design and architectural criteria, development and formulation of the design into construction of the facility, and onsite verification in the selected areas. (The IDR did not include verification of physical installation.) In addition, the IDVP third-party reviews have normally addressed programmatic areas. For example, classification of systems and components, design verification records, interface control and interdisciplinary review, and consistency with the Final Safety Analysis Report (FSAR) are sampled. The NRC staff reviews the selection of the independent review organization and the audit plan before it is implemented, reviews the completed report, and assesses the applicant's response to the audit findings. Plants that have received an NRC integrated design inspection, or that are replicates of plants that have already been subjected to an independent third-party design review, have generally been able to

provide sufficient assurance that the design process has complied with NRC requirements without performing a second design review.

24. At an August 14, 1984, public meeting in Bethesda, Maryland, Bechtel presented an overview of its IDR. Section 1.3 of Volume 1 of the Bechtel IDR Report provides a concise summary of the IDR results. Excerpts from this section are provided below:

   - A total of 49 Potential Observation Reports (PORs) were prepared. Of these, 14 were determined to be invalid and 35 were valid. Each of the valid Observations Reports (ORs) has been resolved to the satisfaction of the IDR team.

   - The resolutions of the ORs required minimal changes in design and other documents, including licensing documents. These resolutions covered not only each specific OR, but had broader implications as well.

   - None of the Observations is regarded as safety-significant by the IDR team, although some Observations did require further design activity or commitments to future action by S&L for IDR resolution. In one case, the adequacy of the design pressure of the CCW system was questioned and resulted in Westinghouse conservatively reporting the situation to the NRC and in a design change being initiated. Also, particular attention was given to the areas of design for high energy line breaks (HELB), moderate energy line breaks (MELB), and fire protection.

- There were some negative trends identified by the IDR team analysis of the apparent root causes of the Observations requiring design or documentation changes for acceptable resolution. The trends observed were categorized into the following four areas:

  - The use of undocumented judgments;
  - Insufficient control of the FSAR;
  - Insufficient review of changes;
  - Noncompliance with Code requirements.

- These trends indicate that certain aspects of the design activities appear to have been controlled less systematically and rigorously than currently appropriate; however, review of the specifics of the relevant Observations resulted in a judgment that these aspects are not sufficiently significant to justify further investigation.

- Generally, the IDR team found that the design reflected acceptable standards of technical adequacy and design process, and that the apparent intent of key licensing requirements was consistently met.

25. At a September 14, 1984, public meeting in Chicago, (with Mr. T. Wright, one of the attorneys for Intervenors in attendance) the NRC staff discussed its comments on the Bechtel IDR Report with Commonwealth Edison and its architect-engineer, Sargent & Lundy, and the IDR contractor, Bechtel Power Corporation. Comments were resolved to the NRC staff's satisfaction or commitments to provide additional information or to document corrective actions were obtained. Significant items from the September 14, 1984,

public meeting are noted in Attachment 4. On October 1, 1984, Commonwealth Edison responded to the majority of NRC comments (Attachment 5). In addition, Commonwealth Edison has provided additional information regarding IDI Finding 2-19 and a description of the methodology used to address pipe whip in the jet impingement study as discussed in the October 1, 1984 letter. FSAR changes to close IDR Observation 8.47 are still to be submitted. However, this is part of the staff's continuing review of high energy line break analysis (see Paragraph 20 above). The staff has reviewed the Commonwealth Edison submittals and staff comments are noted in Paragraphs 27 and 32 below.

26. The Intervenors state that they have now completely reviewed the four volumes of the Bechtel IDR and are greatly concerned with the number of potentially safety-significant design problems found in what they characterize as the very limited review conducted by Bechtel. The Intervenors conclude that the IDR, when read in the context of the IDI, shows enough serious defects at Byron that a complete and comprehensive independent design review is warranted. (Motion, pp. 1-2.) Although it is correct that a limited number of systems were reviewed, those reviews involved detailed examinations as described in Paragraph 4 above and it is therefore inaccurate to characterize the reviews as "very limited". Contrary to the Intervenors' contention that serious defects were identified, as also noted in Paragraph 4 above, the evaluations conducted to date (IDI and IDR) have not identified design errors which required substantial design changes. The IDR had five observations in response to which the applicant made design changes or imposed an additional administrative procedure. These five observations were technically reviewed by IDI team members with the following comments.

a. The design of the Class 1E dc bus (OR 8.9) was technically adequate prior to the addition of fuses which the applicant agreed to install to provide an additional measure of conservatism.

b. The plant condition in which the battery cross-tie breakers (OR 8.10) would be shut (e.g., one unit shutdown in an outage with its battery disconnected) is too remote a condition to consider the administrative procedure which was established to be a substantial design change.

c. Bringing certain component support weld sizes (OR 8.16) into conformance with ASME Code requirements added a further measure of conservatism to welds already analyzed to be technically adequate.

d. The condition which precipitates over-pressure in the CCW system (OR 8.38) involves the postulated passive failure of safety-grade tubing (RCP thermal jacket or letdown heat exchanger). While system piping and valves probably have sufficient margins to accommodate the over-pressure, there is no guarantee that pump seals, heat exchanger flanges or heat exchanger tubes will not fail. Nevertheless, in this case the CCW system design is sufficiently flexible to permit rapid isolation of the in-leakage, isolation of the affected CCW component, and orderly plant shutdown.

e. The addition of covers to power cable tray (OR 8.39) provides an additional measure of conservatism to an already technically adequate design.

27. With respect to resolution of IDI findings to date, no item has required a design change for its resolution with the possible exception of a piping change associated with a flood level calculation (Finding 2-19). In its October 1, 1984 letter to Mr. R. C. DeYoung, Commonwealth Edison stated that additional information on auxiliary building flooding would be provided to address IDI finding 2-19. On the afternoon of October 10, 1984 the NRC staff was telecopied the following information:

"The completed flooding calculation (Calculation 3C8-1281-001) was transmitted to the responsible design groups (Structural, Electrical, Control and Instrumentation, HVAC) for review. Each group reviewed the impact on their area of design. Structural incorporated the calculated flood levels into the Structural Final Load Check. Electrical walked down the areas containing safety related electrical components and identified those below the predicted flood level which could be adversely affected by flooding. In the areas of Control and Instrumentation and HVAC, affected components were identified by a review of the design documents.

"As anticipated, the Structural Final Load Check confirmed that flooding would affect certain block walls which had not been designed to withstand flood loads. The potential failure of these walls has been shown to not adversely affect the safe shutdown capability. Because of the combination of increased equipment loads and flood loads, the floor in one pump room was found to be potentially overstressed. Because of the difficulty in establishing the load at which the block wall or door would fail, additional outflow area from the room is being added. No other changes have been made.

"The Project Management Division has reviewed the safety related components potentially affected by flood and documented that safe shutdown capability is maintained for the postulated flooding events. No design changes were required as a result of this flood review."

The staff will request to see the documentation referenced above. The staff, based on the above information, is not certain a design change was necessary to resolve the IDI finding.

28. The Intervenors assert that Bechtel's conclusions in the IDR are suspect because Bechtel used an unduly high threshold for determining safety significance of observations (Motion Pg. 3). In the IDR, Bechtel defined the term "safety significant condition" as "a condition confirmed to exist which results in a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety," (IDR, Volume 1. p.8).

29. The NRC staff also had questions concerning the use of the term "safety significant condition" and at the September 14, 1984, public meeting posed questions to Bechtel to assure that it had an understanding of the term and its effect on the conduct of the IDR. Bechtel assured the staff that every observation noted by the IDR Team was included in the IDR report and the

NRC inspection team had access to all Bechtel files during its inspection regardless of the classification assigned to an IDR team member observation. Thus, the NRC had access to all observations in our review of the IDR report.

30. The Intervenors assert that the definition of "safety significant condition" also "raises questions about the cumulative impact of numerous conditions, no single one of which, by itself, is adjudged to be sufficiently 'major' to qualify as 'safety significant' within Bechtel's definition" (Motion, p. 4). The staff does not believe that the Bechtel findings (either collectively or individually) raise a significant question regarding the adequacy of the design of the Byron plant. Nonetheless, as discussed in paragraph 32 below, certain matters identified in the IDR will result in additional description by the applicant of its corrective action. Although the Intervenors view the IDR as having been a very limited study which uncovered a significant number of items, I do not find it surprising that deficiencies were noted due to the large amount of effort and the depth of review involved in the IDR. Based on my review of IDI reports for Byron and other plants, the number and extent of deficiencies is not inconsistent with past inspection report findings. Also, the IDR and the IDI report findings relate primarily to deficiencies in the design process rather than the design itself.

31. With respect to the impact of individual IDR observations, the staff requested at the September 14, 1984 public meeting to be provided a documented basis for Bechtel's statements that for individual observations there is no reason to expect the situation is cause for a significant

concern elsewhere in the plant. Such statements were frequently used by Bechtel in close-out of observation reports. In a Commonwealth Edison letter of October 1, 1984, Bechtel provided its response to the staff's request. The staff has reviewed the Bechtel response and believes it adequately responds to the staff request with respect to close out of individual observations. Trends identified by the individual observations are discussed in paragraph 32 below.

32. The Intervenors express concern over Bechtel's identification of negative trends in the design process (Motion, pp. 2-3). In particular, the Intervenors express considerable concern over Bechtel's finding that Sargent and Lundy's engineering judgments are often poorly documented. (Motion, pp. 8-9 and 15). At the September 14, 1984 public meeting Commonwealth Edison was requested to document its corrective actions with respect to the four negative trends. In its October 1, 1984 letter to Mr. R. C. DeYoung, Commonwealth Edison responded to this request. The staff has reviewed the response and its comments are provided below. Commonwealth Edison will be requested to respond to these comments and further document its corrective action program, as necessary, prior to exceeding 5% power operation. The below comments notwithstanding, the staff does not believe that these trends indicate an inadequacy in design or a pervasive breakdown in control of the design process.

a. Use of Undocumented Judgments

1. Response: Standards have been issued by Sargent & Lundy in the Electrical, Structural, and Mechanical areas via standards

ESI-253, SAS-22, and MAS-22. These standards require documenting engineering judgments.

