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Re: In the Matter of Commonwealth Edison Company
(Byron Nuclear Power Station, Units 1 and 2)
Docket Nos. 50-454 and 50-455 -OL

Dear Adminstrative Judges:

Please find enclosed for your review a letter dated December 30, 1983 from Mr. Cordell Reed to Mr. R.C. De Young with the attached "Byron IDI [Integrated Design Inspection] Response."

This document is relevant to Intervenor's pending "Motion to Reopen the Record...to Include the Byron Station Design as an Issue" and to Commonwealth Edison Company's forthcoming response to that motion.

Very truly yours,

Victor Copeland

Victor G. Copeland
One of the Attorneys for
Commonwealth Edison Company

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December 30, 1983

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

Subject: Byron Generating Station Units 1 and 2
Integrated Design Inspection
Inspection Report 50-454/83-32
NRC Docket Nos. 50-454 and 50-455

References (a): September 30, 1983 letter from R. C. DeYoung to Cordell Reed

(b): October 28, 1983 letter from R. C. DeYoung to Cordell Reed

(c): October 12, 1983 letter from D. G. Eisenhut to Cordell Reed

Dear Mr. DeYoung:

References (a & b) provided the results of the integrated design inspection conducted in May and June, 1983 regarding Byron Generating Station. The report of that inspection identified a number of findings, observations, and unresolved items. This letter contains Commonwealth Edison's response to those items and addresses the general NRC conclusions presented in reference (a).

In Reference (c) the NRC also requested that we address the generic implications of the Byron IDI issues with respect to Byron's duplicate units at Braidwood Generating Station. Most of the IDI issues pertain to duplicate design aspects and the information presented here adequately addresses both Byron and Braidwood stations. There are certain issues, however, that involve field construction and design changes which need to be addressed separately for Braidwood Station. These site-specific issues which will be addressed early next year for Braidwood are Findings 2-1, 3-2, 3-8, 3-10, 3-13, 3-14, 3-15, 4-8, 4-10, 6-4, 6-6, and Unresolved Items 2-3, 3-5, 5-1, and 6-1.

At the outset, we would like to thank the members of the IDI Team for the professional and courteous manner in which they conducted the inspection. This inspection occurred during a period of intense activity on our part and the IDI Team made concerted efforts to avoid disrupting our routine work. Nonetheless, it was a thorough and painstaking inspection and required a considerable manpower commitment by

Commonwealth Edison, Sargent and Lundy, NPS, and Westinghouse. It is estimated that approximately 8000 manhours will have been devoted to the inspection itself. Naturally, in a review of such scope and depth, communication difficulties are bound to arise. In spite of such difficulties, the IDI Team made a commendable effort and accomplished a great deal in a relatively short time.

Attachment A to this letter contains Commonwealth Edison's response to the 96 specific findings, unresolved items and observations which were identified by the IDI team. In some cases, new information is presented to support our conclusions regarding specific issues.

Reference (a) also requested that Commonwealth Edison address three general areas of IDI findings and the possible need for additional design audits. These issues were reviewed with NRC management personnel on December 13, 1983. Substantial agreement has already been achieved regarding our responses on these general issues, as outlined below. To fully appreciate our responses on these general areas it is necessary to review the responses to the relevant specific findings, observations and unresolved items. General discussions are also provided in the areas of electrical separation and instrument setpoints.

Line Break and Flooding Analyses

With regard to the analyses of postulated cracks and breaks in high and moderate energy piping and internal flooding, the IDI Team was unable to conclude that the design effort was complete, adequate, and controlled. This matter has been reviewed in considerable detail and we do not concur in that overall assessment. We find that this aspect of plant design work is generally adequate and is being performed in a manner consistent with the FSAR commitments. Responses to specific deficiencies are provided in Attachment A but these relatively minor problems are not indicative of systemic weaknesses. None were found to be significant from a safety standpoint.

Our review confirms that separation of redundant and diverse engineered safeguards equipment has been a fundamental element of the Byron/Braidwood design. This approach provides inherent protection in the event of a crack or break in a high or moderate energy pipe. It also minimizes complex engineering analyses of pipe break consequences and makes the plant less sensitive to minor changes in pipe routing and potential break locations. The Standard Review Plan clearly advocates separation as the primary mode of protection and the Byron SER confirms the acceptability of this approach.

The information necessary to determine the adequacy of the plant design with respect to the effects of high and moderate energy line breaks exists in studies completed prior to the IDI. A planned report which provides a road map to this data and explains its application was not completed at the time of the IDI. This report, "Jet Impingement and Water Spray Documentation Summary" will be completed in January 1984.

For jet impingement, the Byron/Braidwood FSAR describes the analytical methods to be applied when separation is not a practical way to deal with postulated pipe breaks. The response to Finding 2-16 delineates the work done by Sargent and Lundy to assess the potential impact of jet impingement. It also delineates the extent of the documentation which was available at the time of the IDI. To more clearly show the adequacy of the Byron design, additional documentation is now being prepared. This documentation will be in the form of an additional jet impingement report which addresses the potential for jet impingement on each safe shutdown component and examines the potential results of jet impingement damage in conjunction with postulated single active failures. This report, "Verification of HELB Design Approach for Jet Impingement Effects on Safe Shutdown Equipment," is expected to be completed by the end of January 1984.

With regard to the analysis of moderate energy pipe breaks, our review has confirmed that the approach taken satisfies the FSAR commitments. The response to Finding 2-17 contains a detailed description of the approach taken to evaluate moderate energy line failure and, in particular, explains in detail why the plant is not vulnerable to water spray. Flooding, which is also the subject of Findings 2-18 and 2-19, was evaluated and documented prior to the IDI. However, the auxiliary building flooding calculation is being revised to better document the engineering judgements originally made.

Our review has indicated that a total of six findings and one unresolved item may have been pertinent to the recommendation for the review, audit and corrective action program made by the IDI Team. The findings were 2-14, 2-15, 2-15, 2-16, 2-17, 2-18, and 2-19 and the unresolved item was 2-3. On the basis of the foregoing discussion and the results of additional reviews described in the individual responses, we conclude that further audits in this area are unwarranted.

<u>Mechanical Systems Design Calculations</u>

After reviewing some of the mechanical systems design calculations, the IDI Team was unable to conclude that valid, updated analyses are generally available. After further review, we agree that mechanical systems designs are generally adequate although documentation of certain calculations should be reviewed. We found that hydraulic effects of design changes have always been adequately considered. In some cases, design changes have been made without documentation of superficial effects upon the hydraulic aspects of system design. Normal conservatisms and design margins have been shown to adequately encompass the effects of such changes.

To address the IDI concerns relative to mechanical systems calculations, all safety-related calculations (approx. 100) in the Project Management Division (PMD) calculation books (Books C-1A, C-1B and C-1C dated prior to October 1983) are being reviewed to verify that if they are technically adequate to support the current Byron/Braidwood design and to determine if the format conforms to the applicable version of GQ 3.08 in effect at the time the calculations were performed. These reviews are being done in accordance with the requirements of an approved project instruction. To date, approximately 80% of the calculations have been reviewed and no hardware changes have resulted from the reviews. All calculations are expected to be completed by January 20, 1984. No hardware changes are expected to result from the remaining reviews.

Additionally, a review of all safety-related systems is being made with respect to maximum pressure as noted in the response to Unresolved Item 2-2. No piping changes are expected to result from this review.

Our review indicated that eight findings and one unresolved item may have been pertinent to the IDI Team's recommendation for a systematic review and update of the mechanical systems design calculations. The findings were 2-5, 2-6, 2-7, 2-8, 2-9, 2-10, 2-11 and 2-20 and the unresolved item is 2-2. As discussed above and in the attached responses, specific reviews have been undertaken in response to some of the findings. Because no safety-significant deficiencies have been identified, we conclude that existing design control measures are generally adequate without further audits. Reviews presently underway are considered confirmatory and not essential to plant licensing. We have every reason to believe that no plant changes will be required as a result of these reviews.

<u>Instrument Setpoints</u>

The IDI report noted the lack of documented bases for instrument setpoints, reset values, accuracy requirements and margins. The actual design was found to be generally sound but improvements in the documentation were suggested. To address this concern, S & L will make an assessment of the safety related instruments in their scope, to identify the instruments which are complex in application and scope. A documented calculation will be provided for those instruments identified, if one does not adready exist, to verify the adequacy of the setpoints (Finding 6-3) and will include a verification of the setpoint accuracy (Finding 6-7), and the reset value (Finding 6-8), if applicable.

Electrical Separation

We agree with the IDI Team's determination that some of the engineering analyses utilized to justify exceptions to the electrical separation criteria did not individually document all of the potential means by which Non-Class 1E cables might degrade Class 1E cables. However, this does not mean that all of these potential means were not considered in the analysis. Certain Commonwealth Edison Company generic design practices are very conservative, well-known to Sargent & Lundy, and were not included in each individual analysis (Interface Review Report) because they are documented elsewhere. Examples of such conservative practices relating to fault current and voltage transients in Non-Class 1E circuits (i.e., the use of Class 1E qualified cables for Non-Class 1E applications, the use of 600 volt cable for 120 volt applications, and the use of "oversized" conductors for control applications) are described in the response to Finding 5-1.

Westinghouse balance of Plant Piping Design

The IDI Team was unable to review enough of the Westinghouse piping design work to make any conclusions. The Team raised some questions and recommended that additional review work be done to look for indications of systematic weaknesses. We have conducted additional reviews as documented in the attached responses. It is apparent from our review of the IDI Team's findings, and from the additional reviews, that there are no systemic deficiencies in the control of this work.

There are a number of isolated discrepancies between licensing commitments and detailed plant implementation. These isolated instances are all minor in significance and do not contradict the overwhelming evidence of good engineering practice employed in the conduct of this work. A review of the individual safety implications of each item has been conducted with the conclusion that the aggregate of these items, had they not been noted, would have had negligible impact on the safety of the plant.

Three categories are identified as describing the nature of the unresolved item or findings made by the IDI Team:

Obvious Conservatism - In these cases, the analytical work was performed in such a manner that certain effects of weights, loads and other parameters potentially affecting the analytical result were not explicitly addressed because they were insignificant. Evidence to support these actions has been included in our responses. Our review indicates that one unresolved item (3-4) and three findings (3-13, 3-14, 3-17) are included in this category.

In-Process Work - Several unresolved items and findings addressed issues which would automatically be rectified or corrected once the entire design process is completed. In short, the IDI Team inferred inconsistencies and errors while viewing the design process as yet uncompleted. Our review indicates that one unresolved item (3-6) and three findings (3-12, 3-15, 3-16) are included in this category.

Documentation Inconsistencies - In a few instances, the IDI Team found inconsistencies in detail between design criteria contained in licensing documents and those of the procedures being followed in the analytical design process. In these cases, however, good engineering practice was followed and design margins were maintained. Our review indicates that one unresolved item (3-5) and one finding (3-11) are included in this category.

Future actions will be taken by Commonwealth Edison to amend the Final Safety Analysis Report with design criteria that are consistent with the procedures being used in conducting the work. These cases are minor in number and do not present serious concerns with respect to degradation of design margins. No apparent evidence of systemic deficiencies or significant departure from good engineering practice was found in our reviews. Since our review has indicated that the unresolved items and findings of the IDI report do not constitute a significant safety deficiency or any serious lapse of quality control, we conclude that good engineering practice has been maintained in the balance of plant design work being performed by our contactor.

Audits of Other Areas

In the IDI inspection report, the NRC requested that we address the necessity for conducting audits of design implementation in areas other than those audited by our inspection so as to assure ourselves that deficiencies of similar importance either do not exist or are corrected. As indicated in the responses to specific IDI findings, unresolved items, and observations, in many cases we have examined other areas to assure ourselves that significant deficiencies do not exist. Since the IDI, considerable time has also been spent by our Q.A. Department in reviewing the adaquacy of the responses for IDI findings, unresolved items and observations. The lessons learned from this inspection will be incorporated into their future audit plans.

Here are some examples where our responses include examination of areas other than those audited by the IDI Team:

1. Findings 2-2, 2-6, 2-7, and 2-10 deal with flow calculations for pumps in the auxiliary feedwater and containment spray systems. We have reviewed and upgraded where necessary these calculations to verify that they are consistent with the current design. We have also reviewed and upgraded calculations for the essential service water system and the suction portion for the residual heat removal and containment spray system.

2. Finding 2-4 dealt with pipe whip calculations. We have performed a review of all calculations for other unrestrained pipe ruptures in the plant and determined that pipe impacts onto concrete walls have been addressed in the calculations.

3. In response to Finding 2-22, an S & L Project Management Division file index has been generated that indicates that files exist and are located, for the most part, in the individual engineer's offices.

4. In response to Finding 3-2, all safety-related subsystems requiring a functionality check have been identified.

5. In response to Finding 3-8, a field procedure has been revised to provide a check to assure that piping supports that have been deleted are removed.

6. In response to Finding 3-11, all 521 subsystems in the Westinghouse scope were audited by Westinghouse with respect to damping values.

7. In response to Finding 5-1, all Interface Review Reports have been reviewed and revised where necessary to address all potential means by which a Non-Class 1E cable might degrade Class 1E cables.

8. In response to Finding 6-10, the C&I logic diagrams were revised as required to include safety-related stamping.

Commonwealth Edison Company Quality Assurance Department has also conducted an in-depth audit of the 66 findings, 19 unresolved items and 11 observations identified in the above report. The audit was conducted concurrent with the accumulation and evaluation of information by CECo Engineering, S & L, and Westinghouse which was required in order to respond to the items. This approach resulted in Q.A. directly auditing the input used in the preparation of the response to the NRC IDI items. The Edison audit, which took about 400 man-hours, was directed at assuring that all the facts of the NRC IDI report were properly addressed, that there was adequate documentation to support the Edison response, that the approach to the response for each item was reasonable and appeared technically acceptable, and finally, to determine if there were any significant programmatic problems associated with the implementation of design activities.