2. Comments: Training of cognizant personnel is not discussed. Additional information will be requested of Commonwealth Edison concerning training to assure that cognizant personnel will be knowledgeable of the issued procedures. In addition, Commonwealth Edison (and Sargent & Lundy) have committed to follow Regulatory Guide 1.64, "Quality Assurance Requiremnts for the Design of Nuclear Power Plants" which endorses, with supplementary provisions, ANSI Standard N45.2.11-1974. Commonwealth Edison will be requested to provide assurance that Sargent and Lundy procedures reflect the requirements of ANSI N45.2.11-1974 with respect to documenting design activities.

b. Insufficient Control of the FSAR

1. Response: The FSAR is being updated for all Observation Reports requiring FSAR update. Other minor updates will be made in future amendments as appropriate.

2. Comments: Commonwealth Edison will be requested to further describe how it assures that the FSAR is revised to reflect the actual design of the Byron plant when design changes are made. Since the technical design adequacy has not been challenged, with the exception of high-energy line breaks which is being separately addressed, this action is also regarded as confirmatory in nature

and need not be a prerequisite to fuel loading and low power operation.

c. Insufficient Review of Changes

1. Response: Sargent and Lundy Quality Assurance Procedure GQ-3.07, Sargent and Lundy Drawings, requires a review of the drawing for technical adequacy in accordance with departmental standards. Other Quality Assurance Procedures cover design activities other than Sargent and Lundy drawings. These procedures also require that revisions be prepared, reviewed, and approved, in accordance with the same procedures as the original activity.

Bechtel concluded "The review of the S&L design process indicated that each of these processes was controlled, but IDR Observations were made for each area related to reviewing changes and coordinating them within S&L. This indicated that certain minor deficiencies may exist in the S&L process but does not lead the IDR to conclude that the process is generally inadequate."

Sargent and Lundy has, however, made the IDR Report available to the Design Directors in the Mechanical, Electrical, and Structural disciplines and has requested that the Design Directors emphasize to design personnel the requirements for the review of design changes.

2. Comments: Commonwealth Edison is committed to Regulatory Guide 1.64 and ANSI N45.2.11. Commonwealth Edison will be requested to describe actions taken to assure that its procedures and Sargent and Lundy procedures reflect provisions of ANSI N45.2.11 with respect to review of changes. Commonwealth Edison will be requested to describe training conducted to assure that personnel are aware of their responsibilities for design verification.

d. Noncompliance with Code Requirements

1. Response: Sargent and Lundy recognizes that code compliance is required and has addressed and resolved the Observation Reports that deal with OR 8.16, 8.31 and 8.49.

Furthermore, Sargent and Lundy does not consider this to be a trend. The code circumstance identified in OR 8.16 was recognized by Sargent and Lundy prior to the IDR and corrective action was being pursued. The partial penetration weld of OR 8.31 is considered to be an isolated case and OR 8.49 is a difference of opinion on an interpretation of what the code requires. Sargent and Lundy performed flange analysis in response to the OR, which demonstrates that the moment requirements of ASME Section III have been met. None of the ORs has resulted in a question of design adequacy.

In addition, with respect to the code interpretation identified in OR 8.49, Sargent and Lundy is developing a generic procedure for flange analysis. This procedure will require flange analysis for future ASME Section III piping analysis. In the interim, piping analysis personnel have been instructed to perform the flange analysis for Section III piping containing flanges.

2. Comments: Based on the type and limited number of observations noted it is not clear that this item represents a legitimate trend.

33. Conclusion: The NRC conducted an IDI at the Byron station to obtain additional assurance of the quality of design. For the most part, the IDI provided the staff additional assurance of the quality of design. However, certain deficiencies were noted. Commonwealth responded to the IDI findings, including initiation of a comprehensive independent design review of three selected systems. In addition, Commonwealth Edison initiated a confirmatory study of the design adequacy for jet impingement effects. Commonwealth Edison's responses and the additional reviews and studies have been reviewed and items have been completed or action has been identified to achieve resolution. No remaining item from the IDI or IDR is regarded as necessary to be completed prior to fuel load and low power (up to 5% of rated power) operation. However, the staff will require that all remaining items be completed prior to exceeding 5% power or that the applicant demonstrate why it may proceed to full power operation pending completion of any corrective actions. No significant design changes have been necessary to date. In light of the above and absent any new significant findings resulting from the HELB review (see Paragraph 20), I see no technical

reason to require further comprehensive design reviews of the Byron Station.

34. The conclusion in this affidavit is based on the IDI and the staff's review of the IDR. Two other matters are pending as part of the Region III inspection program which may reflect on Sargent and Lundy's control of the design process and the documentation of engineering judgments. These matters relate to steam generator snubbers supplied by Boeing and to the Byron design for pipe whip restraints. These items have or will be the subject of Board Notifications. Final staff conclusions on these matters have not been reached. At the present time, these matters do not seem pervasive and therefore do not impact the staff's conclusion in this affidavit.

James L. Milhoan

James L. Milhoan

SUBSCRIBED and SWORN
to before me this
\ day of October
1984.

NOTARY PUBLIC

JAMES L. MILHOAN

Organization:	Office of Inspection and Enforcement
Title:	Chief, Licensing Section, Quality Assurance Branch
Education:	B.S. Math and Physics, West Texas State Univ., 1963 M.S. Nuclear Engineering, Univ. of New Mexico, 1971

Experience:

1983 - Present	Chief, Licensing Section, Quality Assurance Branch Responsible for performing quality assurance licensing functions; conducting NRC Integrated Design Inspections; and, managing NRC Independent Design Verification Program. (NRC)
1982 - 1983	Technical Assistant To Commissioner Ahearne - Responsible for providing independent analyses of policy-related matters and recommendations on a full spectrum of technical and programmatic issues requiring Commission attention. (NRC)
1980 - 1982	Senior Policy Analyst - Performed independent technical analyses of policy issues and regulatory programs under consideration by the Commission. (NRC)
1972 - 1980	Senior Nuclear Engineer - Worked in positions of increasing responsibility in the AEC and NRC regulatory standards program. Developed reactor codes, standards, and criteria associated with the design, construction, and operation of nuclear power plants. Served, from 1975 - 1981, as the Alternate U.S. Member to the IAEA Technical Review Committee on Nuclear Power Plant Operations. Served for five months on the Lessons Learned Task Force in the Office of Nuclear Reactor Regulation and six months on the TMI Action Plan Steering Group which reported to the Executive Director for Operations.
1970 - 1971	Graduate School - University of New Mexico
1963 - 1970	U.S. Naval Officer - Served in the Naval Nuclear Power Program from 1965 to 1970. Assigned to the Polaris submarine, USS SAM RAYBURN (SSBN-635). Duties included SONAR Officer, Electronic Material Officer, Communications Officer, Electrical Officer, Reactor Controls Officer, and Operations Department Head. Qualified as Officer of the Deck, Diving Officer, Engineering Officer of the Watch, Ship's Duty Officer, and Engineering Duty Officer.

ATTACHMENT 2

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

SEP 11 1984

MEMORANDUM FOR: T. Novak, Assistant Director for Licensing, Division of Licensing

FROM: L. S. Rubenstein, Assistant Director for Core and Plant Systems, Division of Systems Integration

SUBJECT: BYRON INTEGRATED DESIGN INSPECTION - PROTECTION AGAINST POSTULATED PIPE BREAKS - AUXILIARY SYSTEMS BRANCH

At the request of IE and DL, the Auxiliary Systems Branch (ASB) has reviewed the Commonwealth Edison Company report entitled "Byron 1 - Confirmation of Design Adequacy for Jet Impingement Effects" dated August 1984 and the applicant's responses to the three concerns previously identified by ASB as a result of our assistance to IE in resolving the findings of the Byron Integrated Design Inspection (IDI). These concerns were documented in a memorandum from O. Parr (ASB) to T. Ankrum (IE) dated April 20, 1984. At the request of IE, we have also reviewed the applicant's response to staff concern 8.d as identified in the NRC letter dated April 9, 1984 to Commonwealth Edison Company. The enclosed evaluation covers our review of the additional information provided by the applicant regarding the assumptions used in the analysis of postulated failures in high and moderate energy lines outside containment at Byron.

Based on our review of the above information, we conclude that the applicant has satisfactorily demonstrated the adequacy of the Byron design with respect to protection against the effects of jet impingement from postulated failure of high and moderate energy lines outside containment. Therefore, the design of the facility for providing protection from the effects of high and moderate energy pipe failures outside containment is acceptable and in accordance with the acceptance criteria of SRP Section 3.6.1. This review confirms the ASB conclusions previously documented in Byron Safety Evaluation Report (NUREG-0876) Supplement No. 2. It should be noted that two of the concerns require review by the Mechanical Engineering Branch (MEB) as they involve MEB areas of responsibility. We understand that MEB will address these items in a separate evaluation. Our conclusion is contingent on the satisfactory outcome from the MEB review. We consider our assistance to IE regarding the Byron IDI to be complete.

L S Rubenstein

L. S. Rubenstein, Assistant Director
for Core and Plant Systems
Division of Systems Integration

Enclosure:
As Stated

cc w/enclosure:
See next page

Contact:
R. Anand
X29465

cc w/enclosure:
R. Bernero
O. Parr
B. J. Youngblood
R. Bosnak
T. Ankrum
D. Allison
J. Milhoan
D. Norkin
R. Paskhill
J. Wermiel
J. Rajan
L. Olshan
R. Anand

BYRON STATION, UNITS 1 AND 2
INTEGRATED DESIGN INSPECTION
PROTECTION AGAINST POSTULATED PIPE BREAKS
AUXILIARY SYSTEMS BRANCH

Introduction

As the result of the Integrated Design Inspection (IDI) of the Byron Station, two unresolved items were identified by IE which impacted safety evaluation report conclusions within the Auxiliary Systems Branch (ASB) area of responsibility. These items were documented and are identified as Finding 2-16, "Jet Impingement Analysis" and Finding 2-17, "Moderate Energy Pipe Crack Analysis." In response to these two findings Sargent-Lundy, architect-engineer for Byron, provided two reports for staff review: 1) "Verification of High Energy Line Break Design Approach for Jet Impingement Effects on Safe Shutdown Equipment" (Calculation No. 3C8-1083-001), dated February 13, 1984 and 2) "Jet Impingement Summary Documentation Report" (Report BB-JI-01) dated March 9, 1984. These two reports documented the Sargent-Lundy evaluation of the Byron plant design capability to protect against and mitigate the effects of high and moderate energy pipe breaks. At the request of IE, ASB reviewed these two reports and the applicant's December 30, 1983 response to the IDI report and identified three concerns with the assumptions used in the analyses. These concerns were documented in the memorandum dated April 20, 1984 from O. Parr (ASB) to T. Ankrum (IE). An additional staff concern, item 8.d, which also related to ASB areas of responsibility was identified in a letter to Commonwealth Edison Company dated April 19, 1984.