As part of the audit process, it became clear that many of the NRC findings resulted because of changing practices over the years in documenting assumptions, calculations, approach to calculations and updating design information to reflect actual conditions. For the most part, the key problem identified by the IDI Team was the manner in which some supporting design information was documented. In the early stages of design, documentation was not always as complete as we now require it to be. Documentation inadequacies for design calculations similar to those identified by NRC in the IDI report were also found during our past audits. In the specific cases challenged in our audits, re-calculations were required to be made during the audits to demonstrate the adequacy of the design. Although the initial calculations were not fully documented, it was proven to our satisfaction by such recalculations that the designs were adequate and met the design requirements. In the upcoming Edison audits of the Sargent & Lundy corporate office, which are generally done about three times a year, continued attention to documentation of design activities will be maintained in our coverage of the total scope of design.

As to our auditing of engineering design, Commonwealth Edison has an extensive record of regular technical design analysis audits conducted of the Sargent & Lundy design in all major engineering disciplines by Quality Assurance Engineers and independent consultants with design experience. These audits are directed at a broad range of project design activities, including those areas examined in the IDI. The scope of Edison's Q.A. coverage is described in detail in the report titled "Commonwealth Edison Company Quality Assurance Statement Regarding Verification of Adequacy of Design and Construction of Byron Nuclear Power Station Unit #1," dated April 19, 1983. (See Pages 13-24; 29-34; Exhibits A, M, N, P). Specifically, Commonwealth Edison Corporate Quality Assurance Mangement has given special attention to architect/engineer and NSSS vendor design activities. Audits have covered all areas of Quality Assurance Program and procedure implementation and in many cases went into considerable depth to examine specific designs including evaluation of design bases, computer code validation, and review of calculations. Problems identified in those audits were pursued to determine root causes and to seek out generic deficiencies. These comprehensive audits have served Edison well as a basis for demonstrating that the Byron plant is properly designed in accordance with the FSAR, applicable codes, regulatory guides and standards. Edison will continue this auditing approach until completion of design activities.

Summary

The IDI was a significant effort in that it provided an independent assessment of the effectiveness of a particular set of design control measures on the Byron/Braidwood project. Eighteen inspectors went over a single plant system in great detail. In numerous cases, the IDI Team looked at design and construction activities in systems other than the auxilary feedwater system. A large number of findings, unresolved items and observations were documented in the IDI report, partially because not all of these issues could be satisfactorily resolved during the inspection period due to time constraints. The additional information provided in the attached responses responds to all of the issues.

None of the issues raised in the IDI report are significant in the context of design adequacy of the plant. They deal largely with documentation and analytical techniques and resolution requires no physical changes to the plant. We view the IDI as an independent confirmation of design adequacy and verification that licensing commitments have been fulfilled. We believe that this response contains enough information for the NRC to conclude that all IDI issues have been adequately addressed.

Please contact this office if additional discussion of our specific responses is needed.

Very truly yours,

Cordell Reed

Cordell Reed
Vice President

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Attachment A

Byron IDI Report Response

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Section

Section I

Detailed Responses to Findings

FINDING 2-1: DIESEL ENGINE AIR INTAKE

This finding states that there was no analysis or justification for the seismic vulnerability of the air intake line for the auxiliary feedwater pump diesel. The finding indicates that this conflicts with the Auxiliary Feedwater Design Criteria which states that a single active failure plus a design basis seismic event shall not prevent the auxiliary feedwater system from meeting its functional requirements.

RESPONSE

We do not agree with the conclusions of this finding. The auxiliary feedwater pump diesel combustion air intake line was intentionally routed as close as possible to the common turbine/auxiliary building seismically designed wall ("L" wall) to an area in the turbine building that was judged by the Sargent & Lundy design team to be free of non-safety-related components and equipment that would jeopardize or impair the function of the line during a seismic event. Although the air intake line is routed inside the turbine building, which is not classified as a seismic Category I structure, it was recognized by the engineers that the design of the turbine building is such that it will not fail during a seismic event (reference FSAR Subsection 3.7.2.8 and the response to Question 130.30). We do agree that at the time of the IDI there was no documented evidence that the seismic vulnerability of a carbon dioxide tank located approximately 10 feet from the air intake line was considered. Sargent & Lundy has recently verified by calculation that this carbon dioxide tank will not fail during a seismic event. Further support can be found in the Zion Probabilistic Safety Study where the fragilities for non-seismic tanks far exceeded the design basis earthquake. In addition, a documented walkdown of this area has concluded that there are no additional non-safety-related components in the vicinity that will impair the function of the air intake line. The stress report for the air intake line, reviewed by the IDI Team but not referenced in the IDI Report, demonstrates that this line was seismically supported and was seismically analyzed.

We believe that the routing of the air intake line (proximity to "L" wall), the calculation verifying seismic capability of the carbon dioxide tank (performed as a result of the IDI), the documented walkdown of the area (performed as a result of the IDI), the recognition by the S&L design team that the turbine building will not fail during a seismic event, and the existence of a seismic calculation for the line indicates that the seismic vulnerability of the air intake line was considered.

We therefore conclude that the design of the air intake for the diesel-driven auxiliary feedwater pump does not violate the Design Criteria for single active failure during a design basis seismic event and it will not prevent the auxiliary feedwater system from meeting its functional requirements.

FINDING 2-2: FLOW CALCULATION PROCEDURAL ITEMS

This finding states that Sargent & Lundy calculation AFJK-1 (auxiliary feedwater flow to the steam generators under accident conditions) had some documentation deficiencies that added to the difficulty of reconstructing the calculations which is contrary to Sargent & Lundy Procedure GQ-3.08. For example, the bases are not stated for the constant used for pipe surface roughness, the resistance coefficient constants and the static head for the faulted loop. The revision number is not stated for the drawings used to calculate friction losses. The calculation does not list the assumptions or input data that must be verified as the design proceeds in accordance with GQ-3.08. Despite the above documentation deficiencies, the IDI Team considered the calculation to be technically adequate.

RESPONSE

We do not agree with the conclusions of this finding for the following reasons. Sargent & Lundy Procedure GQ-3.08, Revision 3, in effect at the time AFJK-1 was prepared, did not require a listing of assumptions or input data that must be verified as the design proceeds. This was added in Revision 4 of Procedure GQ-3.08 dated 3-5-79. Therefore, AFJK-1 did not violate Revision 3 of GQ-3.08.

Calculation AFJK-1 was based on input data and assumptions furnished by Westinghouse to Sargent & Lundy. The revision numbers of the drawings used for the calculation were not listed; however, a review of the latest drawing revisions indicated that there have been no major piping changes that would affect the validity of the calculation. The static head used in the calculation is the elevation difference between the auxiliary feedwater pump discharge and the steam generator nozzle. Pipe friction losses were based on the well-known and widely accepted Williams & Hazens formula. The friction factor (C) chosen is an engineering judgment. An engineer often provides margin in the design by using a conservative "C" factor. For conservativeness, the calculation used the friction factor, C=100 (an input to the Williams & Hazens formula), used for water flow in "old pipes." Calculation AFJK-1 also reviewed the effect on the results for friction factors C=120 and C=140 (used for new pipe). The change in friction factors produced negligible differences in the calculated flowrates. Although the auditor was apparently not familiar with the Williams and Hazens formula, fluid dynamics literature was provided to him during the IDI to verify the wide acceptance and use of this formula. Procedure GQ-3.08 does not require documenting well-known engineering formulas. Calculation AFJK-1 is a difficult and complex calculation that we feel can be reconstructed based on the above discussion.

Sargent & Lundy approved the test results for the flow orifice pressure drops in 1981. At the time (1981), it was judged that the test results had a negligible effect on the calculated flow rates, and therefore, a revision to AFJK-1 was not necessary. Sargent & Lundy performed a calculation during the IDI to confirm to the IDI Team that the actual orifice data did not affect the flow results.

FINDING 2-3: BASIS FOR TIME DELAY

This finding states that the technical basis was deficient for approval of the addition of a time delay to the trip circuitry of the motor-driven auxiliary feedwater pump.

RESPONSE

We agree that there was no test or analysis performed to determine if a time delay was actually needed for the motor-driven auxiliary feedwater pump. The decision to add the time delay was based on a judgement by the responsible design engineer that the time delay was necessary due to the similarity between the trip circuits of the motor-driven auxiliary feedwater pump and the essential service water pumps (which were tested and found to require a time delay).

The time delay on the trip circuit for the motor-driven auxiliary feedwater pump has been removed because it has been determined that it is not necessary. Because the time delay has been removed, we have not addressed in detail the questions the IDI Team felt should be addressed relative to whether the 15 second time delay would be harmful. We, however, conclude that pump damage would not have occurred because of the amount of water in the condensate storage tank available to provide adequate suction pressure.

FINDING 2-4: ADDITION OF A TIME DELAY ON LOGIC DIAGRAM

The finding describes a field change request (FCR) which, although not specific to the auxiliary feedwater system, resulted in a schematic diagram revision to the motor-driven auxiliary feedwater pump trip circuit on starting of the pump but did not indicate the corresponding change on the pump's logic diagram. The finding also states that the responsible Project Management Division engineer had not been informed of the change.

RESPONSE

We agree with conclusion of the IDI Team that a logic diagram had not been updated. The design process early in the project requires that logic diagrams be prepared first, with electrical schematic diagram preparation following based on the logics. Later in the project life, as construction proceeds, design changes to modify the design are requested by the field on FCR's. FCR's are written referencing drawings issued for construction, or in this case, electrical schematic diagrams. Logic diagrams are not issued for construction. FCR's received by Sargent & Lundy are reviewed with the responsible engineers to determine the acceptability of the design change. If the FCR is approved, the drawings referenced therein are revised to close the FCR. Subsequently, other affected documents such as logic diagrams are revised. As such, at any point in time, logic diagrams revisions may lag the schematic diagrams revision.

Contrary to the finding, the details of the design change relative to FCR-21265 were discussed with the Project Management Division at the time the FCR was received. Also contrary to the finding, since the FCR was only written against the motor-driven essential service water pumps, design drawings for the auxiliary feedwater pump would not (and should not) have been referenced therein. A design change to the trip circuit of the motor-driven auxiliary feedwater pump controls was made for the reasons described in the response to Finding 2-3 and not explicitly due to the FCR.

Commonwealth Edison Company, with Sargent & Lundy concurrence, has decided to remove the time delays as it has been determined that they are not required.

FINDING 2-5: RECIRCULATION ORIFICE CALCULATION

The finding describes a safety-related, preliminary orifice plate calculation which was never completed although signed off.

RESPONSE

We agree with the conclusions of the IDI Team that an orifice plate calculation was incomplete although signed off, and that this is contrary to procedure GQ-4.3 (current procedure is renumbered as GQ-3.08). Upon further investigation, it was explained to the IDI Team that the calculation in question was incorrectly classified as safety-related. The orifice plate referenced in the calculation is not and was not safety-related. Subsequent to the IDI the applicable non-safety-related calculation for the recirculating orifice plate was found in the non-safety-related calculation book.

The IDI Team found the error after reviewing the Control and Instrumentation Division, Byron Project, safety-related calculation book which contained twenty calculations. No other problems were cited by the IDI Team.

As a follow-up to the IDI Team review, Sargent & Lundy has performed an additional review of the twenty calculations and confirmed that no other incomplete calculations exist.

The incomplete orifice calculation represents an isolated error and has been removed from the safety-related calculation book and nullified.

FINDING 2-6: NET POSITIVE SUCTION HEAD FOR AFW

This finding states that Sargent & Lundy Calculation AFJD-1 is deficient as a documented basis for determining that adequate NPSH is available to the auxiliary feedwater pumps and violates Procedure GQ-3.08.

RESPONSE

We do not agree with the conclusions of this finding. While we agree that Calculation AFJD-1 is not as well documented as desired by the IDI Team, we disagree that this calculation was technically deficient for determining that adequate NPSH is available to the auxiliary feedwater pumps. AFJD-1 calculated the minimum and maximum NPSH available to the auxiliary feedwater pumps based on a proposed revision to the suction piping from the condensate storage tank. The results of this calculation provided sufficient information to the Sargent & Lundy designers, relative to NPSH, to provide a design change to the suction piping. Calculation AFJD-1 established that 21 feet of NPSH is available based on an empty condensate storage tank. The Sargent & Lundy designers recognized that adequate NPSH would be available because 2 feet of water was all that was required in the 45-foot tall condensate tank to provide the required 23 feet of NPSH.

Normal makeup to the condensate storage tank is initiated at a tank water level of 26 feet which corresponds to approximately 283,000 gallons of water available above the auxiliary feedwater pumps required NPSH. Note that an additional design feature is the automatic switchover to essential service water on low suction pressure to insure that water is available for the auxiliary feedwater system should adequate NPSH from the condensate storage tank not be available.

Calculation AFJD-1 has been superceded and replaced with Calculation AFTH-01 (Reference 2.98) which was reviewed by the IDI Team. Calculation AFTH-01 is based on the current piping arrangement and includes all pipe fittings, branch tees, and pipe entrance losses. Based on the current piping arrangement, it was determined that calculation AFJD-1 did not neglect the friction loss from six branching tees. The documentation deficiencies noted for AFJD-1 did not affect the validity of the calculation and did not result in a design deficiency.

FINDING 2-7: NET POSITIVE SUCTION HEAD FOR CONTAINMENT SPRAY

Several alleged deficiencies, given below, were noted in Calculation CS-5 which determined the available NPSH for the containment spray pumps and residual heat removal pumps:

1. a. Assumptions are not listed.

   b. Input data and/or assumptions that must be verified as the design proceeds are not identifed.

2. The assumed piping and valve arrangement does not represent the current piping configuration.

3. The calculation does not account for a partially blocked screen per Regulatory Guide 1.82.

RESPONSE

We do not agree with the conclusions of this finding for the following reasons. Although the input for Calculation CS-5 did differ in some respects from the current configuration of the containment spray system, the calculation is adequate to determine that the system will operate properly. The system changes, subsequent to the calculation, all served to increase the NPSH margin. Since no potential existed for a system design deficiency, a revision to the calculation serves only to quantify the NPSH margin, not to determine adequacy. The alleged deficiencies can be addressed individually:

1. a. We agree that the assumptions are not listed separately. However, the input to the calculation is clearly shown and is not ambiguous. The calculation was reviewed and approved in accordance with Sargent & Lundy Quality Assurance requirements. In addition, Calculation CS-2, dated 8-25-83, has been performed to verify that adequate NPSH has been provided for the RHR and CS pump 1B. This calculation (CS-2) has a listing of design information and pipe flow parameters.

   b. This requirement was implemented in Revision 4 (3-5-79) to Sargent & Lundy QA Procedure GQ-3.08 and therefore, was clearly not applicable to Calculation CS-5 (7-31-75). In addition, the responsible engineers judge whether or not a system requires a review to determine if reverification of the adequacy of the design is required. This judgment is made based on the type of change (whether the change increases or decreases the margin), and the severity of the change (whether a large decrease in the margin will occur).