By letter dated August 16, 1984, Commonwealth Edison Company, the applicant for Byron Station provided responses to the above concerns and also submitted a report entitled, "Byron 1 - Confirmation of Design Adequacy for Jet Impingement Effects," dated August 1984. At the request of IE, ASB has reviewed the above report and the responses to the three previously identified concerns. The results of our evaluation of the additional information are provided below.

II. Discussion and Evaluation

A. Concern: "Commonwealth Edison letter dated December 30, 1973, in response to Finding 2-17 of the subject report, states, "in the event spray disables one AFW train and single failure disables the other, safe shutdown can be achieved per Figure 1 by feed and bleed of the primary system with or without RHR." The team considers that feed and bleed is not an acceptable alternate means of decay heat removal in the event of high and moderate energy pipe failures. Sargent & Lundy should identify specific piping breaks/cracks which could result in damage to essential decay heat removal equipment and for which feed and bleed cooling was assumed in order to achieve safe shutdown. For these cases, there should be sufficient protection to assure that at least one train of equipment would be available for an acceptable decay heat removal method."

Applicant's Response: The applicant stated that the Byron design provides the capability for feed and bleed decay heat removal for certain feedwater transients in terms of defense-in-depth for beyond design basis events. They further stated although feed and bleed cooling is part of the Byron emergency operating procedures, credit for feed and bleed is not required for safe shutdown as a result of jet impingement effects. There are no postulated high or moderate energy line breaks which would result in loss of both auxiliary feedwater trains and a reactor trip.

Evaluation: The auxiliary feedwater (AFW) system consists of a motor-driven auxiliary feedwater pump train, a diesel-driven auxiliary feedwater pump train and associated piping and valves. There are no high energy line breaks which will adversely affect the AFW system. However, spray from cracks in moderate energy lines such as service water or fire protection could effect one AFW train. However, the line failure could not cause a reactor trip or loss of offsite power. The criteria of SRP Section 3.6.1 (BTP ASB 3-1) states that loss of main feedwater need not be assumed if a reactor trip does not result from the pipe break. Thus, main feedwater or one train of AFW will be available for the above postulated moderate energy line breaks assuming a single failure.

Conclusion: Based on our review, we conclude that the plant design adequately meets the acceptance criteria of SRP Section 3.6.1, and this concern is therefore resolved.

B. Concern: "Calculation 3C8-1083-001 states that, in the event the RHR system is incapacitated, cold shutdown could be achieved by using the seconary system to remove decay heat by dumping water to the condenser and feeding the steam generators with main or auxiliary feedwater. The steam generator functions as an RHR heat exchanger. The steam generator can be flooded and the overflow will flow down the steam pipes and bypass to the condenser. We consider that this method of attaining cold shutdown in the absence of RHR is only minimally acceptable. Accordingly, you should identify all areas where pipe breaks or cracks could incapacitate the RHR system. In these areas you should perform a more rigorous jet impingement or water spray analysis (e.g., based on specific break/crack locations as opposed to Sargent and Lundy's previous practice of postulating breaks/cracks throughout the general area) to determine if the RHR system would be damaged. For these cases where this more rigorous jet impingement or water spray analysis results in the RHR system being incapacitated, you should consider modifications to protect the RHR equipment from jet impingement or water spray.

Applicant's Response: The applicant stated that stress levels in piping in the area of the RHR system outside containment are below those necessary for postulation of cracks or breaks. Therefore, no jet impingement is postulated in areas outside containment affecting the RHR system. The applicant further stated that emergency procedures will include guidelines for achieving cold shutdown using the steam generator, as a backup to the RHR system.

Evaluation: The applicant has identified an acceptable means for assuring protection of the RHR system from jet impingement effects in accordance with the criteria of SRP Section 3.6.2. It is our understanding that the Mechanical Engineering Branch (MEB) will confirm the acceptability of the stress levels for break and crack exclusion. In addition, MEB will review the effects of pipe breaks on the RHR system inside containment. MEB will provide a separate evaluation regarding these subjects. Further, while we recognize that emergency procedures include guidelines on use of steam generators for achieving cold shutdown, we can not grant credit for operation in this mode in lieu of assuring protection of the RHR system from postulated pipe breaks.

Conclusion: Based on the above, we conclude that the plant design provides adequate protection for the RHR system against postulated pipe breaks outside containment, and this concern is therefore resolved, pending the satisfactory outcome from tne MEB review.

C. Concern: "The Sargent and Lundy pipe break and crack analyses do not consider loss of offsite power concurrent with a break or crack in nonseismic Category I piping, such as the fire protection system piping. A seismic event could be expected to damage offsite power equipment as well as cause breaks and cracks in nonseismic Category I piping. Sargent & Lundy stated that all nonseismic Category I piping in safety-related areas has seismic Category I supports and is, therefore, not postulated to break or crack as the result of a seismic event. Based on our internal staff review, we consider that you have not provided sufficient information to verify that nonseismic Category I piping in safety-related areas would not fail in the event of a safe shutdown earthquake (SSE). The use of Category I supports, by itself, would not ensure that this piping would intact in an SSE. You should provide additional information to justify the position that nonseismic Category I piping with Category I supports would remain intact in an SSE. Alternatively, you should re-evaluate the consequences of breaks and cracks in non-seismic Category I piping, using the assumption that an SSE could result in piping failure concurrent with loss of offsite power.

Applicant's Response: The applicant stated that all piping in safety-related areas is designed to the requirements of ASME Section III or ANSI B31.1 and supported to withstand seismic loads. Cracks are postulated at all fittings of non-safety-related piping in safety-related areas in accordance with the guidelines of Standard Review Plan Section 3.6.2.

Evaluation: We understand the applicant's position to be that the pressure boundary in nonseismic Category I (non-safety-related) piping which is seismically supported will be maintained in the event of an SSE. The Mechanical Engineering Branch (MEB) will review the applicant's position and provide a separate evaluation regarding the concern for the capability of seismically supported piping to maintain its pressure boundary as part of their areas of review responsibility.

Conclusion: We can not provide a conclusion regarding this concern. MEB will address this item in a separate evaluation.

D. Concern 8.d: In the NRC letter to Commonwealth Edison Company dated April 9, 1984, the staff identified a concern regarding a potential water spray hazard in the component cooling water (CCW) pump area. This area contains the four Unit 1 and Unit 2 CCW pumps, a common CCW pump for both units and associated valves used to align the common pump to either unit. An essential service water line to the CCW heat exchangers and a fire protection line are also routed in this area. The staff was concerned that a combination of water spray damage and single active failure could result in loss of component cooling water to one unit.

Applicant's Response: To resolve the above concern, the applicant has committed to provide spray shields on the CCW pump motors and to install partial height walls between the CCW pumps to prevent potential water spray damage. These modifications will be completed prior to fuel load.

Evaluation and Conclusion: We conclude that the modified design as described above will provide adequate protection for the CCW system in accordance with the acceptance criteria of SRP Section 3.6.1 and is acceptable. This concern is, therefore, resolved.

III. Overall Conclusion: Based on our review of the applicant's report "Byron 1 - Confirmation of Design Adequacy for Jet Impingement Effects" dated August 1984 and the applicant's responses to our concerns as discussed above, we conclude that the applicant has adequate demonstrated the adequacy of the Byron design with respect to protection against the effects of jet impingement from postulated failure of high and moderate energy lines outside containment. Therefore, the design of the facility for providing protection from the effects of high and moderate energy pipe failures outside containment is acceptable and in accordance with the acceptance criteria of SRP Section 3.6.1. This review confirms the ASB conclusions previously documented in the Byron Safety Evaluation Report (NUREG-0876), Supplement No. 2, Section 3.6.1, pending the outcome of the MEB review as noted above.

ATTACHMENT 3

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

OCT 11 1984

MEMORANDUM FOR: Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

FROM: James P. Knight, Assistant Director
for Components and Structures Engineering
Division of Licensing

SUBJECT: BYRON INTEGRATED DESIGN INSPECTION

At the request of IE and DL, the Mechanical Engineering Branch (MEB) has reviewed the Commonwealth Edison Company report entitled "Byron 1 - Confirmation of Design Adequacy for Jet Impingement Effects" dated August 1984, and the applicants responses to staff's concerns identified in a memorandum from O. Parr to T. Ankrum (IE) dated April 20, 1984. The enclosed evaluation covers our review of all relevant information provided by the applicant regarding the assumptions used in the analysis of postulated pipe breaks in high and moderate energy lines inside and outside containment at Byron which involve MEB areas of responsibility.

for James P. Knight, Assistant Director
for Components and Structures Engineering
Division of Engineering

Enclosure : As stated

cc: w/enclosure
R. Vollmer
L. S. Rubenstein
R. P. Bosnak
B. J. Youngblood
D. Allison
R. Parkhill
J. Wermiel
J. Milhoan
S. Lewis
L. Olshan
J. Brammer
J. Rajan

BYRON STATION, UNITS 1 AND 2

INTEGRATED DESIGN INSPECTION

PROTECTION AGAINST POSTULATED

PIPE BREAKS

MECHANICAL ENGINEERING BRANCH

I. Introduction

By letter dated August 16, 1984, Commonwealth Edison Company, the applicant for Byron Station provided responses to several concerns raised by the Auxiliary Systems Branch (ASB) and also submitted a report entitled "Byron I - Confirmation of Design Adequacy for Jet Impingement Effects" dated August 1984. Two of these concerns fall within the Mechanical Engineering Branch scope of review. At the requst of IE, MEB has reviewed the above report and the responses to these two previously identified concerns. The results of our evaluation are provided below.