2. Revision to the design of the containment spray and RHR pump suction piping systems since 1975 have served to increase the available NPSH. Only two potentially significant differences exist. The suction line from the containment sump has been changed and the centerline of the pump suction connection has been changed.

   Calculation CS-5 included 20-inch sump suction piping and associated fittings and valves. The current design utilizes a 24-inch suction line. A revised calculation (Calculation CS-2) shows that the flow losses in the piping are in fact reduced, as a result of these changes, by over 1.5 feet in the containment spray system and in the RHR system. The final pump design and installation resulted in a pump suction centerline approximately 2 feet lower than utilized in Calculation CS-5. As a result, the NPSH is about 3.5 feet greater than calculated in CS-5.

3. At the time Calculation CS-5 was made (July 1975), the screen arrangement in the plant did not conform to Regulatory Guide 1.82 (June 1974). An additional screen was added in 1982. Regulatory Guide 1.82 states "The effect of partially blocked screen should be considered in the evaluation of the overall NPSH." However, the preceeding sentence in Regulatory Guide 1.82 states "For the recommended design velocity at the fine inner screens considered in this guide, a negligible pressure drop is anticipated across the screens." It is not clear why omission of a negligible quantity from a calculation constitutes a deficiency. Nonetheless, revised Calculation CS-2 does include losses for a partially blocked screen. The screen loss has increased from 0.017 feet in Calculation CS-5 to 0.031 feet in Calculation CS-2 (additional pressure drop of 0.014 feet [0.0061 psi]).

In summary, we do not agree that Calculation CS-5 is deficient as a basis for determining that adequate NPSH is available to the containment spray and RHR pumps. CS-5 did not totally reflect the existing system but it did conservatively represent it. Calculation CS-2 has been completed to quantify the NPSH margin available.

FINDING 2-8: MISSING CALCULATION FOR CONTAINMENT SPRAY

FSAR Section 6.5.2 contains a detailed discussion of the NPSH available for the B train containment spray pump. No calculation was available to support the January 1979 revision of this section. The lack of an available calculation was contrary to Procedure GQ-3.08.

RESPONSE

We do not agree with the conclusions of this finding. The "missing" calculation in question is not a design calculation because the purpose of GQ-3.08 is to "describe the quality assurance requirements for the preparation, review and approval of calculations that support the design of safety-related structures, systems, and components." The information in FSAR Section 6.5.2 did not provide the basis for, or in any way support the design of the containment spray system. As explained in response to Finding 2-7, Calculation CS-5 provided an adequate basis for the system design. During the IDI, Calculation CS-2 was completed which quantifies the NPSH margin based on as-built data. This calculation shows that the final design provides more margin than indicated by the FSAR description. The FSAR is being revised to eliminate outdated information.

The FSAR description is of a preliminary system layout which was slightly modified to provide additional NPSH. Inclusion of this description or calculation in the FSAR is not required by Standard Review Plan Section 6.5.2 and was not required to meet any licensing commitment. The information was not the basis for any safety-related design and was not part of a licensing commitment.

FINDING 2-9: FSAR DESCRIPTION OF CONTAINMENT SPRAY

The discussion of the containment spray system in FSAR Section 6.5.2 is not consistant with the current design. Although the specific items did not constitute deficiencies of technical significance, the FSAR description and the design should be consistent.

RESPONSE

We agree with the conclusion of the IDI Team that the FSAR description and the design should be consistent. We also agree that the noted discrepancies are not of technical significance. As explained in the response to Finding 2-8, the FSAR description was not used as design input and was not required to support licensing commitments. FSAR Section 6.5.2 will be revised to eliminate the outdated information. The FSAR will be updated to reflect the change.

FINDING 2-10: CONTAINMENT SUMP SCREEN VELOCITY

A calculation to support the design of the containment sump screens could not be found during the inspection. Calculation SI-03 was performed during the inspection and indicated a worst case velocity of 0.24 fps in contrast to the recommended velocity of 0.20 (Regulatory Guide 1.82).

RESPONSE

We agree with the IDI Team that the calculation could not be found during the inspection. The missing calculation (CS-012 dated 4-6-83) was unavailable during the inspection because it was sent to the Sargent & Lundy Quality Assurance Division for microfilming as required by procedure. Calculation CS-012 was a design calculation performed to determine the required screen area. Examination of the calculation revealed the reason for the difference in flow velocities (0.24 fps in contrast to 0.20 fps). Calculation SI-03, which was completed very quickly during the IDI to demonstrate to the IDI Team the adequacy of the design was overly conservative in that the area occupied by stiffeners and braces was not considered to be flow area. Calculation CS-012 included the area of these small but numerous supports under the assumption that the approach velocity, a short distance from the screen, is more indicative of the likelihood of debris to settle out than the velocity in the screen itself. Calculation SI-03 has been revised such that the assumptions and bases are consistent with Calculation CS-012. The velocity with a 50% screen blockage is actually calculated to be 0.16 fps. This demonstrates the design to be adequate.

FINDING 2-11: MAXIMUM PIPING PRESSURE

This finding stated that Sargent & Lundy had not performed a calculation to determine the maximum pressure of the auxiliary feedwater system to assure that the piping is acceptable. The finding states this is contrary to ASME Code, Section III, Subsection HD-3612.4 which states that pump discharge piping shall be designed for the maximum pressure exerted by the pump.

RESPONSE

We do not agree with the conclusions of this finding. The values for auxiliary feed pump discharge maximum pressure are as follows:

	Design Pressure (psig)	Maximum Operating Pressure (psig)
Design Criteria	--	2080
Mechanical Department Piping Line List	1750	2080
S&L Wall Thickness Calculation	1750	--
Design Specification DS-AF-01BB*	1750	--

*Test Loads - Section 404 of DS-AF-01BB indicates "The auxiliary feedwater pumps will be tested at 100 gpm and 2080 psig."

The ASME Section III Code (1974 Code, Summer 1975 addenda, is the applicable code for Byron) defines the design pressure as follows:

ND-3112.1 Design Pressure

"Components shall be designed for at least the most severe condition of coincident pressure and temperature expected in normal operation."

The normal operation of the auxiliary feedwater system is when water is being supplied to the steam generators. Under this operating condition, the most severe condition of pressure is 1750 psig.

The test load represents the maximum operating pressure (2080 psig) at which the auxiliary feedwater system would be normally operated, other than the design condition.

In April of 1982 the Byron project decided to consolidate and reverify the minimum wall requirements for the pipe. Confirmatory calculations were performed at that time based on design pressures. The calculations indicate the maximum required wall thickness (t_m) per the Code for the design pressure and the nominal wall thickness actually provide for the piping. For the auxiliary feedwater system piping, these calculations indicate:

	Code Calculated Minimum Wall (t_m) Required for a Design Pressure of 1750 psig	Wall Thickness Provided	
		Nominal Wall	Minimum Wall (t_m)
4" Sch 120 pipe	0.251 inches	0.438 inches	0.383 inch
6" Sch 120 pipe	0.369 inches	0.562 inches	0.492 inch

If these wall thicknesses are converted to maximum allowable design pressure (this is readily done for nominal wall using Sargent & Lundy Standard MES 2.5 [IDI Ref. 2.64]), the information would be as follows:

	MINIMUM WALL-4" PIPE	
	0.251 inch Code Calculated Min. Wall Required (t_m)	0.383 inch Wall Thickness Provided (t_m)
Maximum Design Pressure @15,000 psi allowable stress	1750 psig	2742 psig

	MINIMUM WALL-6" PIPE	
	0.369 inch Code Calculated Min. Wall Required (t_m)	0.492 inch Wall Thickness Thickness Provided (t_m)
Maximum Design Pressure @15,000 psi allowable stress	1750 psig	2367 psig

We do not agree that non-conservative values were used for wall thickness calculations. We believe that design conditions were identified as required by the applicable Code and that substantial margin was provided for the piping.

The IDI Team indicated that they "would consider it more appropriate to use, concurrently, a condition where suction is taken from the essential service water system, which would result in a pressure of 2165 psig." This condition would be extremely unusual and if it did occur, would occur less than 1% of the time. Nonetheless, it is only 85 psig greater than what has been identified as the test load (2080 psig). It can be seen by inspection that the 2080 psig test load and the 2165 psig postulated by the IDI Team are well within the capability of the piping system. This was indicated by the IDI Team in the IDI Report as follows:

> "However, the installed piping is adequate since it is rated for 2367 psig based on the allowable working pressure listed in Sargent & Lundy (S&L) Standard MES-2.5 for 6-inch schedule 120 pipe (Reference 2.64)."

It should be noted that S&L engineers have access to and utilize the above-mentioned S&L standard. This S&L Standard is a document where code maximum allowable working pressures are delineated for various pipe sizes, schedules, and materials.

The IDI Team indicated that the calculations performed were contrary (relative to the IDI Team's postulated "more appropriate pressure") to the ASME Code, Section III, Subsection ND-3612.3, which states (as indicated in the IDI Report) that the pump piping shall be designed for the maximum pressure exerted by the pump. The complete code statement is:

> "Pump discharge piping shall be designed for the maximum pressure exerted by the pump at any load and for the highest corresponding temperature."

We conclude that we have satisfied this requirement. The pump discharge piping provided (nominal wall pipe) has a code design maximum pressure capability in excess of the postulated pressure. The Code does not define this postulated pressure as the "design pressure" (as indicated earlier, the design pressure definition in accordance with ND-3112.1 is 1750 psig), nor does it require that this pressure be used in determining the minimum wall.

In addition, the following Code section states:

> "<u>ND-3612.3 Allowance for Variations From Design Conditions</u>
>
> (a) It is recognized that variations in pressure and temperature inevitably occur and therefore the piping system shall be considered safe for occasional operation

for shot periods at higher than the design pressure or temperature as limited in (b).

(b) Either pressure or temperature, or both, may exceed the design values if the stress in the pipe wall calculated by the formulas using the maximum expected pressure during the variation does not exceed the S-value allowable for the maximum expected temperature during the variation by more than the following allowances for the periods of duration indicated:

(1) Up to 15% increase above the S-value during 10% of the operating period;

(2) Up to 20% increase above the S-value during 1% of the opening period."

<u>Note</u>: There is a typographical error in the Code, 1% of the opening should be 1% of the operating period.

The postulated 2165 psig that the IDI Team considers more appropriate would fall under ND-3612.3 which allows a 20% increase above the S-value (allowable stress) if the duration is expected 1% of the operating period. We have demonstrated that this condition can be met with no code permissible increase in allowable stress. <u>This is clearly conservative</u>.

Information was forwarded to IDI Team after the IDI by letter dated July 5, 1983 transmitting additional information for calculating minimum wall thickness. This information included a table provided by the Piping Fabricator. We indicated that it was a practice (non-required, non-safety-related function) of the Fabricator at the time (1975) to do a minimum wall calculation. The Fabricator used the larger of the maximum operating pressure or design pressure in his calculation. At the time, the maximum of the two pressures was 1830 psig. The Piping Fabricator did include, as part of his tabulation, the maximum calculated pressure for the piping. His tabulation indicates that the piping is capable of 2369 psig for 6-inch Sch 120 pipe. The purpose of sending the information to IDI Team was to indicate that the piping is checked in several non-documented ways (in addition to documented ways) and to show that it is not uncommon to establish maximum calculated pressures for piping. The maximum calculated pressure indicates for the designer, and in this case the Piping Fabricator, the margin that exists in the design that can be used as a comparison for unusual or postulated operating conditions. The fact that the Piping Fabricator had calculated the maximum pressure was not mentioned in the IDI Report.

Sargent & Lundy Standard MES-2.5 is a tabulation of maximum calculated pressure for nominal wall pipe. This tabulation for the piping under discussion was also forwarded to the IDI Team by the above-mentioned letter and was referenced in the Report.

The tabulation showed, as noted in the IDI Report, a design pressure for the 6-inch Sch 120 pipe of 2367 psig.

The concern of the IDI Team was probably caused because the minimum wall calculation was based on the design pressure (1750 psig) which is lower than the maximum operating pressure (2080 psig).

Although we conclude that this is consistant with the Code and that adequate margin exists, we have reviewed the entire Pipe Line list to verify that the piping is capable of withstanding the greater of the design and operating pressures. There are only 12 safety-related lines where the operating pressure is identified to be in excess of the design pressure as noted below:

Quantity	Line Size (in)	System	Design Pressure (psig)	Max. Operating Pressure (psig)	Pipe Schedule	Pressure Capability (psig)
4	4	FW	1750	1855	160	3600
4	6	FW	1750	1855	120	2367
4	1½	SI	2485	2735	160	6199

In addition, as indicated in response to Unresolved Item 2-2, the maximum pressure will be identified for all safety related piping and the ability to satisfy code pressure considerations will be addressed and documented for the maximum pressures.

FINDING 2-12: DESIGN REVIEW CLOSEOUT

This finding indicated that Sargent & Lundy procedures and standards do not adequately describe the method for completing system design reviews. Procedure GQ-3.10 states that the department standards shall describe the method for verifying and documenting the resolution of open items and discrepancies. Mechanical Department Standard MAS-4 does not describe the method for verifying and documenting that correction of discrepancies has been completed.

RESPONSE

We agree with the conclusion of the IDI Team that a revision to MAS-4 would clarify the required method for verifying and documenting that correction of discrepancies has been completed. Mechanical Department Standard MAS-4 states that the Mechanical Project Engineer shall maintain a file of all design review records for the project as well as a list of all unresolved items. The Mechanical Project Engineer assigns the unresolved items to a Responsible Engineer and documents when each item is satisfactorily resolved. The System Design Review Status Report is prepared and issued by the Mechanical Project Engineer. A system review designated as "completed" documents that all open items have been resolved. A commitment to update a design drawing or document was considered by the Mechanical Project Engineer to be a satisfactory response and a resolved item. It was not required to maintain the system review open until the drawing or document was actually released.

Sargent & Lundy is currently reviewing the Byron/Braidwood system design reviews and the Responsible Engineers are advising the Project Manager that all open items have been, in fact, resolved and that any commitments to update drawings or documents have been completed. In addition, Sargent & Lundy Mechanical Department Standard MAS-4 is being revised to indicate that a system design review shall not be designated as completed on the status report unless all required documents are revised or a follow-up close out system is provided.

FINDING 2-13: DESIGN CRITERIA UPDATING

This finding indicated discrepancies between the auxiliary feedwater system Design Criteria DC-AF-01-BB and the actual installation. These discrepancies were summarized as follows:

1. Design criteria states that the minimum auxiliary feedwater flow rate supplied to one or more unfaulted steam generators is 470 gpm. Sargent & Lundy calculation AFJK-1 calculated 459 gpm which was used in the Westinghouse accident analysis and documented in Chapter 15 of the FSAR.

2. The design criteria references a pressure drop across the flow restricting orifices of 155 psi based on a flow of 160 gpm. Calculation AFJK-1 used a pressure drop of 230 psi at 160 gpm based on data points from the Daniel flow orifice calculator. Data from the orifice supplier indicated the pressure drop to be 197.7 psi for 160 gpm.

3. The design criteria states that the developed head of the auxiliary feedwater pump under minimum flow conditions is 4800 feet while the actual pump performance curve indicates a developed head of 4700 feet under minimum flow conditions.

This finding also requested that the general practice of not updating design criteria be addressed.

RESPONSE

We do not agree with the conclusions of this finding. The Byron/Braidwood design criteria were prepared and issued as the basis and starting point for the initial system designs. Design criteria are not intended to be up-to-date system descriptions, nor are they intended to reflect the final design. The design drawings and documents themselves do this function.

The following responses address the individual IDI Team's concerns regarding the auxiliary feedwater system design criteria:

1. The auxiliary feedwater design criteria was based on the recommended Westinghouse flow rate of 470 gpm to the intact steam generators. Although AFJK-1 calculated 459 gpm (which was accepted by Westinghouse), the design criteria remains as 470 gpm. The deviation from the design criteria was documented in the FSAR (prior to the IDI) and no change to the design criteria is necessary.

2. The pressure drops listed in the design criteria for the flow orifices and pipe friction were estimated values based on the preliminary piping layout to demonstrate the derivation of the auxiliary feedwater pump discharge head requirements. Calculation AFJK-1, based on the actual piping configuration, determined there was less pipe friction pressure loss and therefore more pressure drop across the orifice. The adequacy of the auxiliary feedwater pump was verified based on the actual differential pressure of the system. Purchase of the flow orifices was based on a specified beta ratio, not on the differential pressure listed in the design criteria. As stated in the response to Finding 2-2, the actual flow orifice test data had no impact on the calculated flow rates.

3. The pump developed head of 4800 feet under minimum flow conditions (100 gpm) specified in the auxiliary feedwater design criteria was based on preliminary information received from the pump manufacturer. This number is not a design point but was included in the design criteria for information to assure that the piping is properly designed. The actual test performance curves provided by the pump manufacturer indicate a developed head of 4700 feet at minimum flow. The pump head/flow condition has been found through existing calculations to satisfy system design requirements.

We recognize that some design criteria have been updated to reflect modifications made to the design while others contain either outdated or obsolete information. To eliminate potential confusion, the Design Criteria Status Report will be revised to include a status for each design criteria. The classification of the design criteria will address the IDI Team's concern about future use of these documents or attempted use by someone not familiar with the actual status of a particular design criteria. In general, modifications to a system at this time in the project are based on the latest design documents not on design criteria.

FINDING 2-14: PIPE WHIP CALCULATIONS

The feedwater piping inside the main steam tunnel has 27 postulated breaks. The design calculations address 22 of these breaks. Three of the remaining five breaks did not require any calculations. For the last two breaks, C22 and C29, there were no calculations and it did not appear obvious that the impacted concrete tunnel walls would be capable of withstanding the pipe impact forces.

RESPONSE

To the original design engineer, it was obvious that the 5 foot 6 inch thick concrete walls being impacted by pipe breaks C22 and C29 identified in EMD File 6535 (Reference 2.34) are structurally adequate based on previous calculations done for more critical combinations of break force and wall thickness. For this reason, no design calculations were performed. During the inspection, calculations were performed to confirm this conclusion and submitted to the NRC IDI Team (Reference 2.154). These calculations proved the structural adequacy of the impacted walls and substantiated the original design engineer's engineering judgment.

A review of all calculations for other unrestrained pipe ruptures in the plant has shown that all pipe impacts onto concrete walls have been addressed in the calculations. Therefore we consider this finding to be resolved.

FINDING 2-15: ENERGY DISSIPATION ASSUMPTION

The IDI Team reviewed the basis for Sargent & Lundy's 50% assumption, a report by Chelapati and Kennedy on "Probabilistic Assessment of Aircraft Hazard for Nuclear Power Plants" (Reference 2.90). The report indicates that, for an aircraft striking a nuclear structure, the energy is diminished by the process of deforming the aircraft. This is conceptually similar to the situation for a whipping pipe, but we found no direct connection or specific basis for the 50% assumption that had been used in the calculations. Procedure GQ 3.08 states that the calculation should list the data and assumptions which establish the basis of the design. In this case the assumption was listed, but an adequate technical basis for the assumption was not established by the listed reference.

RESPONSE

The technical basis for the assumption that 50% of the kinetic energy of a whipping pipe is absorbed by plastically deforming the pipe is a report by Chilapaki and Kennedy on "Probabilistic Assessment of Aircraft Hazard for Nuclear Power Plants" (Reference 2.90). Figure 9 in this report indicates that for plastic impact, the maximum absorption factor for the wall is 0.5. This report was used because the missile effects of the aircraft and the pipe are similar in as much as both of them are relatively soft when compared to the rigid concrete structures they are impacting. Therefore, the pipe will plastically deform as the aircraft does and the original design assumption is valid.

However, in order to further substantiate the adequacy of the design, subsequent calculations have been made using an alternative approach and have been submitted to the NRC IDI Team (Reference 2.151). These calculations employ the conservation of momentum principle to show that the wall can adequately absorb the impact energy without considering the deformation of the pipe. Since both techniques confirm design adequacy, we consider this finding to be resolved.

FINDING 2-16: JET IMPINGEMENT ANALYSIS

The Team found that, although FSAR Section 3.6.2.2.2.1.4 indicates, for postulated breaks, how the jet impingement force will be calculated, no jet impingement analysis had not been performed. The Team found that, although Sargent & Lundy personnel indicated they had intended to do the work, no documented program or procedure was in place to provide assurance that the work would be completed

RESPONSE

We agree that the documentation of the jet impingement analysis was not complete and available. However we do not agree that a jet impingement analysis had not been done for the following reasons. FSAR section 3.6.2.2.2.1.4 describes the methodology for calculating jet forces if jet impingement shields are found to be necessary. It does not contain the procedure for assessing the potential for jet impingement damage. This is most effectively done by assessing the vulnerability of safe shutdown systems rather than calculating the properties of the jets. Although no specific analysis of jet effects exists, various studies have been completed which demonstrate that jet impingement effects will not prevent safe shutdown. The "work" that had not been completed was not an additional analysis but an accumulation of information (i.e., a summary "road map" report) that recreates the engineering considerations and assessments that were made and which demonstrates the applicability of the existing studies to the question of jet impingement. This summary report, "Jet Impingement and Water Spray Documentation Summary," will be completed in January 1984.

These existing studies and documentation can be categorized in three areas: those which identify and locate high energy line breaks, those which demonstrate the separation and protection of safe shutdown systems, and those which describe the Byron approach to protection against jet impingement. The following is a list of the more significant studies and documentation (asterisk indicates information made available to the IDI Team):

1. Location of High Energy Lines and High Energy Line Breaks

    a. High Energy Line Location

        1) FSAR - 1978*
        2) Final Review - 1981 (S&L Calculation)*

    b. High Energy Line Location (Based on Stress)

        1) 1980 - 1982 (S&L/W Calculations)*

c. HELB Located on Composites

1) 1982 (Informal)*

2. Separation of Safe Shutdown Systems

a. Color Coded Mechanical Systems Composite - 1975 (Informal)

b. Fire Protection Report - 1977 (Docketed)*

c. Safe Shutdown Report - 1981 (Docketed)*

d. Color Coded Mechanical/Electrical Composites - 1981 (Informal)*

e. Environmental Qualification Report - 1982 (Docketed)*

3. Jet Impingement Approach

a. FSAR Sections 3.6.1.1.2, 3.6.2.3.2.7, 3.6.2.3.1.2*

b. NRC Question Response (10.40) - 1982*

c. NRC Byron SER Supplement No. 2 - 1983*

In order to more clearly show the adequacy of the Byron plant, additional documentation is being prepared which will clearly show the potential for and effects of jet impingement on safe shutdown equipment and components.

This additional report will be structured as an independent document rather than relying heavily on other documents such as the Fire Protection Report. The input to this additional jet impingement report will be the list of Safe Shutdown Equipment, the identified high energy lines, high energy line break locations, and the plant design drawings showing the system configuration and plant arrangement. Each physical area in the plant which contains safe shutdown equipment will be identified. Areas with no high energy lines in the proximity of the equipment will be eliminated. In the remaining areas, an assumption will be made that all equip-ment is disabled by a jet, and, in addition, a limiting single active failure occurs. The capability of the plant to achieve safe shutdown will then be evaluated. In the event that a safe shutdown path cannot be found, a more detailed analysis of the area will be performed, including a review and evaluation of alternate shutdown paths. The final conclusion is expected to be identical to the summary "road map" report; namely, that jet impingement will not jeopardize the ability to safely shut down the plant. This report is scheduled to be complete by January 31, 1984.

The remainder of this response describes the approach which has been used to address jet impingement and the documentation which was available at the time of the inspection.

FSAR Subsection 3.6.2.2.2.1.4 is a description of a mathematical method for calculating the force of a fluid jet from a postulated pipe rupture. This section contains no commitment to perform this calculation for all high energy pipe ruptures. The relevant FSAR commitment is in FSAR Section 3.6.1.1.2:

> "The criteria used for protection against pipe whip and the Commission's letter from Mr. Giambuso, dated December 1972, have been met for designs inside and outside the containment respectively. By virtue of the Construction Permit date for this plant, the above is the required minimum.
>
> "Subsequent criteria, including that in the Commission's letter from Mr. O'Leary, dated July 1973, and Branch Technical positions APCSB 3-1 and MEC 3-1, have been employed to the extent possible and practical, given the stage of design/construction. Essentially all of the above criteria have been met, with the exception of maximizing optimization of plant layout to provide remote location of potential sources of pipe failure with respect to equipment essential to protect against such failures. In these few cases, physical separation or whip restraint/impingement barriers have been employed."

Clearly, no jet force calculation is required if protection has been provided by the preferred means of remote location of the high energy lines and/or the safe shutdown equipment or by physical separation. In the event a barrier is required to protect from jet impingement, the design approach is described in FSAR Subsection 3.6.2.3.2.7 for piping other than reactor coolant system (RCS) piping:

> "Jet impingement shields are provided as required to protect safety-related equipment and components.
>
> "To account for these effects in the design, a combination of component restraints, barriers, and layout is utilized to ensure that for a loss-of-coolant, steam or feedwater line break, propagation of damage from the original event is limited, and the components as needed, are protected and available. Design loads are obtained from Subsection 3.6.2.2.2.1.4. Allowable stresses are the same as in Subsection 3.6.2.3.2.4."

No jet deflectors were required on non-RCS piping and therefore, the equations in FSAR Subsection 3.6.2.2.2.1.4 were not used. The approach to jet impingement analysis for RCS breaks is explained in FSAR Subsection 3.6.2.3.3.3 and is consistent with the commitment in FSAR Subsection 3.6.1.1.2.

Jet deflectors were required at four locations on the reactor coolant piping. Design of these deflectors is described in FSAR Subsection 3.6.2.3.1.2.

A summary of the jet impingement design approach was provided in the response to NRC Question 10.40 (FSAR page Q10.40-9):

> "The approach to jet impingement is described in FSAR Subsection 3.6.2. The break locations defined for the pipe whip investigation were also examined for jet impingement effects. The majority of locations had no effect on equipment required for safe shutdown. This was a result of the criteria used in design to maintain separation of redundant systems and the use of compartments to isolate high energy line effects. Equipment which could be affected by jet impingement was analyzed and moved or protected if protection was required."

The Byron design approach of using separation of redundant safe shutdown systems and diversity of safe shutdown paths is consistent with the NRC guidance in the Standard Review Plan (SRP), NUREG-75/087. Branch Technical Position APCSB 3-1 (included in Section 3.6.1 of the SRP) states:

> "Although various measures for the protection of safety-related systems and components are outlined in this position, the preferred method of protection is based upon separation and isolation by plant arrangement."

The Byron approach was reviewed by the NRC staff in preparation of the Safety Evaluation Report (SER), NUREG-0876, and found to be adequate. Supplement No. 2 to the Byron SER (January 1983) addresses this issue in Section 3.6.1 which states, in part:

> "The plant design accommodated the effects of postulated pipe breaks and cracks, including pipe whip, jet impingement, and environmental effects. The means used to protect essential (safety-related) systems and components include physical separation, enclosure within suitably designed structures, pipe whip restraints, and equipment shields. To be consistent with BTP ASB 3-1, the applicant used separation as the primary means of protection, where separation was not feasible, one of the other acceptable methods of protection was used."

The physical separation of safe shutdown systems has been verified by a variety of efforts throughout the design of the plant. In late 1975, the safe shutdown systems piping and equipment were located and color coded by division on Containment and Auxiliary Building Piping plan and section drawings. The Fire Protection Report (FPR) was submitted in November of 1977. The FPR analyzes in detail the ability to safely shut down the plant following an initiating event which affects various unrelated systems in an area. The FPR contains color coded drawings showing safe shutdown equipment, components and cables throughout the plant. The FPR was recognized to be applicable to the jet impingement issue with two qualifications. The FPR did not rigorously consider single failure and did not consider LOCA or Main Steam Line Breaks as initiating events. Ability to withstand single failure is inherent because the FPR did not take credit for diversity in shutdown paths. Each safe shutdown system is backed up by another safe shutdown system or combination of systems that can perform an equivalent function. Since these backup systems are also checked for separation of redundant trains in the FPR, safe shutdown capability is assured. LOCA and MSLB inside containment can be mitigated with two systems not included in the safe shutdown report, Safety Injection and Containment Spray. Both of these systems have well separated redundant trains and do not contain active components inside containment. As a result, jet impingement will not impair safe shutdown capability.