The evaluation in Section II below as it relates to full power operation is contingent upon the applicant justifying the use of information contained in Section 4.7.3, "Jet Impingement Load Definition" in the report referenced above. The applicant has referenced NUREG-CR-2913, "Two-Phase Jet Loads" dated January, 1983 as the basis for determining loads due to two phase and steam jets. This report is under review by the staff and several technical community peer groups. Based on a preliminary review of this report, the staff finds the methodology and the general analytical approach acceptable.

The probability of having a full area double ended pipe break (the break required to produce jet impingement loads of the type under discussion) during the short period of time, approximately two months anticipated for low power testing, is considered low. Also the consequences of a pipe break are considered to be less severe at low power than at operation at higher power levels because of much lower decay heat and smaller fission-product inventory. In the event that additional protection is required to protect against the effects of jet impingement, that protection can be provided after the plant has gone through lower power testing.

In view of the above, the MEB believes that operation at power levels up to five percent is acceptable while the staff is completing its evaluation of the report, including the following information required of the applicant:

Prior to full power operation, the staff will require that the applicant provide the specific use made of NUREG-CR-2913 by identifying all systems and each of the locations in which it is applied, and demonstrate that the use made of NUREG-CR-2913 meets the FSAR commitment on protection against the effects of postulated pipe breaks.

I . Discussion of Concerns Raised by the IDI

A. Concern: "Calculation 3C8-1083-001 states that, in the event the RHR system is incapacitated, cold shutdown could be achieved by using the secondary system to remove decay heat by dumping water to the condenser and feeding the steam generators with main or auxiliary feedwater. The steam generator functions as an RHR heat exchanger. The steam generator can be flooded and the overflow will flow down the steam pipes and bypass to the condenser. We consider that this method of attaining cold shutdown in the absence of RHR is only minimally acceptable. Accordingly, you should identify all areas where pipe breaks or cracks could incapacitate the RHR system. In these areas you should perform a more rigorous jet impingement or water spray analysis (e.g., based on specific break/crack locations as opposed to Sargent and Lundy's previous practice of postulating breaks/cracks throughout the general area to determine if the RHR system would be damaged. For these cases where this more rigorous jet impingement or water spray analysis results in the RHR system being incapacitated, you should consider modifications to protect the RHR equipment from jet impingement or water spray.

Applicant's Response: The applicant stated that stress levels in piping in the area of the RHR system outside containment are below those necessary for postulation of cracks or breaks. Therefore, no jet impingement is postulated in areas outside containment affecting the RHR system. The applicant further stated that emergency procedures will include guidelines for achieving cold shutdown using the steam generator, as a backup to the RHR system.

Evaluation: The applicant has identified an acceptable means for assuring protection of the RHR system from jet impingement effects in accordance with the criteria of SRP Section 3.6.2. The Mechanical Engineering Branch (MEB) has reviewed the stress levels from break and crack exclusion and also the effects of pipe breaks on the RHR system inside containment. Based on our review we conclude that the plant design provides adequate protection for the RHR system against postulated pipe breaks both inside and outside containment. This concern is therefore resolved. The use of the NUREG-CR does not affect this concern.

B. Concern: "The Sargent and Lundy pipe break and crack analyses do not consider loss of offsite power concurrent with a break or crack in nonseismic Category I piping, such as the fire protection system piping. A seismic event could be expected to damage offsite power equipment as well as cause breaks and cracks in nonseismic Category I Piping in safety-related areas has seismic Category I supports and is, therefore, not postulated to break or crack as the result of a seismic event. Based on our internal staff review, we consider that you have not provided sufficient information to verify that nonseismic Category I piping in safety-related areas would not fail in the event of a safe shutdown earthquake (SSE). The use of Category I supports by itself, would not ensure that this piping would remain intact in an SSE. You should provide additional information to justify the position that nonseismic Category I piping with Category I supports would remain intact in an SSE. Alternatively, you should re-evaluate the consequences of breaks and cracks in non-seismic Cagegory I piping, using the assumption that an SSE could result in piping failure concurrent with loss of offsite power.

Applicant's Response: The applicant stated that all piping in safety-related areas is designed to the requirements of ASME Section III or ANSI B31.1 and supported to withstand seismic loads. Cracks are postulated at all fittings of non-safety-related piping in safety-related areas in accordance with the guidelines of Standard Review Plan Section 3.6.2.

Evaluation: The staff concurs with the applicant's position that the pressure boundary in nonseismic Category I (non-safety-related) piping which is seismically supported will be maintained in the event of an SSE since all piping in safety-related areas is designed to meet the requirements of ASME Section III or ANSI B31.1 and supported to withstand seismic loads. The non-safety related piping in safety-related areas meet the requirements of SRP Section 3.6.2 and is therefore acceptable.

III. Overall Conclusion

Based on our review of the applicant's report "Byron 1 - Confirmation of Design Adequacy for Jet Impingement Effects" dated August 1984 and the applicant's responses to our concerns as discussed above and contingent on resolution of the use of NUREG-CR-2913 prior to full power operation, as discussed in Section 1 above, we conclude that the applicant has adequately demonstrated the adequacy of the Byron design with respect to protection against the effects of jet impingement from postulated failure of high and moderate energy lines both inside and outside containment.

Docket No. 50-454

APPLICANT: Commonwealth Edison Company (CECO)

FACILITY: – Byron Unit 1

SUBJECT: SUMMARY OF PUBLIC MEETING - BYRON INTEGRATED DESIGN INSPECTION (IDI)

On September 14, 1984, the NRC staff met in Glen Ellyn, Illinois (NRC Region III) with CECO and its agents to discuss NRC comments on the Bechtel IDR Final Report, the S&L Report on Confirmation of Design Adequacy for Jet Impingement Effects and other remaining IDI issues. Significant items from the September 14, 1984 public meeting are noted below:

a. Commonwealth Edison stated that all commitments noted in the Bechtel IDR Report were in progress. Commonwealth Edison agreed to submit an implementation schedule to the NRC staff. The NRC staff will monitor completion of corrective actions.

b. The NRC staff questioned the basis for the conclusion provided for most of the Observation Reports that "there is no reason to expect this situation is cause for a significant concern elsewhere" (for example, see IDR Report, Observation Report 8.1, Volume 1, Section 2, Page 14). Bechtel, with Commonwealth Edison concurrence, agreed to substantiate, through additional documentation, each conclusion where the above statement is made.

c. The NRC staff questioned when FSAR changes would be submitted. Commonwealth Edison agreed to provide the staff a schedule for FSAR changes submittals and stated that FSAR changes would be submitted prior to fuel loading.

d. The NRC questioned the use of the term "safety significant". Bechtel stated that their definition of "safety significant" did not affect the scope or depth of review performed during the IDR.

e. Commonwealth Edison agreed to submit its explanation regarding administrative action taken with respect to Observation Report 8.10.

f. Sargent & Lundy, with Commonwealth Edison concurrence, agreed to revise documents discussed in Observation Reports 8.21, 8.23, 8.27, and 8.29 and to provide a schedule for doing so.

g. In response to questions on Observation Report 8.32, Commonwealth Edison agreed to provide justification for the use of the 10% overstress factor for small bore supports.

h. Sargent & Lundy, with Commonwealth Edison concurrence, agreed to provide a written response to describe how it had modified past procedures to implement the commitment that engineering judgments would be documented in the future.

i. Commonwealth Edison agreed to provide a schedule for implementation of the design changes associated with Observation Reports 8.9 and 8.38.

j. Commonwealth Edison agreed to document its course of corrective action with respect to Bechtel's observed four negative trends--use of undocumented judgments, insufficient control of the FSAR, insufficient review of changes, and noncompliance with Code requirements.

k. Bechtel, with Commonwealth Edison concurrence, agreed to update the statement of Observation Report 8.49 that "it could not be established that the design process for HELBA was carried out in sufficient depth of detail to assure meeting the design objective" to reflect its subsequent review of the Sargent & Lundy Confirmatory Report entitled "Bryon 1 Confirmation of Design Adequacy for Jet Impingement."

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

ATTACHMENT 5

October 1, 1984

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

Subject: Byron Generating Station Units 1 and 2
Independent Design Inspection
NRC Inspection Report No. 50-454/83-32

References (a): August 16, 1984 letter from D. L. Farrar to J. G. Keppler.

(b): August 16, 1984 letter from Cordell Reed to R. C. DeYoung.

Dear Mr. DeYoung:

This letter provides additional information to address NRC questions raised during the review of our response to the NRC's report on their Integrated Design Inspection (IDI) and to the report of the Bechtel Independent Design Review (IDR). Submittal of this information was requested in a meeting in Glen Ellyn on September 14, 1984 and in a conference call on September 21, 1984.

Attachment A to this letter contains nearly all of the information requested of Commonwealth Edison to resolve the issues related to the IDI. The item numbers were arbitrarily assigned and do not correspond to any numbering scheme previously used. The revised FSAR pages included in Attachment B will be incorporated into the FSAR in the next amendment.

There are only three items which remain to be provided to resolve IDI/IDR concerns. FSAR changes necessary to close IDR Observation 8.47 will be provided later this week. Additional information on auxiliary building flooding will be provided to address IDI Finding 2-19 later this week. A description of the methodology used to address pipe whip in the jet impingement study provided in reference (a) will also be provided later this week.

Please address further questions regarding this matter to this office.

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

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

lm

cc: J. G. Keppler - Region III
J. Streeter
J. Milhoan

9262N

ATTACHMENT A

ADDITIONAL INFORMATION

BECHTEL RESPONSES TO NRC
FROM MEETING OF 9/14/84

Item 2 The statement, "there is no reason to expect this to be a concern elsewhere" was used frequently in close-out of observation reports. Bechtel should document the basis of why the use of this statement was appropriate for each observation report.