Location of safe shutdown equipment remote from high energy lines and the enclosure of high energy lines in protective enclosures (subcompartments and tunnels) also contributes to the safety of the plant. The initial submittal of the FSAR (November 1978) contained piping schematic diagrams (Figures 3.6-1 through 3.6-12), which located all high energy lines. The corresponding P&ID's (Piping Instrumentation Drawings) have high energy lines marked with asterisks to alert designers to the need for special consideration of high Energy Line Break (HELB) effects.

At a later stage in the project, it was felt that design was finalized to the extent that high energy line locations should be final. As a result, Sargent & Lundy calculation 3C8-1181-001 (December 1981) was completed. This calculation located all auxiliary building high energy lines and was used to update FSAR Figures 3.6-1 through 3.6-12. The purpose of this calculation was to ensure that all HELB effects had been considered in the appropriate areas of the auxiliary building. At approximately the same time, a set of piping composite drawings was marked to show postulated HELB locations in the containment and auxiliary building.

Also, in late 1981 and early 1982, sets of mechanical and electrical composite drawings were marked up to show the location and routing of safe shutdown components, piping, and cables. This was done to provide an even more detailed identification of safe shutdown systems and to show the separation of the redundant trains.

The IDI Team was shown all the above-mentioned FSAR sections, the SER, drawings showing the high energy line break locations and safe shutdown components, the FPR, and Sargent & Lundy Calculation 3C8-1181-001. The only additional effort planned in this area was a summary report which documented the applicability of these studies and surveys to the jet impingement issue. The location of breaks was done not for the purpose of defining jet properties but to determine the general areas in which jets could occur. The Byron approach was to identify equipment required to safely shutdown the plant and to verify that redundant trains of this equipment were not susceptible to common damage by a single high energy line break rather than examining the effects of all postulated high energy line breaks. This approach eliminates most uncertainty associated with location and direction of jets and other HELB effects.

As noted, the safe shutdown capability of the plant following damage to safe shutdown components is being documented in detail. These reports will be prepared in accordance with Sargent & Lundy QA requirements and will be retained in the calculation files.

FINDING 2-17: MODERATE ENERGY PIPE CRACK ANALYSES

FSAR Section 3.6.2.1.2.2 indicates that through-wall pipe leakage crack locations are postulated based on stress levels to maximize effects from fluid spraying and flooding. The Team determined that this had not been accomplished and it did not appear that the work was programmed to be done. The design cannot be considered adequate in this regard until this work has been done to locate those instances where leakage might damage essential equipment and to protect equipment as appropriate in accordance with the licensing commitment.

RESPONSE

We do not agree with the conclusions of this finding because they misinterpret the FSAR commitment and disregard the methodology actually used to evaluate the effects of moderate energy pipe cracks. This methodology (briefly discussed below), which is very conservative and exceeds the FSAR commitments, was explained in detail to the IDI Team but is not mentioned in Section 2.4.4 (Moderate Energy Pipe Cracks) of the IDI Report.

FSAR Subsection 3.6.2.1.2.2 contains a commitment to evaluate moderate energy line cracks in accordance with Standard Review Plan guidelines:

> "Through-wall leakage cracks are postulated in Seismic Category I moderate-energy ASME Section III, Class 2 and 3 and seismically qualified ANSI B31.1 piping located both inside and outside containment except where the maximum stress range is less than $0.4\ (1.2\ S_h + S_a)$. In unanalyzed moderate-energy ASME Section III Class 2 and 3 and ANSI B31.1 piping, this exception based on stress is not taken. The cracks are postulated individually at locations that result in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions developed."

Use of the stress criteria in the FSAR will result in cracks being postulated at only a small portion of the potential crack locations. In practice, however, the need for protection must be evaluated early in the design process, before the final stress values are available. As a result, a more conservative approach of designing the plant to accommodate a crack at any moderate energy location was adopted. This effectively bounds the FSAR commitment.

To evaluate possible flooding effects, moderate energy line cracks were postulated at locations which would result in the most severe flooding regardless of stress level. The limiting locations for postulated cracks are documented in Sargent & Lundy Calculation 3C8-1281-001, "Auxiliary Building Flood Level Calculations (IDI Report Reference 2.15) and in Sargent & Lundy Calculation RAS-FL-1, "Flood Level Inside Containment." Calculation 3C8-1281-001 utilized a calculational procedure to determine the crack or break flow rate based on the pipe size, wall thickness and internal pressure. Each area in the auxiliary building was surveyed and the highest break flow rate determined. The location chosen and the flow rate calculation for each area is documented in an appendix to Calculation 3C8-1281-001. Section 2.4.5, Flooding Analysis of the IDI Report states "Sargent & Lundy's Nuclear Safeguards and Licensing Division performed an analysis of flooding in the auxiliary building from postulated high-energy and moderate energy line failures for the lines that would produce the worst flooding conditions in each area." This statement and Finding 2-17 are contradictory.

Evaluation of water spray from moderate energy line cracks is not required in areas where effects are bounded by effects of high energy line breaks (SRP Section 3.6.2). This would eliminate essentially all of the containment. However, FSAR Subsection 3.6.2.1.2.2 does commit to evaluate effects of water spray inside containment. The capability to withstand water spray is established in the Byron Environmental Qualification of Electrical Equipment report (June 1982). All electrical equipment inside containment required for safe shutdown is qualified for spray.

Water spray in the auxiliary building will not damage equipment such that safe shutdown capability is jeopardized. This is a result of the basic design which separates redundant and diverse safety systems. Water spray is postulated to affect only electrical equipment and not cables or mechanical equipment. Water spray is also assumed to not simultaneously affect two systems separated by 20 feet or more. The Byron Fire Protection Report (FPR) documents this separation with color-coded drawings showing safe shutdown components as well as tables and descriptions of the locations of these components. The FPR does not address systems required following a design basis event (LOCA and Main Steam Line Break) but does cover all systems necessary for safe shutdown following damage or failure in a system in the auxiliary building.

The FPR does not rigorously postulate single failure but it also does not take credit for diversity of safe shutdown paths. By examining the requirements for safe shutdown and the systems utilized to achieve safe shutdown, it can easily be seen that water spray resulting from failure of a moderate energy line will not result in the inability to safely shut down the plant.

Safe shutdown requires three functions - (1) prevention of release of excessive offsite radioactivity; (2) negative reactivity; and (3) removal of decay heat. Goal (1) is met for a moderate energy line crack outside of containment because the initiating accident does not breach the primary system and containment isolation is not required. Goal (2) requires only that the reactor protection system function for hot shutdown. This will not be affected by water spray in the auxiliary building. Cold shutdown may require boron addition, depending upon plant conditions. As discussed in the FPR, damage to the boron transfer system can be repaired within 72 hours. The charging system is used to inject boron. In the event both centrifugal charging pumps are lost through a combination of water spray and single failure, the plant could be maintained in a hot shutdown condition until the damage is repaired or the non-safety-related positive displacement charging pump is made available.

Goal (3), removal of decay heat, is of most interest following a moderate energy line crack. FSAR Table 3.6-3 lists six systems required for safe shutdown in this case. These six systems, used to remove decay heat, are:

a. Residual Heat Removal System (RH);

b. Chemical and Volume Control System (CV);

c. Auxiliary Feedwater System (AF);

d. Component Cooling System (CC);

e. Essential Service Water System (SX); and

f. Essenti‑l Service Water Makeup System (Byron Only) (SXM)

The function and redundancy of these systems can best be seen by examining a diagram of the various paths for safe shutdown decay heat removal as shown in Figure 1. This diagram shows four paths. Two of these reach cold shutdown within 72 hours while the other two (without RHR) may taken longer than 72 hours. These paths assume offsite power has been lost.

Section B.3.6.3 of Branch Technical Position ASB 3-1 states that if the initiating failure is in a normally operating moderate energy system, postulation of a single active failure is not required in the redundant train of that system. The RH, CC, SX, and SXM systems fall into this category.

The SX and SXM systems are required for all shutdown paths. The redundant SX trains are separated and contain relatively few active components. A single failure analysis of the SX system is included in the FSAR in Table 9.2-2. The SX pumps are the only vulnerable components of interest. In their location in the auxiliary building basement, the pumps could be affected only by a crack in the same SX train and therefore no single failure is postulated in the redundant train. Even in the event both trains of the SX system for one unit were out of service, the SX valves could be aligned to supply the CC and RH systems from the other unit.

The only source of spray in the vicinity of the CC pumps is the CC piping. Due to the arrangement of the pumps, spray from a single source would not affect more than one other pump in addition to the failed train. The plant can be safely shut down with any two of the five pumps operating. Since single failure is not required and no other components are vulnerable to spray, safe shutdown can be achieved.

As shown in Figure 1, the RH system is required only to reach cold shutdown in a short time. The only RH component outside containment which could be affected by spray is the RHR pump motor. This motor is located in a room adjacent to the containment which contains only one train of the RH system. As a result, any crack which causes failure of one RHR train will not require postulation of a single failure in the redudnant train.

The essential portion of the CV system consists only of the charging path into the containment. The valves in this path are normally open and fail in that position. The pumps are in cubicles separated from the redundant train and other systems. In the event one centrifugal charging pump fails and the other is disabled by a single active failure, the CV system is backed up by the Safety Injection System. The SI system becomes available when the primary system pressure is reduced to 1700 psig and is not subject to common failure with the CV system. Secondary cooling (per Figure 1) is used to reduce the primary pressure.

The AF system has no components which could be damaged by spray, except for the pump drives. These are in rooms which contain only one AF train. Therefore, water spray will affect only one train of the AF system. The reliability of the AF system has been documented in the response to NRC Question 10.53 and accepted by the NRC. In the event spray disables one AF 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.

Capability to safely shut down the plant following a moderate energy line crack and the resulting spray in conjunction with a single failure as required by Branch Technical Position

ASB 3-1 will be documented in a report titled "Jet Impingement and Water Spray Documentation Summary".

Section IV of the FSAR response to NRC Question 10.40 (Pages Q10.40-1 through Q10.40-5, attached) summarizes the approach to moderate energy line break analysis. This approach was reviewed by the NRC and discussed on page 3-2 (Section 3.6.1) of the Byron Safety Evaluation Report, NUREG-0876, Supplement No. 2, January 1983:

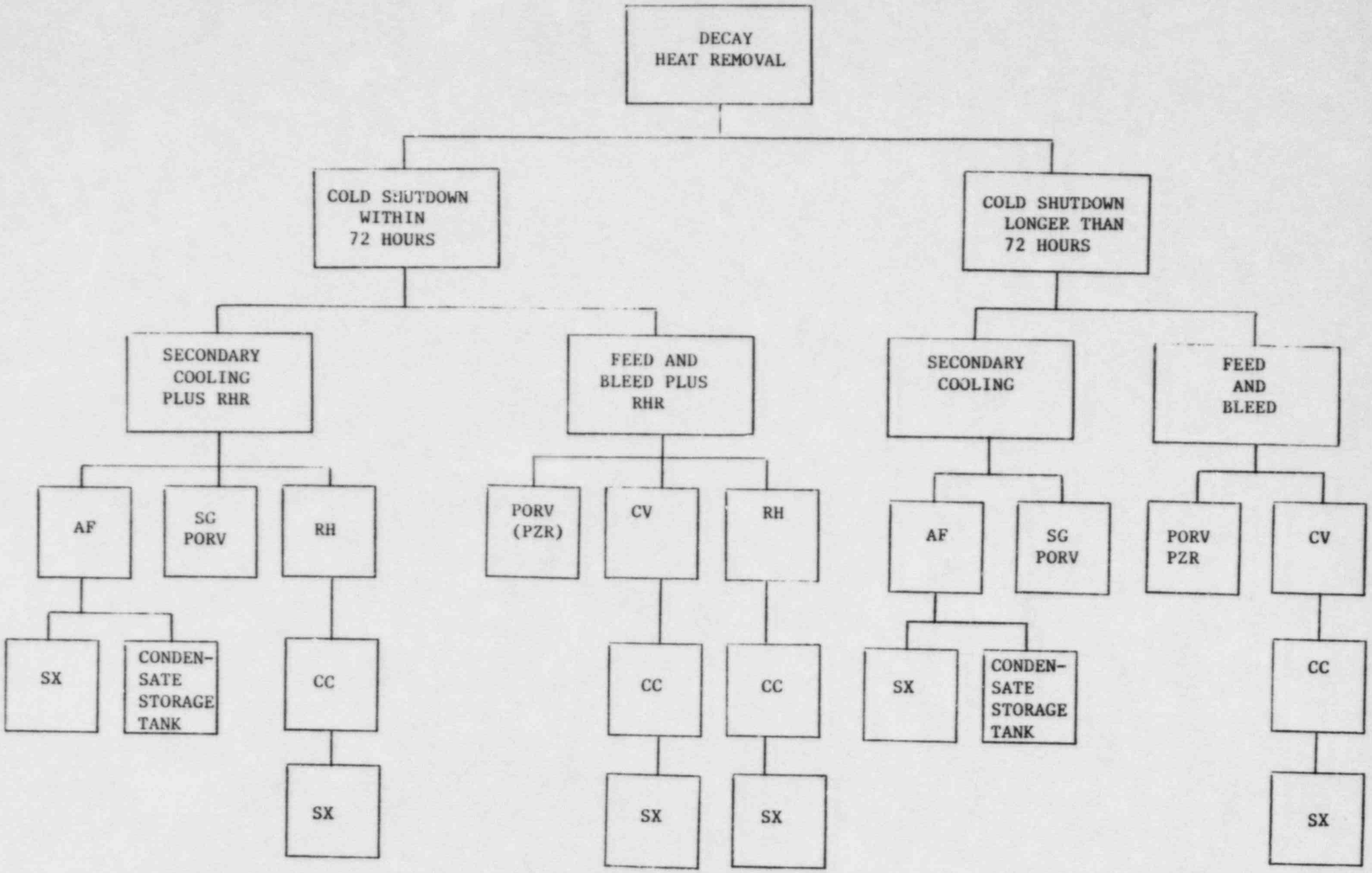
> "The applicant has also analyzed the effects of moderate energy line breaks outside containment on safety-related systems. The moderate energy systems are also designed to meet the criteria set forth in BTP ASB 3-1. The staff has determined that the applicant has adequately demonstrated that a postulated crack in a moderate energy line will not cause loss of function of any safety-related system because of environmental effects such as flooding, high temperatures, or water spray."

The response to FSAR Question 10.40 provides a detailed example of the effects of a moderate energy line crack in an essential service water line in the auxiliary building basement demonstrating that shutdown of the plant is unaffected. In response to a specific request from the Auxiliary Systems Branch, this question response also includes a detailed examination of failures in the component cooling system. Again, the ability to safely shut down the plant is maintained. During the IDI, the IDI Team requested that the area around the Auxiliary Feedwater (AF) regulating valve station be evaluated. This was done and it was found that even though only two locations in the piping examined (the two largest subystems in the AF system) exceeded the criteria for postulation of through wall cracks, a crack could occur at any location on the subsystems without affecting safe shutdown capability. This analysis was transmitted to the IDI Team (IDI Report Reference 2.153) but was not mentioned in the IDI Report.

This finding is based on a misinterpretation of the licensing commitments in FSAR Subsection 3.6.2.1.2.2 and does not accurately reflect the actual methodology used to meet regulatory requirements and the information supplied to the IDI Team.

RESPONE TO . NDING 2-17

FIGURE 1



QUESTION 010.40

"Provide a response to Question 010.17 and include the following in your response. Provide the results of analyses of the effects on safety-related systems of failures in any high or moderate energy piping system in accordance with the J. F. O'Leary letter of July 12, 1973, as defined in Branch Technical Position ASB 3-1, Appendix C. Provide a table which identifies the method of protection provided all safety-related systems listed in FSAR Table 3.6.1 from failures of any high or moderate energy systems listed in FSAR Table 3.6-2. Include figures depicting the locations of failures relative to the systems of FSAR Table 3.6-1 giving dimensions, locations and protective method for each postulated break or crack in a high or moderate energy system. Include the assumptions used in your analysis such as flowrates through postulated cracks, pump room areas, sump capacities, and floor drainage system capacities."

RESPONSE

I. INTRODUCTION

To ensure safe and reliable operation of the Byron and Braidwood Nuclear Power Stations, the possibility of high or moderate energy line breaks have been considered in the design. This response documents a confirmatory study of the potential high and moderate energy line breaks which demonstrates that all design features necessary to mitigate the results of line breaks have been incorporated.

Standard Review Plans (SRP) 3.6.1 and 3.6.2 were used as the basis for this study. SRP 3.6.1 includes Branch Technical Position (BTP) APCSB 3-1. Appendix B of the BTP, the attachment to letters sent to applicants and licensees by A. Giambusso in December 1972, and Appendix C to the BTP, the July 12, 1973 letter to applicants, reactor vendors and architect-engineers from J. F. O'Leary, provide the basis for identification of high energy line breaks and evaluation of their consequences

Piping drawings which identify the high energy lines are included in the FSAR (Figures 3.6-1 through 3.6-12). Breaks have been postulated at the locations required by Branch Technical Position APCSB 3-1 for the purpose of assessing pipe whip and jet impingement effects. Pressure and temperatures in areas were calculated assuming the break occurs in the limiting location in the area. Locations of mitigating features such as pipe restraints and impingement shields are shown in Section 3.6 of the FSAR. Drawings showing the location of high energy lines have been provided to the NRC ASB reviewer. These drawings also indicate location of subcompartment walls and pipe tunnels.

II. SCOPE

The effects of high and moderate energy line breaks inside containment have been assessed as described in FSAR Sections 3.6 and 6.2. Additionally, an investigation into the effects of high and moderate energy line breaks outside containment has been made and is described in this response. Non-safety related areas, such as the turbine building, were not investigated because damage to or failure of equipment in these areas will not affect plant safety.

The possible effects considered are structural loads due to pressurization, increases in pressure and temperature which could affect environmental qualification of equipment, and damage due to pipe whip and jet impingement. Flooding is a potential effect but is not addressed in this response. The response to Question 010.47 demonstrates that high and moderate energy line breaks will not cause flooding which would adversely affect the plant safety.

Because of variations in requirements, techniques, and failure effects, high and moderate energy lines are addressed separately. Similarly, the pipe whip, subcompartment pressurization, and environmental analysis all have somewhat different approaches. The following sections are divided to reflect these distinctions.

III. HIGH ENERGY LINE ANALYSIS

Standard Review Plans 3.6.1 and 3.6.2 were followed in defining and identifying high energy lines. High energy lines are those larger than 1 inch diameter for which either:

a. The service temperature is greater than 200 F; or

b. The design pressure is greater than 275 psig.

Only a limited number of systems in the auxiliary building meet either of these criteria. The following systems have been identified as containing high energy lines in the auxiliary building:

Chemical and Volume Control	(CV)
Auxiliary Steam	(AS)
Steam Generator Blowdown	(SD)
Radioactive Waste Processing	(WX)
Boric Acid	(AB)
Main Steam	(MS)
Feedwater	(FW)
Auxiliary Feedwater	(AF)
Residual Heat Removal	(RH)
Safety Injection	(SI)

Systems which are normally not used or at reduced temperature and pressure are not necessarily required to be considered as high energy lines. A guideline has been established (Branch Technical Position MEB 3-1) that if the system is at high energy conditions less then 2% of the time, it may be considered a moderate energy line and its normal conditions applied to the line break analysis. On this basis, the last three systems (AF, RH, SI) are not considered as high energy systems. The Byron/Braidwood AF system is not used for normal startup as at some other plants. The only high energy line in the boric acid system is a steam supply line to the boric acid batching tank. This line is essentially a part of the auxiliary steam system and, as a result, was not identified in FSAR Table 3.6-2.

Subcompartment pressurization is investigated for all lines with temperatures above 200° F. Lower temperature lines do not have the potential for flashing to steam and thus will not increase the pressure of a subcompartment in the event of a break. Pressurization is of concern only in small subcompartments with relatively large high energy lines or subcompartments with limited pressure relief venting.

High energy lines below 200° F have only minor effects on the environmental conditions. The absence of steam and the ability to drain warm liquid from the break area limits the temperature rise from these breaks. The auxiliary building HVAC has sufficient capacity to accommodate these lower temperature breaks. Breaks of other high energy lines may influence the expected maximum temperature in some areas of the auxiliary building even if high pressures do not result. The auxiliary building contains several large areas with high energy lines that are not subject to pressurization but are investigated for environmental effects.

Certain postulated break locations in high energy piping systems are used to investigate the potential for damage due to pipe whip and jet impingement. The guidelines in Standard Review Plan 3.6.2 are used to determine the number and locations of the pipe breaks. Pipe restraints are added as required to prevent damage to structures and safety-related equipment.

IV. MODERATE ENERGY LINE BREAKS

Moderate energy lines are lines which operate at temperatures below 200° F and pressures below 275 psig. A break in a moderate energy line will not result in flashing of the liquid to steam and, as a result, has no potential for pressurization of areas. The relatively low temperature and reduced heat transfer effects of the liquid blowdown precludes significant temperature increases in the area of the break. The reduced break area applicable to these breaks and the absence of steam allows the auxiliary building HVAC to maintain temperatures within those specified in the environmental qualification program. The results of moderate energy line breaks are, therefore, confined to the physical effects of liquid discharge into the plant. Plant safety is affected only if equipment required to mitigate the break or to safely shut down the plant can be damaged by resultant flooding or water spray. Flooding is discussed in the response to Question 010.47. Water spray was not found to affect plant safety because of the separation of redundant safe shutdown systems and components. Moderate energy line breaks do not result in pipe whip.

As an example, the auxiliary building basement at elevation 330 feet is subject to severe flooding following a moderate energy line failure in the essential service water system. The limiting failure in this area is a crack in a 36-inch essential service water line. The predicted leak rate is 2.8 ft^3/sec. The basement is divided into two completely independent sections. These sections are separated by a wall which has been designed to withstand the flooding. Each section contains redundant essential service water pumps which can supply both units. Therefore, flooding or spray from a break cannot affect the equipment in the other section of the basement and essential service water will be supplied to both units.

This separation is well documented in the Fire Protection Report. This report lists and locates equipment required for safe shutdown. When redundant safe shutdown systems are separated by fire walls or by more than 20 feet, spray from a crack in a moderate energy line would not impair the safe shutdown capability of the plant.

A moderate energy line break in the component cooling system was given special consideration because the component cooling system is not supplied with a Category I source of makeup water. A leak in this system could theoretically drain the surge tanks and result in damage to the component cooling pumps.

A significant leakage in the component cooling system is not expected. The system is a moderate energy, low pressure system and is not subject to severe loading. In the event the system is inoperable, the plant may be safely maintained in a hot shutdown condition until the component cooling system is restored.

If a crack is postulated in one of the large lines in the system, the level in the surge tank of the affected unit will drop. When the level reaches the low setpoint level, alarms will sound and the affected units component cooling pumps will be automatically tripped to prevent damage to the pumps.

If reactor water or demineralized water makeup is available, the component cooling pumps may be restarted and the unit operated normally while the leak is located and isolated. Otherwise, the reactor will be tripped because of the interruption of component cooling to the reactor coolant pumps and the unit will be placed in a hot shutdown condition. Component cooling is not required to safely maintain the unit in hot shutdown mode. The component cooling system can be operated after a failure of the piping by closing the appropriate system valves to isolate the break location and maintain component cooling flow.

V. SUBCOMPARTMENT PRESSURIZATION

For the purpose of protecting subcompartments from overpressurization, the CV, AS, SD, WX, MS, and FW systems were traced through the auxiliary building and all subcompartments containing high energy lines were identified. The most severe break in the subcompartment was analyzed.

The main steam (MS) and feedwater (FW) systems are routed entirely in an enclosed tunnel in the auxiliary building. The limiting break in this tunnel is a main steam line rupture. Section C3.6 fully describes an analysis of a break in this tunnel.

The remainder of the auxiliary building was surveyed level by level to identify all subcompartments which could be pressurized by high energy line breaks. Figures Q10.40-1 through Q10.40-5 identify all areas containing high energy lines. The identification of the limiting line in each zone is also included. The zone numbers do not correspond to environmental qualification zones (Section 3.11).

Figure Q10.40-1 represents elevation 346 feet 0 inch. Zone 1, the recycle waste evaporator room, has been analyzed and the results are reported in Section A3.6 of the FSAR. Zones 2 and 3, letdown reheat heat exchanger rooms and valve areas, have been analyzed and the results are reported in Section A3.6. The assessment in A3.6 addressed Zone 3, the more limiting zone.

Figure Q10.40-2 represents elevation 364 feet 0 inch. Assessment of Zones 5A and 5B, the positive displacement charging pump areas, predict a peak pressure of 2.42 psid and a peak temperature of 190° F. These results are being added to Section A3.6 of the FSAR. Zones 6A, 6B, 7A, and 7B, the centrifugal charging pump rooms, contain high pressure, low temperature lines. Failure of these lines (normal temperature of 115° F) will not cause pressurization or increase temperatures. Pipe whip and impingement are considered. Zones 9A and 9B contain portions of the steam generator blowdown system. Control valves upstream of these lines limit the blowdown flow and prevent the postulated breaks from impacting plant design. Zones 8A and 8B, blowdown condenser rooms, have been analyzed and the results are included in FSAR Section A3.6. Zones 11 and 12, blowdown condenser rooms, have been analyzed and the results are reported in Section A3.6.

FINDING 2-18: FLOODING ANALYSIS

The IDI Team noted that the Project Management Division made an undocumented engineering judgment that high predicted flood levels in the plant would cause doors to open prior to floor damage. The Team felt that this was inconsistent with the FSAR response to Question 10.47 and that other factors such as (1) the lock might be stronger than the minimum rating or (2) the door might open inward and the door frame might be stronger than the lock, should be considered.

RESPONSE

We do not agree with the conclusions of this finding. The primary purpose of the flooding calculation was to evaluate the impact of flooding on safety-related equipment. Structural concerns were addressed at a much earlier stage in design. The response to FSAR Question 10.47 states "Doors are considered to be open or closed to maximize the flood levels." FSAR Section D3.6.2.1 includes the explanation "Doors were assumed to remain closed to calculate the maximum flood in the area containing the break and assumed to open to check on potential flooding of adjacent areas." This is a calculational technique employed to ensure that all affected equipment is identified and is not inconsistent with the engineering judgment made about the relative door and floor capabilities. These assumptions allow the analyst to identify all flooding effects on equipment without determining the specific structural capabilities. However, for completeness the auxiliary building flooding calculation is being revised to address the adequacy of both structural and mechanical/electrical aspects of the design for those areas with high flood levels. This will serve to document the engineering judgment used.

All areas which could be flooded to levels which would be of structural concern are in the lower elevations of the auxiliary building. The floors, walls and supporting structures are generally adequate for loads much greater than the 200 lb/ft^2 live load considered in design. The finding discusses the live load rating as if it were the structural capability when, in fact, it is a design minimum. This value (200 lb/ft^2) also includes a design load factor of 1.7, which is not required under accident conditions such as flooding.

The finding mentions two examples of factors which should be considered. The first was the possibility that the lock might be stronger than the minimum rating. For the areas in question, the real concern is whether the weakest link in the rooms is the door latch, door frame or the hinge attachments, or if it is the structural floor and walls. In the

Byron auxiliary building, the areas with predicted high levels have been analyzed and it has been verified that even if the doors do not open, no structural damage will result.

The second concern about doors opening inward is not valid because the only areas which could have high flood levels are subcompartments and all subcompartment doors open outward.

The six areas noted in the report were reexamined to confirm our assessment of the structural impact as noted below:

A. Two of the areas, the RHR pump cubicles, are located on Elevation 343, 3 feet below adjacent areas. The floor slab and lower 3 feet of the walls (with the exception of one block wall) are mats backed by compacted soil and easily capable of withstanding high flood levels. The reinforced concrete walls above Elevation 346 would easily handle the high flood levels and, in any event, be stronger than the block wall and door. Failure of the block wall would not have an adverse safety effect.