Bechtel Response:

Each Observation Report (OR), was analyzed and a determination was made of whether or not the OR condition was limited and not expected to be a concern elsewhere. Also, a determination was made of whether or not a safety-significant condition existed in accordance with the Program, Plan.

When it was concluded that the condition was not expected to be a concern elsewhere, the above quoted statement was made. The basis for these statements are summarized in Table-1, which give specific reasons for making that judgment on each such OR.

It should be noted that the purpose of Table-1 is only to explain the bases of non-concern elsewhere. It does not deal with resolution of the concern for the specific design work covered by the OR, which is covered by the Final Report.

In making these determinations, each OR was considered from the following standpoints: (a) can it significantly impact design performance, (b) is the condition likely to be transferred, and (c) is it relevant to other safety related designs. Also, in considering impact on design performance, the criterion was consistently applied of being able to achieve safe-shutdown. Using these standards, the IDR Team thus concluded that in the case of each OR "there is no reason to expect this situation is cause for a significant concern elsewhere."

Item 5 We agreed to discuss if any component could not perform its function.

Bechtel Response:

There were no cases where, to the knowledge of the IDR team, any reviewed safety-related component was found which could not perform its intended safety function.

Item 5 (Cont'd)

There was an instance, documented by OR 8.24, of potential damage to portions of the CCW or ESW systems piping, from postulated HELB associated jet forces determined to exist. However, in each case identified in that O.R., the IDR team concluded the affected portions of the systems had no safety function relative to achieving safe plant shutdown for the specific postulated breaks associated with each case.

Another Observation Report, OR 8.38, merits discussion relative to this item. An unanticipated consequence of the issuance of OR 8.38 was the conservative decision by Westinghouse to make a 10CFR21 report to the NRC regarding a potential overpressure condition in the CCW system caused by postulated primary coolant in-leakage to that system. Subsequent Westinghouse clarification was that the decision to make the report was based on generic system design information and not as a result of Byron specific analysis. It was the judgement of the IDR team that, for Byron, such an overpressure condition occurrence would not be expected to cause loss of system function such that loss of capability to achieve safe shutdown would occur.

Item 14

Bechtel was requested to document their present review of the S&L High Energy Line Break Report and provide a description or final statement of how Observation Report 8.47 could be closed out.

Bechtel Response

Regarding the HELB/MELB Confirmatory Report on jet impingement, the IDR Team has reviewed it for responsiveness to OR 8.47 and concludes it meets the resolution commitment. That is, the Report covers the appropriate scope, uses necessary criteria, clearly presents results, and makes an organized, controlled review of design for jet impingement. The IDR team did not review the Confirmatory Report for technical adequacy. However the Report does satisfy the concern for design process identified in OR 8.47. The results reported (no design changes required) evidences that an adequate design process had existed to achieve such results.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
8.1	SRV_Discharge Path	This, as with other minor discrepancies in the FSAR, was a random occurrence. The observation was issued as a result of IDR need to treat each FSAR state-ment as a licensing commitment. No reason was identified by the IDR team for expecting any similar FSAR problems to represent concern for the adequacy of other systems or to have any adverse impact on the plant's ability to achieve a safe shutdown condition.
8.2	Column Baseplate Thickness	The issue was one of insufficient documentation of engineering judgment and not one of adequacy. The IDR concluded there was no real cause for concern elsewhere, because a similar application of judgment would have produced a similar result.
8.3	Alarms for ESW Makeup Pumps	Same as for OR 8.1
8.4	Burial Depth of ESW Pipes	Same as for OR 8.1
8.5	Seismic Analysis for Screenhouse	Same as for OR 8.2
8.6	Valve Disc Require-ments	Same as for OR 8.1
8.9	Relay Protection in 125 V-dc system	A review of the S&L drawings has identified no other instance where non-Class 1E instruments fed from Class 1E power supplies are connected up-stream of the second isolation breaker without fuses. Also, it was concluded the application of these dc instruments does not degrade the Class 1E dc bus below an acceptable level, even without the additional fuses.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
·8.10	Battery Capacity	This condition is not likely to be a problem with the ac system because conservative estimates of the Class 1E ac loads are required by RG 1.9. Further, the SER indicates that the electrical design had previously been reviewed for compliance with RG 1.9 and had been found acceptable. Conservative assumptions of electrical loads have been found in all other cases reviewed by the IDR Team.
8.14	ESW Makeup Pumps Seismic Qualification	This appeared to be a random discrepancy since other items such as the structure and piping were reviewed for the new spectra. Also, only the river screenhouse spectra were revised, at that time, and not those for the other Seismic Category 1 buildings.
8.16	Component Support Weld Sizes	The issue was that an S&L document addressing weld design did not require weld size in strict conformance with the applicable portion of the ASME B&PV Code. The IDR team judged that design was adequate since S&L analysis had established that such welds met stress limits and further qualification of the welds had been performed. While the particular situation exists throughout the design, the IDR team concluded that the other welds would likewise be adequate. While S&L had already applied for a Code case (to allow the situation) prior to the IDR, CECO decided to review all affected welds on all systems to bring them into strict code conformance.
8.17	Structural Steel Weld Size	The issue was similar to OR 8.16 as it relates to conformance to the AISC Code for structural steel weld sizes. The IDR team conclusion was similar to that of OR 8.16. The welds reviewed by the IDR, and those for other safety-related structures, were done to a qualified weld procedure, and the welds had been qualified for strength requirements.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
8.19	NPS Pipe Support Calculation Review	The IDR team, upon receipt of clarifying information, concluded no discrepancy existed.
8.21	Interchangeable Components	The IDR team concluded the situation was unique for Corner & Lada pipe support components and was satisfied with the existing situation, once clarification was received from S&L regarding field commodity control procedures.
8.22	ESW Piping Design Pressures	The issue was one of compliance with the Code and not one of adequacy. Although the higher pressure conditions were not code required, the piping was capable of withstanding these improbable higher pressures. It was shown that there was actually Code compliance.
8.23	ESW Valve Testing	The issue was one of inconsistency between the FSAR and procurement specifications and not one of adequacy. The supplier did, in fact, test the valves. If valves are not tested in the shop they are tested during preoperational testing.
8.25	Stress Calc. 1SX-17	The issue was one of clarity in defining changes in pipe support locations and not one of adequacy. The final piping stress report including addenda does match the actual piping support configuration.
8.27	Pump & Valve Testing	The issue was a minor inconsistency between the FSAR and procurement specifications and not one of adequacy. Testing requirements have been met or will be met during preoperational testing.
8.28	CCW Electrical Penetrations	The issue was one of readily locating documents. Upon receipt of clarification by S&L, the IDR concluded that no discrepancy existed which would adversely affect the intended safety function of the components. This was supported by a review of a significant number of additional packages.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
8.29	Non-pressure Boundary Stress Criteria	The issue was one of documentation which raised a concern of review adequacy. However, upon clarification by S&L of its standard practice, the IDR concluded that S&L had an adequate review process and that it functioned. This was supported by a significant sample of valve stress analyses.
8.31	CCW Partial-Pressure Welds	This issue was one of AWSD1.1 code compliance and there was no concern that the weld in question was adequate to perform the intended safety function. An extensive S&L review of other welds established that this discrepancy was a unique occurrence.
8.32	Aux. steel support overstress	This observation related to a convenient and technically justifiable design practice which used terminology ("overstress factor") which appeared to lack complianc~ with the AISC Code. It was established no discrepancy existed.
8.34	Welded Connections	The issue was one of the lack of adequate documentation of weld design review. The IDR concluded the weld was adequate, based on analysis, and, therefore, the application of judgment was effective. The IDR further concluded that such similar application of judgment for other safety-related systems would have produced an adequate design.
8.35	Piping Support Calculations	The issue was one of documentation of design change review judgments. The IDR concluded the situations reviewed were adequate and that the judgment application was substantiated. The IDR further concluded similar applications of judgments for other safety systems would have produced an adequate design.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
8.36	Expansion plates	The observation dealt with a question of the adequacy of the design margin to meet the NRC IE Bulletin 79-02 requirement provided by a S&L design standard. S&L provided a calculation of an appropriate limiting condition and the IDR team accepted the calculation as demonstrating the standard's adequacy. The standard was used throughout the plant, and since it was judged adequate, no concern exists for its application elsewhere.
8.37	Support Swing Angle Limit	The Observation dealt with a question of the adequacy of the design procedure to ensure proper application of component supports. Clarification by S&L of the design process, and also of the checks of the conditions in question by walkdowns during hot functional testing satisfied the IDR team that an adequate, controlled process existed. Therefore, the process was judged adequate.
8.38	CCW Design Pressure	The Observation originally dealt with the adequacy of the selection of the ASME B&PV Code design pressure for the CCW system. The S&L response on this point was judged adequate by the IDR team. The Observation resulted, for other reasons, in the designer (Westinghouse) notifying the NRC of a 10CFR21 situation as a result of an identified potential overpressure condition. The IDR team judged that the situation was such that the plant's capability to achieve safe plant shutdown was not adversely affected. Also, the IDR team judged that the CCW design, and the effects on it which might lead to the postulated overpressure condition, was unique compared to other safety-related systems, and no concern existed that the situation would be replicated for other plant systems.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
8.39	Power Cables in Cable Spreading Room	The resolution of the observation pointed out that the uncovered power cables actually was not included within the defined area of the cable spreading areas. Therefore, this was not a deficiency.
8.40	Cable Separation	Other manhole drawings were checked and did not contain any conflicting lines or any lines at all. Therefore, this appeared to be isolated to the subject drawings. Field inspection showed that the cables are installed correctly.
8.41	Motor Operated Valve Operators	To resolve this observation S&L performed a calculation to verify that MOVs required to function upon a safety signal will perform their safety function. In this calculation S&L included MOVs of all safety related systems in the plant. S&L expanded the scope of this observation to assure that the design of power supplies to 460V motors and MOVs of other systems are adequate with regard to this concern.
8.42	Cable Saddles in Manholes	Since the design of the cable saddle was proven as adequate, use of these saddles elsewhere would also be acceptable.
8.44	CCW Nozzle Loads	The Observation dealt with a question of whether the designer's judgment that the effects of thermal growth produced insignificant stress levels and nozzle loads was justified. After extensive review within S&L and by Bechtel, the IDR team judged the configuration in question to be unique, and concluded that there was no reason for concern with similar judgments elsewhere in the plant.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
8.47	HELB Jet Impingement	The design process for HELB jet impingement effects is considered adequate for the entire plant based on the process identified by the IDR and supplemented by the results reported in "Confirmation of Design Adequacy for Jet Impingement Effects" which examined postulated breaks plant-wide for jet effects and reported that no plant modifications were required.
8.49	ESW and CCW Piping Flanges	The Observation dealt with a question of strict compliance with the ASME BP&V Code, as interpreted by the IDR team. In this case, there was a difference of opinion on code interpretation. S&L calculations for limiting conditions demonstrated design adequacy. There was a conclusion on the part of the IDR team that a technically adequate situation existed throughout the plant, and the interpretation of the ASME Code did not in any way affect any safety-related system's capability to perform its intended safety function. Despite extensive reviews for code compliance, no significant deficiencies were found elsewhere.