B. The other four areas are the RHR heat exchanger cubicles. These areas, although nominally on the 364 feet 0 inch level, are actually recessed and have a floor at the 357 feet 0 inch level. These cubicles have no outflow path for water other than floor drains below 364 feet 0 inch. In effect, they are designed to be flooded and to prevent RHR leakage from traveling to other parts of the auxiliary building. As a result, the walls and floor of these cubicles are 40 inches thick.

Based on this information, we believe the judgment on the structural implications of the flooding analysis were justified given the location of the high flood levels and the design of the Byron auxiliary building. The reexamination of the structural design demonstrates there is no safety significance of this finding. The revision of the calculation will document the engineering judgments made. This revision will be completed by January 31, 1984.

FINDING 2-19: FLOOD LEVEL

Project Management Division personnel sent a memorandum to the Electrical Department requesting that equipment be located to be protected from a predicted 13 inch flood in the Unit 2 auxiliary feedwater diesel drive pump room. This level was not shown on the General Arrangement drawings. The Team interviewed cognizant Sargent & Lundy personnel and alleged that no record or personnel recollection of the specific PMD memorandum or the resultant activity existed. The Team was also concerned because they did not find a systematic tracking system to assure that actions needed have been taken.

RESPONSE

We agree that the design drawings were not updated in a timely manner. However, we believe the finding is inaccurate in the allegation that cognizant personnel were unaware of the design change. The design drawings have now been updated to show the flood levels which could affect safe shutdown capability. The IDI concern about the need for a systematic tracking system is addressed by the normal design procedure of updating design documents to reflect changes. This procedure, in conjunction with project communications (memos, documented project meetings, etc.) insure that required reviews are made.

This finding is based on an 8-4-82 memorandum transmitting the revised flood level information to the Electrical Department The finding is inaccurate in that the memorandum was in the Electrical Project Engineering files and shown to the IDI Team and the cognizant Electrical Project Engineer was aware of the flood requirements. A copy of the memorandum was also available in the files of the Electrical Design and Drafting Division at the site. These files were not reviewed by the IDI Team. The memorandum was written as a result of a discussion between the mechanical and electrical design personnel involved with this area. It was determined that transmittal of the information by memo prior to inclusion on design drawings would insure an early review of the affected area. This review was, in fact, made as a result of the memo but not documented at that time. The review has since been documented and the design drawings have been updated. Although the drawings should have been revised more quickly, we believe initial transmittal of the data by memo was effective in initiating a design review in this instance where the design was almost complete.

The flood levels are contained in Sargent & Lundy Calculation 3C8-1281-001, "Auxiliary Building Flood Level Calculations" (December 1981) and Calculation RAS-FL-1, "Flood Level Inside

Containment" (Revision 1, July 1982). These are approved design calculations in accordance with Sargent & Lundy procedure GQ3.08. At the time the information in question (13 inch flood level) was developed, the design of the Unit 2 auxiliary feedwater diesel driven pump room was complete, and construction on this room was in progress. As a result, the most effective way to insure that the postulated flooding could be accommodated, was to make a review of the affected design.

Appropriate design checks were made prior to the IDI to insure that electrical equipment would not be damaged by the predicted flood levels. No design changes were required. Revised flood levels in the containment and the auxiliary feedwater diesel-driven pump rooms have been added to piping composite drawings.

FINDING 2-20: DIESEL ENGINE HEAT LOAD

This finding states that Calculation VA-100 used a non-conservative number for calculating the total heat released to the ambient from the auxiliary feedwater pump diesel due to a revised insulation package and exhaust piping.

RESPONSE

We do not agree with the conclusions of this finding. Calculation VA-100 was performed prior to the award of the insulation contract for the auxiliary feedwater pump diesel exhaust manifold and associated piping. The diesel heat load value used in Calculation VA-100 was supplied by the engine manufacturer based on his proposed insulation package. Although the actual insulation installed is different from that proposed by the diesel engine manufacturer, the difference in heat loads is not significant in comparison with the design margin of the room cooling system. Similarly, the short length of piping did not add significantly to the heat load. These facts were obvious to the designers and no calculational refinement was deemed necessary. However, to confirm this point, Calculation VA-100 has been revised to reflect a 2.6% increase in the heat load based on the actual installed insulation on the exhaust manifold and piping. Based on the revised calculation, there is still an additional 11.4% margin in the capability of the HVAC system for this area. The calculated room temperature is 108° F while the design temperature for this area is 122° F.

Due to the adequate margin in the design of the HVAC system, there were no design errors resulting from the revision to the exhuast insulation. We have demonstrated that this finding is dealing with minor effects, which the designer correctly judged to be insignificant.

FINDING 2-21: EMERGENCY SERVICE WATER TEMPERATURE

The Westinghouse feedline break analysis assumed the auxiliary feedwater temperature to be a maximum of 95° F. This temperature is inconsistent with FSAR Section 9.2.5.3 which indicates that essential service water temperature may be as high as 98° F, Design Criteria DC-AF-01-BB which indicates a water temperature of 100° F, and specification F/L-2758C for the auxiliary feedwater pumps which indicates a temperature range of 40-100° F.

RESPONSE

We agree with the IDI Team's assessment that these differences are not significant. The minor temperature discrepancies listed in this finding have no impact on the Westinghouse accident analysis or the auxiliary feedwater system design. The 100° F temperature listed in the design criteria is based on ambient conditions that the outdoor condensate storage tank will experience. This temperature is used as a reference for the purchase of equipment such as the auxiliary feedwater pumps and valves. The design outlet water temperature of the essential service water cooling towers is 98° F based on maximum heat load and worst case environmental conditions.

The difference in water density between 95° F and 100° F is less than 0.1%. Since this has no impact on the accident analysis, no document changes are necessary or required.

FINDING 2-22: PROJECT MANAGEMENT DIVISION FILING SYSTEM

This finding states that the Project Management Division File Index was incomplete and did not meet the requirements of Sargent & Lundy Procedure GQ-6.01.

RESPONSE

We agree with the conclusions of this finding that there should have been an updated file index. The majority of the project information in the Project Management Division (PMD) is maintained by the Responsible Engineers in the division. At the time of the audit, information on hand with the Responsible Engineers was not identified by the File Index as required by GQ-6.01. The Project Management File Index has been updated and supplements the file index used for the main file ("Records Center"). If a responsible engineer leaves or is transferred, his files (those identified by the PMD File Index) will remain part of the project files. We do not consider this to be a problem, systematic or otherwise, since the information was being retained even though the File Index was not being employed. We believe the corrective action taken resolves this finding.

FINDING 3-1: DIGITIZED RESPONSE SPECTRUM

This finding addresses a discrepancy between the OBE Response Spectrum 101-OB-EW and the Response Spectra Design Criteria.

RESPONSE

We agree with the conclusions of the finding. We also agree with the IDI Team's conclusion that this did not indicate any systematic weakness and did not cause any design deficiency. The Digitized Response Spectra (101-OB-EW), which is one of the 18 Response Spectra used in the analysis, had a differential acceleration value of 0.02g (0.26g instead of 0.28g) at 18Hz.

Furthermore, the envelope of the 3 E-W Response Spectra, including 101-OB-EW used in the analysis, resulted in an acceleration value of 0.8g at 18 Hz. This one discrepancy represents 0.3% of all digitized points comprising the 18 Response Spectra identified in the analysis.

FINDING 3-2: FUNCTIONALITY CRITERIA

In summary, the memorandum defining systems to be checked for functionality was incomplete. Some of the analysis packages that had been defined as needing the check did not state that the criteria had been met and, for some of the packages that did state the criteria was met, it was not clear that valid analyses had been performed to make this determination. These items indicated systematic weaknesses in the Program for Meeting Functionality Criteria so that the licensing commitment was not being met.

RESPONSE

We agree that the memorandum for defining the sytems to be checked for functionality had not been updated and that some of the analysis packages that had been defined as needing the check (inclusion of a Class 1 stress calculation or have documentation that the piping met the Class 2 and 3 functionality criteria) did not state that the criteria had been met. We also agree that this item is not expected to cause any design deficiencies because the stresses are generally low. Sargent & Lundy has updated all the system design specifications outlining Functional Capability Requirements for ASME Section III Piping in the Sargent & Lundy scope. This provides a documented basis for systems requiring a functionality check.

S&L has also revised EMD TP-2 to clarify qualification methods available to satisfy functional capability requirements (EMD TP-2, Rev. 4, EMD-046032). The review of fifteen (15) auxiliary feedwater systems for functional capability resulted in the following:

a. Four (4) system stress packages did not contain a statement concerning functional capability (AF-08, AF-09, AF-10, and AF-15). Two of these stress packages, AF-09 and AF-10, did contain a Class 1 Analysis Qualification (Run ID 650 UGS and 327 UGS, respectively). Stress Package AF-08 required the Stress Scanning Method for qualification, and Stress Package AF-15 required a calculation which was not included in the stress package at the time of the audit.

b. The remaining eleven(11) system stress packages did have a statement concerning functional capability. Five (5) of these packages used the Class 1 Analysis Method for Qualification (AF-01, AF-06, AF-07, AF-14, and AF-16). Four (4) of these packages used the Stress Scanning Method for qualification (AF-03, AF-04, AF-11, and AF-12). The remaining two (2) stress packages (AF-02 and AF-05) required calculations which were not included in the stress packages at the time of the audit.

Based on the above discussion, all the auxiliary feedwater systems satisfied the functionality requirements. There were three (3) stress packages (AF-15, AF-02, and AF-05) that required a calculation method to satisfy the functional capability criterion per NEDO-21985. These calculations were provided to the IDI Team (EMD-044004, June 10, 1983).

The IDI Team also questioned the applicable code version used for NEDO-21985 and the PIPSYS program. In particular, they indicated that there had been "many stress intensification factor changes" for the Class 2 and 3 piping analysis requirements versus the functional capability requirements. The NEDO report references ASME Section III 1977 issue through summer of 1978 addenda. The PIPSYS computer program has been updated for the 1977 issue through the winter of 1978 addenda for Class 2 and 3 piping. Only minor differences occurred between these issues of the code, none of which impacted the stress intensification factors used for analysis. As a result of this, it is possible for an analyst to screen the Class 2 and 3 stress results and compare them visually against the functional capability criteria since the design rules are the same in both cases.

It should be noted that the NEDO report also references the 1977 issue through the summer of 1978 addenda for Class 1 analysis. In this case, PIPSYS has been updated to the winter of 1979 addenda. It should be noted that in this case, the PIPSYS version being used is either equal to or more conservative than the criteria recommended in the NEDO report. It should also be noted that the Section III code revisions for Class 1 analysis were based on the same "seed" report by E. C. Rodabaugh and S. E. Moore (Reference 1 of NEDO-21985) which was prepared for the NRC to evaluate this very subject. Consequently, Sargent & Lundy believes that the practice being followed is either equivalent to or more conservative than the NEDO report criteria.

The IDI Report states "it is difficult to assure that the General Electric Report Class 1 criteria are met when only ASME Code Class 2 stresses are calculated". This is not Sargent & Lundy's practice, nor has it ever been. The IDI team member's observation is invalid since the functional capability requirement for ASME Section III Class 2 & 3 piping, which Sargent & Lundy follows, is delineated in Section 2.2.e of NEDO-21985 (IDI Reference 3.93) which is referred to as the General Electric Report by the IDI Team, and states as follows:

> "Piping constructed in accordance with the code rules for Class 2 or 3 may be evaluated for functional capability using the criteria given herein for Class 1 piping."

All the safety-related subsystems in the S&L scope requiring a functionality check have been identified and will have a documented functionality check performed.

FINDING 3-3: NUMERICAL INTEGRATION ALGORITHM

The numerical integration scheme outlined in FSAR Subsection 3.7.3.1.1.5 is not identical to the algorithm actually employed in the PIPSYS Computer Code.

RESPONSE

We do not agree with the conclusions of this finding. As stated in the IDI Report, the Numerical Integration Method actually employed is technically adequate and the differences between the two methods are insignificant.

All piping analyses for Byron performed by Sargent & Lundy utilized the PIPSYS Computer Code which is referenced in Appendix D of the FSAR.

The Numerical Integration Method outlined in FSAR Subsection 3.7.3.1.1.5 is noted as "one available method for the numerical integration," and as such, does not constitute a commitment to use this specific method.

FINDING 3-4: SUBSYSTEM NUMBER

This item indicates that, on Drawing M-37 (Reference 1.57), the reference to the subsystem boundary showing the line from the Condensate Storage tank to the suction sides of the auxiliary feedwater pump should be labeled 1CD06 rather than 1AF06. This finding states that it is an isolated drafting error.

RESPONSE

We agree with the conclusions of this finding. However, the subsystem identification number on piping and instrumentation diagram is for reference only as stated on Drawing M-34, Sheet 2. It is not a design input for piping analysis. However, on Drawing M-37, Revision AA, the drafting error has been corrected to 1CD06.

FINDING 3-5: FLANGE WEIGHTS

This is a discrepancy of flange weights in Stress Report 1AF03.

RESPONSE

We agree with the IDI Team that a discrepancy of 19 lbs between the weight shown on the analytical drawing (62 lbs) of the 6-inch flanges and the weights used (43 lbs) in the piping analysis will not constitute any design deficiency and does not require any further consideration. This has been documented in an addenda to the stress report.

FINDING 3-6: PIPE SUPPORT ADDED MASS

The input data for two out of six hangers were not properly prepared to account for the additional weight or mass.

RESPONSE

We do not agree with the IDI Team's conclusion that this is a systematic weakness for the following two reasons:

1. The addition of supports, weight and mass into the piping analysis is governed by the criteria outlined in Paragraphs 6.4.5, Table 6.4.5-1, and Figure 6.4.5-1 of Engineering Mechanics Division's "Lesson Plan for Training Personnel in Piping Analysis," EMD TP-1, Revision 3 (Reference 3.50).

2. According to the criteria, the weight of the support assembly 1AF14003R (45 lbs) does not have to be considered in the analysis since it does not exceed the weight of 1 foot of pipe (47 lbs). The weight of the support assembly 1AF03014X (53 lbs), while it exceeds the limitation of the criteria for the exclusion of the weight by one pound, represents 2% of the weight of a 1-foot section of pipe (52 lbs) and is considered not significant in the context of the low stresses in the piping system and the conservativeness inherent in the analysis techniques used.