SARGENT & LUNDY RESPONSE TO NRC FROM MEETING OF 09-14-84

Item 1

Edison has agreed to do all items committed in the Bechtel Report. Sargent & Lundy has developed a tracking mechanism for Byron I and will make periodic submittals of the close-out status to the affected project distribution. In addition, the Bechtel Report should be reviewed for actions to be taken on Byron II and Briadwood I and II. A similar tracking mechanism will be developed and distributed.

Sargent & Lundy Response

A tracking mechanism has been developed for the Byron Unit I IDR. The only remaining open item is the required FSAR update resulting from OR 8.47 dealing with HELB. The applicability of any Byron I IDR Commitments will be tracked and implemented as appropriate for Byron II and Braidwood I and II.

Item 3

A schedule for updating the FSAR for those items committed in the Bechtel Report should be provided.

Sargent & Lundy Response

All items are attached except the changes associated with OR 8.47 which will be submitted the week of October 1, 1984.

Item 4

In discussing the battery cross-tie, we agreed to document the operating limitations. We will prepare a discussion with input from CECo Operating Station personnel.

Sargent & Lundy Response

The dc cross-tie consists of a manually operated breaker at each end of the cross-tie (i.e., one manually operated breaker in the Unit 2 dc distribution center, and one in the Unit 1 dc distribution center). All cross-tie breakers are normally padlocked in the open position with administrative controls on release of keys. Use of the dc cross-tie is presently limited by Station Technical Specifications such that at least

Item 4 - Sargent & Lundy Response (Cont'd)

one of the two units must be in either a cold shutdown or refueling mode of operation (Modes 5 or 6). The purpose of the cross-tie is to supply dc power to some of the loads in the bus of the "down" unit when, and if, it is desirable to isolate the battery of the "down" unit for maintenance or testing.

With one unit shutdown (Mode 5 or 6), the operating procedures for closing the dc cross-tie ACBs, including the limitation on the allowable cross-tie load, will include the following:

1. Specific circuit breakers on the distribution panel will be opened to ensure that the cross-tie load will be properly limited.

2. The cross-tie breakers at Bus 111 and 211 will then be unlocked and closed. (Note that a "cross-tie ACB closed" alarm at the MCB annunciator will alert the Control Room operator when the ACBs are closed.)

3. The battery breaker at Bus 111 would then be opened (note that a "battery 111 ACB open" alarm on the MCB annunciator will alert the Control Room operator when the breaker is opened).

With this procedure, closing the dc cross-tie is an administratively controlled procedure in which the load circuit breakers are opened in a deliberate and preplanned order, prior to closing of the cross-tie breakers and disconnecting the battery.

The reconnection of Battery 111, the opening of the cross-tie breakers, and the closing of the load circuit breakers for return to normal operation, will be carried out in the reverse order, again using documented procedures/checklist and administrative controls.

Item 7 (Observation Report 8.21)

Provide a schedule for clarified and revised drawings for OR 8.21.

Sargent & Lundy Response

The drawings are currently being revised to clarify the interchangeability of safety and non-safety related hanger parts. The drawings are scheduled to be revised, reviewed, and approved by about October 5, 1984.

Item 8 (Observation Report 8.29)

Provide a commitment and a schedule to change the Sargent & Lundy design procedures to document when active allowable stresses were used rather than passive values.

Sargent & Lundy Response

No change to Sargent & Lundy procedures are required because a change to the procedure has already been made which addresses the documentation of the allowable stress values used. Sargent & Lundy is currently using, and has used since November 1982, a revised checklist which requires the reviewer to list the total stresses and the allowable stress values at critical locations. This allows an auditor to determine whether active or passive allowables were used by the reviewer and satisfies documentation requirements.

Item 9 (Observation Report 8.32)

Provide an expanded basis why a 10% over-stress is not a problem. The answer should address both the past and future.

Sargent & Lundy Response

For the assessment of as-built small bore pipe supports, a criterion was established such that up to a 10% calculated over stress was considered acceptable before additional calculations were required to establish code compliance. This was due to the fact that the hanger analysis was known to be very conservative, and that refinements to this analysis would demonstrate that the hanger met all applicable design requirements.

The known conservatism in small bore pipe support design include conservative loadings and conservative analysis techniques.

Loadings

The design loads used for small bore pipe supports are conservative because each support is designed for the peak plant seismic excitation. The actual excitation of any wall or slab in the plant can be much smaller than the peak excitation. This is a simplifying loading assumption which is reasonable considering the small amounts of material required for small bore pipe supports. Also, the loads used are not based on the actual gravity load on a given support but rather the upper bound load. This is because small bore pipe supports are chosen by the contractor from a table based on allowable loads. The support load always falls between two table capacities. For example, if support Detail 1 on the table is designed for 50 pounds and Detail 2 is designed for 100 pounds, a contractor with a 60 pound load must choose Detail 2. The results of the use of design tables and the use of peak plant acceleration values is a very conservative design load on any given small bore pipe support.

Item 9 - Sargent & Lundy Response (Cont'd)

Analysis

The analysis technique used for small bore piping analysis involves a simplified method of piping analysis which gives conservative piping loads at the supports. This method basically considers one support at a time. This is a very conservative analytical procedure. A detailed dynamic computer analysis of the piping including all supports will always give smaller calculated pipe support reactions.

Conclusion

The criterion to allow an apparent 10% increase above design allowables when simplified design methods are used is justified because these conservative engineering methods of determining loads and performing analysis for small bore pipe support design would not result in an actual over-stress if specific calculations were made.

Item 10 (Observation Reports 8.34 and 8.35)

Describe the basis for the engineering judgement that was used on these two items. Discuss the relationship of the departmental standards with respect to these items also.

Sargent & Lundy Response

The calculation for 1CC01009R indicates that the connection design was performed by utilizing the Review Manual with additional hand calculations. This "Review Manual" contains design guidelines and assumptions. These design guidelines and assumptions apply to standard hanger configurations with member sizes and weld requirements and contain associated load tables. The load tables have, among other things, built in considerations of the effects of installation tolerances and member deflections.

The original hand calculation performed verified the adequacy of the plate. Comparison of the weld capacities in the "Review Manual" provided a basis for weld adequacy. Documentation for weld adequacy has been provided in a revision to the origianl calculation.

Item 10 - Sargent & Lundy Response (Cont'd)

OR 8.34 (4.2) Pipe Support 1CC01047, (4.3) Pipe Support 1CC01042 and (4.4) Pipe Support 1CC01034

The weld evaluation of the specified flare-bevel weld on the support drawings 1CC01034, 1CC01042 and 1CC01047 was based on engineering judgement. The judgement was made by comparing the actual load to the maximum load carrying capability of the strut (all three supports are Elcen Size 2 Struts).

Maximum load carrying capability of the Elcen Size 2 Strut is:

Strut Design Load	Strut Emergency Load
2870 lbs.	3710 lbs.

Piping loads on the support drawing are:

	Actual Design Load	Actual Emergency Load
1CC01034	831 lbs.	1597 lbs.
1CC01042	421 lbs.	1018 lbs.
1CC01047	647 lbs.	1344 lbs.

The piping loads are less than 50% of the load as tabulated above. The flare-bevel weld (the effective throat of the flare-bevel weld is 0.156" compared to 0.176" for the fillet weld) was judged to be adequate for the actual design and emergency loads.

Sargent & Lundy has performed calculations to verify the engineering judgement. The calculation demonstrated that the design as specified is acceptable.

OR 8.35, Item 4.1, Pipe Support 1CC01010X

The original weld configuration - outside and inside weld at both flanges - was based on an "Emergency" load of 6967 lbs. Through subsequent minor revisions, this weld configuration remained the same even though the actual "Emergency" load was reduced by almost one-half to 3639 lbs.

The weld configuration was subsequently changed by omitting the weld at the inside of both flanges. The weld configuration prior to this change had a design margin of approximately 5 to 1. The judgement to reduce the weld section was based on the actual loading for the support. Calculations have been performed verifying this judgement. The design margin for the weld as revised was in excess of 2 to 1.

Item 10 - Sargent & Lundy Response (Cont'd)

OR 8.35 (4.2) Piping Support 1CC01051X

Sargent & Lundy has developed standard concrete expansion anchor tables and charts for given anchor bolt assemblies. These tables and charts allow a graphical selection of expansion anchor sizes. For Support 1CC01051X, the support design was changed from a 4-bolt assembly to an 8-bolt assembly. New calculations on the 8-bolt assembly were not generated since the strength of the two assemblies can be determined by comparing two charts in the standard. As a result of this observation, calculations have been generated verifying that the determination that was made by comparing the two charts was accurate.

OR 8.35, Item 4.3, Pipe Support 1CC01012R

The calculation accounts for the location tolerance and the proper load for Support No. M-1CC01012R and M-1CC14009R utilizing the "Review Manual" which was referenced in the calculation. No engineering judgement was used.

Documentation of Engineering Judgements

In the future, engineering judgements similar to those described above will be documented as required by the following Sargent & Lundy standards that are in place:

Electrical Standard	ESI-253
Structural Standard	SAS-22
Mechanical Standard	MAS-22

Item 13 (Observation Reports 8.23 and 8.27)

We agreed to revise the specifications or the FSAR as necessary to clarify the testing requirements to aid future purchases. A schedule for these revisions should be provided.

Sargent & Lundy Response

The specification and the FSAR have been reviewed relative to the in-shop testing requirements for pumps and valves. The FSAR is being revised to clarify the testing requirements. The specifications contain all of the necessary testing requirements and do not require revision.

Item 15

Provide a summary of corrective actions taken as a result of the trends shown on Page 72. Discuss that no corrective action was needed on code items.

Sargent & Lundy Response

The following actions have been taken by Sargent & Lundy relative to the trends identified on Page 72 of Volume I of the Bechtel IDR.

Use of Undocumented Judgements

Standards have been issued by Sargent & Lundy in the Electrical, Structural, and Mechanical areas via Standards ESI-253, SAS-22, and MES-22, these standards require documenting engineering judgements.

Insufficient Control of the FSAR

The FSAR is being updated for all Observation Reports requiring FSAR update. Other minor updates will be made in future amendments as appropriate.

Insufficient Review of Changes

Sargent & Lundy Quality Assurance Procedure GQ-3.07, Sargent & Lundy Drawings, requires that the reviewer of the drawing review the drawing for technical adequacy in accordance with departmental standards. Other Quality Assurance Procedures cover design activities other than Sargent & Lundy drawings. These procedures also require that revisions be prepared, reviewed, and approved, in accordance with the same procedures as the original activity.

Bechtel concluded "The review of the S&L design process indicated that each of these processes was controlled, but IDR Observations were made for each area related to reviewing changes and coordinating them within S&L. This indicated that certain minor deficiencies may exist in the S&L process but does not lead the IDR to conclude that the process is generally inadequate."

Sargent & Lundy has, however, made the IDR Report available to the Design Directors in the Mechanical, Electrical, and Structural disciplines and has requested that the Design Directors emphasize the requirements for the review of design changes to design personnel.

Item 15 - Sargent & Lundy Response (Cont'd)

Noncompliance with Code Requirements

Sargent & Lundy recognizes that code compliance is required and has addressed and resolved the Observation Reports that deal with OR 8.16, 8.31 and 8.49.

Furthermore, Sargent & Lundy does not consider this to be a trend. The code circumstance identified in OR 8.16 was recognized by Sargent & Lundy prior to the IDR and corrective action was being pursued. The partial penetration weld of OR 8.31 is considered to be an isolated case and OR 8.49 is a difference of opinion on an interpretation of what the code requires. Sargent & Lundy performed flange analysis in response to the OR, which demonstrates that the moment requirements of ASME Section III have been met. None of the OR's have resulted in a question of design adequacy including OR 8.49.

In addition, with respect to the code interpretation identified in OR 8.49, Sargent & Lundy is developing a generic procedure for flange analysis. This procedure will require flange analysis for future ASME Section III piping analysis. In the interim, piping analysis personnel have been instructed to perform the flange analysis for Section III piping containing flanges.

Item 16

Change the appropriate page in the FSAR to state that the valve performs an isolation function not a throttling function.

Sargent & Lundy Response

The required FSAR change is attached.

HIGH ENERGY LINE BREAK REPORT

Item 1

A phone call will be held with the NRC the week of September 21, 1984, to discuss hinge points, whipping pipe, secondary hinges, shape of breaks, zone of influence, etc.

Sargent & Lundy Response

The phone call was held and Sargent & Lundy will provide the one additional item requested as a result of the phone call for submittal to the NRC the week of October 1, 1984.

Item 2

Provide a schedule to revise the FSAR to make it consistent with the High Energy Line Break Report. We should make sure that we reflect the existence of the existing jet impingement shields and the various longitudinal break locations.

Sargent & Lundy Response

The FSAR update will be submitted to CECo the week of October 1, 1984.

Item 3

Provide a copy of the Westinghouse letter which agreed with the confirmatory High Energy Line Report.

Sargent & Lundy Response

Copy attached.

CAW-7732
CBW-4754

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Operations Division

Box 355
Pittsburgh Pennsylvania 15230

August 1, 1984

Ref: SLWC-3121, 7/26/84

Mr. D. L. Leone, Project Director
Sargent and Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

Attention: K. J. Green

COMMONWEALTH EDISON COMPANY
BYRON AND BRAIDWOOD STATIONS - UNITS 1 AND 2
SARGENT AND LUNDY JET IMPINGEMENT REPORT - WESTINGHOUSE REVIEW

Dear Mr. Leone:

Per your request, Westinghouse has reviewed the subject draft report and has no comments.

Our staff had reviewed a previous draft and our comments have been incorporated.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

J. L. Tain Jr.

W. E. Kortier, Manager
Commonwealth Edison Projects

JLT/bms/3645D

D. L. Leone, 30L

cc: J. D. Deress, 2L
C. W. Fruehe, 2L
K. J. Green, 1L
W. C. Cleff, 1L

COMMONWEALTH EDISON COMPANY FIELD
VERIFICATIONS IN RESPONSE TO
NRC MEETING OF 9-14-84

Item 6 (Observation Report 8.16)

Provide status of NF weld size review and a schedule for completion.

Commonwealth Edison Co. Response

The program regarding the NF weld size matter has been completed and the component supports have Q.C. inspections verifying that subsection NF minimum fillet weld size requirements have been met. You will recall that resolution on this item was in progress prior to the Bechtel IDR.

Item 11

Provide a status and a schedule for the completion of this design change including its implementation in the field.

Commonwealth Edison Co. Response

The design change for the revision to the CCW system has been issued. Field completion should occur by about 10-22-84.

Item 12 (Observation Report 8.9)

Provide confirmation if the fuse has been added in the field.

Commonwealth Edison Co. Response

New fuse blocks are currently being purchased and are anticipated to be installed by about October 12, 1984.

ATTACHMENT B

10/1/84

Bechtel Observation Report Number	FSAR Pages Changed
8.1	9.2-17
8.3	9.2-31
8.4	Q10.8-1
8.6	3.9-96
8.14	3.9-94
8.23	3.9-50, 51
8.27	3.9-47
8.38	9.2-16, 9.2-17

In addition, FSAR page 9.2-19 has been revised per discussion on page A.2-34 in Volume II of the Bechtel Final Report.

3.9.3.2.1 Pumps

Balance of Plant

All active pumps as listed in Table 3.9-15 are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts; and (2) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor parameters. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic inservice inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

NSSS

All active pumps, listed in Table 3.9-15 are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 150% of the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, and (2) performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer based on the bearing material, clearances, oil type, and rotational speed. These limits are approved by Westinghouse. After the pump is installed in the plant, it undergoes the cold hydro tests, hot functional tests, and the required periodic inservice inspection and operation. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant.

In addition to these tests, the safety-related active pumps are qualified for operability by assuring that the pump will start up, continue operating, and not be damaged during the faulted condition.

The pump manufacturer is required to show by analysis correlated by tests, prototype tests, or existing documented data that the pump will perform its safety function when subjected to loads imposed by the maximum seismic accelerations and the maximum

In case the natural frequency is found to be below 33 Hz, a dynamic or pseudo dynamic analysis is performed to determine the amplified input accelerations necessary to perform the stress analysis.

b. Additional loads considered in the stress analysis of the pumps and their supports are the nozzle loads for the applicable plant condition from interconnecting piping systems.

c. In addition to the stress analysis, a static shaft deflection analysis of the rotor is performed. The deflection determined from the static shaft analysis is compared to the allowable rotor clearances.

d. To complete the seismic qualification procedures, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation during the maximum seismic event in accordance with IEEE Standard 344-1975 (see Section 3.10). In the analysis interaction between the pump and motor is considered.

e. Alternatively, the entire pump assembly with appurtenances may be qualified by testing in accordance with IEEE Standard 344-1975. In performing the seismic testing the nozzle loads for the applicable plant condition must be applied.

From this, it is concluded that the safety-related pump/motor assemblies will not be damaged, will continue operating under SSE loadings and will perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

3.9.3.2.2 Valves

Balance of Plant

Safety-related active valves as listed in Table 3.9-16 must perform their mechanical motion in times of an accident. Assurance must be supplied that these valves will operate during a seismic event. Qualification tests and/or analyses have been conducted for all active valves to assure valve operability under seismic and/or environmental conditions.

The valves are subjected to testing prior to service (in-shop and preoperational-field) and in situ (during plant life) as required by specific service and functional requirements.

In-shop tests include the following: a) ASME Code - required hydrostatic tests to assure pressure boundary integrity;

b) Specified conformance to Manufacturers' Standard Practice code requirements regarding hydrostatic tests and main seat leakage; c) Specified timed operational tests (valve stroking) when additional verification of design requirements is necessary.

Cold hydro qualification tests, hot functional qualification tests, and periodic inservice operation are performed in situ to verify and ensure the functional ability of the valve. These tests and appropriate maintenance ensure operability of the valve for the design life of the plant. The valves are designed using either the standard or the alternate design rules of ASME III.

On all active valves, an analysis of the extended structure is also performed for static equivalent seismic loads applied at the center of gravity of the extended structure. The maximum stresses and deflection allowed in these analyses demonstrate operability and structural integrity.

Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves and safety-relief valves, are considered separately.

Due to the particular simple characteristics of the check valves, they will be qualified by a combination of the following tests and analysis:

a. stress analysis including the seismic loads where applicable,

b. in-shop hydrostatic tests,

c. in-shop seat leakage tests, and

d. periodic in situ valve exercising and inspection to ensure the functional capability of the valve.

The safety/relief valves are qualified by the following procedures. These valves are also subjected to tests and analysis similar to check valves; stress analyses including the seismic loads, in-shop hydrostatic seat leakage and performance tests. In addition to these tests, periodic in situ valve inspection, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation are performed to ensure the functional capability of the valve.

Using the methods described, all the safety-related active valves in the systems are qualified for operability during a seismic event. These methods, proposed conservatively, simulate the seismic event and ensure that the active valves will perform their safety-related function when necessary.

TABLE 3.9-8

DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS

CONDITION	DESIGN CRITERIA*
Design and Normal	ASME Section III Subsection NC-3400 and ND-3400
Upset	$\sigma_m \leq 1.0$ S $\sigma_m + \sigma_b \leq 1.5$ S
Emergency	$\sigma_m \leq 1.2$ S $\sigma_m + \sigma_b \leq 1.65$ S
Faulted	$\sigma_m \leq 1.2$ S $\sigma_m + \sigma_b \leq 1.8$ S

*The stress limits specified for active pumps are more restrictive than the ASME III limits. For the Faulted Condition (membrane plus bending), stresses may exceed 1.8 S but must remain below the material yield stress. In such cases, a deflection analysis is performed to assure that the maximum displacements are within the deflection limits which will not impair the operability of the equipment.

TABLE 3.9-9 (Cont'd)

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet.

5. The maximum pressure resulting from upset, emergency, or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.

6. Stress limits are taken from ASME III, Subsections NC and ND, or, for valves procured prior to the incorporation of these limits into ASME III, from Code Case 1635.

9.2.2.4.1 System Availability and Reliability

Either unit may be aligned with two completely independent, parallel trains, each consisting of one pump and one component cooling heat exchanger. Either train provides sufficient cooling to accommodate the heat loads experienced by that unit during a loss-of-coolant accident. Hence, any single active or passive failure in the system does not prevent it from performing its design function.

Inside the containment, most of the piping, valves and instrumentation are located outside the shield wall at a location above the calculated water level in the bottom of the containment at postaccident conditions. In this location, the portions of the system within the containment are protected against missiles and against flooding during postaccident operations. This location also provides radiation shielding which permits maintenance and inspection to be performed during normal power operation.

The component cooling pumps, heat exchangers, surge tanks and associated valves, piping and instrumentation are located outside the containment and are, therefore, available for maintenance and inspection during power operation. Replacement of a pump or heat exchanger may be performed in accordance with technical specification limitations while the other units are in service.

Sufficient cooling capacity is provided to fulfill all system requirements under normal and accident conditions. Adequate safety margins are included in the size and number of components to preclude the possibility of a component malfunction adversely affecting operation of safety features equipment. The relief valves on the component cooling water lines downstream from each reactor cooling pump are designed with a capacity equal to the maximum rate at which reactor coolant can enter the component cooling system for a severance-type break of the reactor coolant pump thermal barrier cooling coil. The valve set pressure equals the design pressure of the component cooling piping.

The relief valves on the cooling water lines downstream from the sample, excess letdown, letdown, seal water, spent fuel pit, and residual heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated and high-temperature coolant flows through the tube side. The set pressure equals the design pressure of the shell side of the heat exchangers.

9.2.2.4.2 Leakage Provisions and Activity Release

Welded construction is used where possible throughout the Component Cooling system piping, valves and equipment to minimize the possibility of leakage. The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the chemical and volume control, the sampling, the residual heat removal or the spent fuel pit cooling systems or due to a leak in the cooling coil for the reactor coolant pump thermal barrier.

Leakage from or to the component cooling system can be detected by a change of level in the component cooling surge tank. The rate of water-level change and the area of the water surface in the tank permit determination of the leakage rate. In-leakage is detected anytime by radiation monitors located on the main return headers. To assure accurate determinations, the operator must check that temperatures are stable.

A cooling water temperature increase of about 250° F in one of the units would be required to overfill its component cooling surge tank. However, should a large leak develop in a residual heat exhanger, letdown heat exhanger, or due to a ruptured reactor coolant pump thermal barrier, the water level in the component cooling surge tank of that unit would rise, and the operator would be alerted by a high-water level alarm. The vent on the surge tank is automatically closed in the event of high radiation level detected at the component cooling heat exchanger discharge header. If the leaking component is not isolated from the loop before the inflow fills the surge tank, the overflow line with a loop seal will prevent component cooling system overpressurization. The overflow is routed to the chromated drains system.

Three heat exchangers are provided to serve the two units. During all conditions of plant operation, this provides for one backup exchanger. If all three exchangers are available, however, the backup exchanger may be employed on the unit undergoing a LOCA or shutdown (RHR heat exchanger in operation). Design cooldown rates are determined on this basis (two exchangers operating on the unit recovering from a LOCA or shutdown), but the consequence of the loss of one heat exchanger during this time only slows down the cooldown rate from the design value and does not affect the safe operation of the plant.

Five pumps are provided to serve the two units. Under the limiting case, four pumps are required for the two units leaving one pump as backup pump for either unit.

9.2.2.4.3 Incident Control

Containment isolation valves are automatically closed on a safety features actuation "T" signal. The cooling water supply header to the reactor coolant pumps contains a check valve inside and remotely operated valves outside the containment wall. The

The instrumentation in the CCWS is provided primarily for initial system flow balancing and for monitoring purposes during normal operation. Thus failure of any of this instrumentation has no effect on system performance. Exceptions to this are:

a. letdown heat exchanger CCWS flow controllers,

b. reactor coolant pump thermal barrier outlet flow controller, and

c. component cooling surge tank radiation control valve.

The letdown heat exchanger tube side outlet temperature controls a butterfly valve which regulates the CCWS flow to the shell side of this heat exchanger. Should the controller fail in a way to shut off CCWS flow to the circuit, a high temperature alarm will sound in the control room allowing the operator to take corrective action.

Safety-related indication of component cooling water flow from the reactor coolant pump motor oil coolers is provided and alarmed in the main control board. The reactor coolant pump (RCP) thermal barrier outlet header has a flow controller which causes a motor-operated valve to close in this line in the event of high flow (an indication of a broken RCP thermal barrier). Should the controller not operate properly, an increasing level is noted in the CCWS surge tank, resulting in a high level alarm, if not isolated. A second motor-operated valve in series with the flow control valve is available for manual isolation of the line if required. Additionally, two level instruments are provided on each surge tank, both of which will give a high level alarm in the control room.

Each component cooling surge tank vent has an air operated valve which will close on a high radiation signal from the radiation monitors in the discharge headers from the CCWS heat exchangers. This high radiation alarm normally indicates a primary to CCWS leak. Three radiation monitors are provided. The monitor on the common heat exchanger will alarm and close the vent valve on both surge tanks. The radiation monitors on each unit's heat exchanger will alarm and close its respective surge tank vent valve.

9.2.2.4.5 Electrical Power Supply

The normal power supply to the system is from the ESF buses. A full description of the power supply is given in Subsection 8.3.1.1.

9.2.2.5 Tests and Inspections

During the life of the Station, the Component Cooling System is in continuous operation and performance tests are not required. Standby pumps are rotated in service on a scheduled basis to obtain even wear. Preoperational tests are performed on the system. The equipment manufacturer's recommendations and station practices are considered in determining required maintenance.

The worst case heat transfer to atmosphere condition of 82° F wet bulb for 3 hours on July 30, 1961 would result in a cold water outlet temperature of 94.6° F at a heat rejection rate of 580 x 10^6 Btu/hr based upon predicted tower performance curves.

The cooling tower is, therefore, adequate for all worst case meteorological conditions concurrent with a loss-of-cooling accident in one unit while the other unit is being safely shut down.

The essential service water makeup pumps may be started manually from the control room, locally at the river screen house, or automatically on level controls of the cooling tower basins. Once started automatically, they continue to operate until the 2000-gallon fuel supply to each engine drive (approximate fuel consumption is 10 gallons per hour) is exhausted or until the engines are manually stopped from the control room or locally. The engines and pumps are capable of meeting makeup requirements for the actual post-LOCA heat rejection rates under worst case evaporative loss conditions.

9.2.5.4 Tests and Inspections

Since complete redundance is provided in the system, both towers are normally operated, with one tower providing cooling for one unit and the other tower providing cooling for the other unit. The normal operating heat load of one unit (142 x 10^6 Btu/hr) or the refueling and maintenance outage heat load (13 x 10^6 Btu/hr) are more than adequate to prevent freezing of the basin and fill under winter design ambient conditions. Tower makeup may be switched from the Rock River source to the onsite wells. In this manner, continuous surveillance of all equipment availability and operability is maintained.

9.2.5.5 Instrumentation Requirements

Category I level switches are provided in each essential service water cooling tower basin. In the event of low level in a cooling tower basin, the corresponding essential service water makeup pump is automatically started. It continues operating until it is manually stopped, or exhausts the supply of diesel fuel oil in its 2000-gallon storage tank.

Local alarms and shut down equipment for the diesel engine makeup pump drives are provided for high cooling water temperature in the closed cycle cooling system, low lubricating oil pressure, engine overspeed, and engine overcrank. Annunciation is transmitted to the main control room indicating "Engine Trouble," auto-start, and auto trip for each engine.

QUESTION 010.8

"Provide piping arrangement drawings (plan and elevations) for the essential service water supply and return lines from the ultimate heat sink to the essential service water pumps. Verify that the essential service water piping has not been routed through areas such that a seismic event will not prevent the system from performing its safety function."

RESPONSE

The essential service water supply and return lines from the ultimate heat sink to the essential service water pumps has not been routed through areas such that a seismic event will not prevent the system from performing its safety function. At Byron, these lines are buried minimum 25 feet below grade level and the soil is such that through a seismic event, it will retain its supporting and restraining capability and limit the seismic movements of the buried essential service water pipe to an acceptable level.

At Braidwood, the top soil has a potential for liquefaction. Therefore, the essential service waterlines have been buried below the top soil level and rest within the undisturbed till, which will retain its supporting and restraining function through a seismic event and limit the seismic movements of buried essential service waterline to an acceptable level.

Note: This response has been superseded by the response to Question 010.21.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

'84 OCT 16 A11:17

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of)
)
COMMONWEALTH EDISON COMPANY) Docket Nos. 50-454 OL
) 50-455 OL
(Byron Station, Units 1 and 2))

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S RESPONSE TO INTERVENORS' MOTION TO REOPEN" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or as indicated by an asterisk through deposit in the Nuclear Regulatory Commission's internal mail system, this 11th day of October 1984:

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