FINDING 3-7: OVERLAP OF PLATE

This finding indicates that support drawing 1AF03009R, Revision E, shows a 1/4-inch overlap between the edge of embedded plate and expansion anchor plate, and that the 1/4-inch overlap is a violation of Drawing M-919, Section F-2, Figure 29, which specifies a minimum overlap of 1½ inches. This finding stated that the technical significance with respect to this support should be evaluated to determine if it has any effect on the design.

RESPONSE

We do not agree with the conclusions of this finding. This finding was based on Revision E of support drawing 1AF03009R, without any consideration of other material provided to the IDI Team during the inspection.

FCR F-9079 was written on 4-16-81 to request approval of the 1/4-inch overlap instead of 1½-inch overlap stated on drawing M-919, Section F-2. FCR F-9079 was reviewed and approved with a backup calculation. Subsequently, the FCR was incorporated on Revision C drawing of 1AF03009R and issued on 12-23-81.

The IDI Team was provided a copy of FCR F-9079 and a copy of the backup calculation supporting the FCR although this information was not documented in the IDI Report.

In summary, S&L had evaluated the change to determine no effects on the design through the review and approval process of the FCR; therefore, we do not agree that this is a valid finding.

FINDING 3-8: RELOCATION OF SUPPORT

A field walkdown conducted to examine the pipe supports for the 100 percent as-built subsystem 1AF14 indicated the existence of two different supports having the identical designation 1AF14009R. Hunter Corporation had originally requested relocation and redesign of the original configuration 1AF14009R via a discrepancy report (Reference 3.220; see also References 3.221 and 3.222). The new support was properly designed and installed, but Hunter Corporation had no document on file that explicitly flagged the removal of the original configuration. The team did not determine the exact cause of this error, which should be addressed in the resolution of this item. (Finding 3-8)

Corrective Action Taken And Results Achieved:

Hanger Field Installation problems normally require only the modification/revision of a portion of an installed support. The resolution to the field problem on support 1AF14009R relocated the entire support along the pipe. The field crew received a revised drawing and installed the revised support in the new location without the knowledge that a revision had previously been installed. The voided revision has subsequently been removed.

Hunter Corporation S.I.P. 4.201, installation verification, has been revised to include a requirement for the Type 4 inspection to verify the removal of temporary and deleted Hardware/ Supports.

At the time the IDI was performed, the Type 4 inspection (per Hunter Corporation S.I.P. 4.201, installation verification), had not been performed on subsystem 1AF14. For pipe supports, a Type 4 inspection consists of a walkdown of a complete support subsystem utilizing a list of all valid supports to verify that previously accepted installations are intact and undamaged. The walkdown checklist, at the time of the IDI, identified only valid supports. Temporary or invalid supports are identified by Q.C. personnel for removal (by production crews) or resolution (by the Architect/Engineer). The redundant installation of pipe support 1AF14009R would have been identified during the Type 4 inspection of subsystem 1AF14.

FINDING 3-9: FORCE AND MOMENT DIRECTION

This finding addresses the fact that an incorrect nozzle load coordinate system was used by the auxiliary feedwater pump vendor due to the correct system not being defined in the equipment specification. This error had no effect on the pump design and resulted in a small increase in pump stresses, still well within ASME allowables.

Five additional pump specifications were checked. In three specifications the nozzle load coordinate system was defined, and in the other two it was not. However, in one of these a letter was later sent with new loads and the coordinate system to be used, and in the other the vendor included the piping in the pump model up to the point it became embedded in the floor slab.

It was not determined if other specifications written prior to 1977 contained adequately defined nozzle load coordinate systems.

RESPONSE

We agree with the conclusions of this finding that the specification did not delineate the nozzle load coordinate system, and consequently, an incorrect nozzle load orientation was used in the seismic analysis of the auxiliary feedwater pump. We agree that there is no effect on the pump design. Additionally, actual piping loads from the formal piping analysis have been checked and determined to be less than the piping nozzle loads as used by the vendor in the pump seismic report (Ref. S&L File CQD-008303).

We also agree that of the five pump specifications checked, two did not delineate the load coordinate system (the containment spray pump and the essential service water make-up pump) and the other three did. However, piping loads for the containment spray pump were transmitted by S&L letter dated 1-22-80, used in the seismic report and documented in the S&L review checklist (S&L File EMD-025038 dated 8-12-80). The essential service water make-up pump discharge piping is included in the pump model up to the pipe embedment and is noted in the S&L review checklist (S&L File EMD-009698 dated 8-23-77). This documents that at the time of the review of the seismic reports for these pumps the reviewer was aware of the S&L letter and the modelling of the pipe and determined that these were acceptable.

All safety-related S&L specifications on the Byron/Braidwood projects have been reviewed and it was determined that the nozzle load coordinate system was correctly specified in all except the two above and one tank which had the loads separetely transmitted by S&L letter dated 1-23-76. These loads were used correctly in the seismic report (S&L File EMD-006761). This is documented in S&L File CQD-008699 dated 7-18-83.

FINDING 3-10: PIPE SPAN LENGTH

This finding indicates that as-built dimensions that Nuclear Power Services obtained from Hunter Corporation were incorporated into the Nuclear Power Services as-built piping isometrics. In one instance, incorporation of piping as-builts caused the span length (Ls) between supports 1SX92012R and 1SX92013R to increase to 10 feet, 7-1/8 inches, which exceeded the maximum span length (9.5 feet) given in Sargent & Lundy's Small Piping and Tubing procedure.

The Sargent & Lundy procedure provided an installation tolerance on support locations of ± 6 inches, a total of 1 foot, 10 inches for each span. This finding states that Nuclear Power Services' analysis failed to note this change in its review of the piping Subsystem 1SX92, and also states that it is an isolated case of review error and the discrepancy will not result in support loads or pipe stress that will exceed ASME code allowances.

RESPONSE

We agree with the IDI Team that the as-built dimension, 10 feet, 7 1/8 inches between supports 1SX92012R and 1SX92013R, exceeded the maximum allowed span of 10 feet, 6 inches (9 feet, 6 inches plus total installation tolerance of 1 foot 0 inch), and that this error should not have any adverse effect on the piping or support stress analysis and did not appear to indicate a systematic weakness. Nuclear Power Service issued a request for information (RFI-799) to S&L on 6-23-83, for approval of the overspan between the supports 1SX92012R and 1SX92013R and S&L reviewed and approved the change.

FINDING 3-11: PIPE DAMPING VALUE

The finding stipulates that 4% damping was used for SSE analysis on piping having diameters 12 inches and greater, and this is inconsistent with the requirements of the FSAR for balance of plant piping.

RESPONSE

The 4% damping for SSE analysis of piping 12 inches and greater was used on subsystems attached to the reactor coolant loop. The Westinghouse criteria of 4% damping for SSE analysis of NSSS piping is applicable to piping attached to the reactor coolant loop. The FSAR will be amended to clarify the use of 4% damping for piping only 12 inches and larger attached to the reactor coolant loop.

Further, Westinghouse audited the piping analysis input with respect to damping used for all piping. This review of all 521 subsystems in the Westinghouse scope, of which 20 subsystems contained piping 12 inches and greater, indicated that there were two subsystems not attached to the reactor coolant loop that used the 4% SSE spectra. Westinghouse has reevaluated these two subsystems using the appropriate SSE spectra, and no modifications were necessary. Included in the remaining 519 subsystems are 9 subsystems attached to the reactor coolant loop that have piping 12 inches or greater, which were analyzed using 4% spectra in accordance with the applicable NSSS criteria. All the remaining 510 subsystems used the 2% spectra applicable to piping less than 12 inches in diameter. This included subsystems having piping 12 inches and greater but coupled with piping less than 12 inches and, therefore, the more conservative 2% spectra were used.

FINDING 3-12: SPRAY ADDITIVE TANK CALCULATION

The finding indicates that there is poor document control because a preliminary calculation was not noted preliminary or voided when the vendor's actual allowables were received.

RESPONSE

We do not agree with the conclusions of this finding. Westinghouse performs the quality aspects of the Byron piping project and the design control in accordance with the Westinghouse Quality Assurance Program in WCAP 8370. There is no specific violation of document control in accordance with the Westinghouse procedures.

The initial calculation for the nozzle load evaluation was based on the best available information at the time. The evaluation was subsequently redone when the actual allowable loads were received from S&L. This calculation was also available in the piping analysis notebook and pointed out to the IDI Team. Therefore, there is sufficient document control without voiding or revising the original calculations.

In addition, there are checks in the process that are applied by Westinghouse to ensure control of the design process. These include:

1. Independent engineering review of calculations;

2. Internal audits against QA procedures;

3. Review of analytical results at 70% and 100% as-built stages.

For equipment nozzle load qualification, the following steps are used:

1. At the initial design stage (Phase 1 analysis), the equipment loads are compared to the best available information regarding the nozzle limits. In the case of the spray additive tank, in lieu of actual vendor limits, the actual loads were compared to Westinghouse limits for similar equipment.

2. At the 70% support installed as-built reconciliation stage, the loads are submitted to S&L for final review and approval for S&L-supplied equipment.

3. During the 100% support installed as-built reconciliation stage, the equipment loads are finalized and retransmitted to S&L if the loads have changed. For the spray additive tank, the 70% as-built loads remained valid.

To ensure design control for evaluation of Westinghouse piping loads on non-Westinghouse supplied equipment, Westinghouse furnishes all equipment nozzle loads to S&L for final review and approval. This is done whether or not the limits are satisfied.

We believe that there is no deficiency to the document control applied to the work at Byron.

FINDING 3-13: PIPE MOVEMENTS IN RELATION TO GAP CLEARANCE

The finding addresses the adequacy of the calculation done to check the piping displacements at wall penetrations 1AB16080 and 1AB38 to verify modeling assumptions and concludes that the calculation failed to confirm the modeling assumption.

RESPONSE

We do not agree with the conclusions of this finding. To ensure that all penetrations are reviewed, S&L has supplied details of each penetration, incorporating the final radiation barrier design for all subystems. Westinghouse then reviews the pipe displacements at these penetrations to verify modeling assumptions. The penetrations are systematically reviewed by Westinghouse in the piping analysis process.

The specific calculations mentioned in the finding were done to check pipe displacements at wall penetrations and tabulate the displacements in the three directions. This tabulation showed that the magnitudes of these movements were small. Specifically, for the two penetrations cited in the finding, the calculated global displacements (X, Y, and Z) are 0.135, 0.074, 0.053, and 0.149, 0.073, 0.082 inch compared to the allowable gaps of 0.375 inch and 0.25 inch, respectively. The analyst did not explicitly perform the "last step" of obtaining a vector displacement to compare against the gaps within the penetrations because the displacements were so small. The magnitudes of the calculated displacement components were such that even the SRSS of all three components are less than the gap for both penetrations. Therefore, although the calculations did not include vector displacements, the proper conclusions were appropriately drawn, confirming the modeling assumption. The calculation of the vector displacement has been made, and verifies the previously drawn conclusion.

FINDING 3-14: BRANCH LINE ANALYSIS

The finding addresses the evaluation of the 1/2-inch branch lines 1CS22BA and 1CS22DA which were not included in the model for Subsystem 1CS09, and concludes that the analysis does not support seismic qualification of the lines.

RESPONSE

We disagree with the conclusion of the IDI Team that the seismic qualification is not adequate. The evaluation of the instrument line was simplified by using a conservative valve weight and peak response spectra acceleration values. The weight of the flexible hose was not included and was deemed negligible. A more detailed evaluation was performed as recommended by the IDI Team, explicitly following the load combinations set forth in Ref. 3.178 of the IDI Report. This resulted in calculated stresses approximately a factor of 3 lower than the simplified results. It should be noted that, since instrument lines do not experience valve thrust loads, the appropriate load combinations are Items B.a and D.b of Ref. 3.178 and not B.d and D.d as stated in the IDI Report.

Per ASME Section III-NC3650 methodology, the calculations of axial and shear deadweight and seismic stresses are not required. The simplified calculations considered the effects on the cantilever branch line to be purely bending, using an SRSS of the three directions of peak seismic accelerations. The effects of torsion were shown to be small in the more detailed evaluation as indicated by the factor of 3 reduction in stresses as stated above.

Typically, these instrument lines are included as part of the main piping models. Separate hand calculations are performed in cases when they are not included with a main piping model. As demonstrated above, the hand calculations which were performed prior to the IDI result in conservative values of pipe stress. Furthermore, to ensure unformity in this type of evaluation, standard methods are now incorporated into the Westinghouse Procedure and Guideline Manual for the Byron Project.

FINDING 3-15: HANGER DIMENSION

There is a discrepancy between the piping isometrics and the support drawings involving the location of two supports (M-1CS09010X and M-1CS09014X).

RESPONSE

Discrepancies between Hunter piping isometric drawings and Westinghouse support designs do occur in the design process. However, these are expected and are resolved systematically.

The process that resolves discrepancies between Hunter as-built isometrics and design dimensions requires that CECo be notified of differences in the as-built isometric dimensions, and that the isometric be reissued if the discrepancy is on the as-built isometric. Westinghouse site personnel review all 100% as-built packages to verify the correctness of all documentation. Information on support drawings is compared to that on the isometric drawings. This process had not been completed at the time of the IDI for the hangers in question, but has since been completed.

For the specific instance cited in Finding 3-15, field personnel have verified the support location by measuring the dimensions from the top elbow down to suports 1CS09010X and 1CS09014X. In both cases, the information on the Westinghouse design documents was correct. The dimensions used for piping analysis are from the Westinghouse design documents. Commonwealth Edison has been notified of the error on the Hunter Isometric drawing for 1CS09 and has reissued the isometric. There are no violations in documentation or design deficiencies for the supports cited in the finding.

FINDING 3-16: U-BOLT ANALYSIS

The calculations for support 1CS09001G revealed that a cantilever angle section and a U-bolt were overstressed. A decision not to redesign the U-bolt was not supported by calculations at the time of the IDI.

RESPONSE

We do not agree with the conclusions of this finding nor do we agree that it is a systematic problem. For all supports that require modifications, support calculations are completely verified after modifications are completed. The cycle for the design process had not been completed as discussed below and therefore we do not believe that this should be a finding.

At the time of the inspection, the calculations that document the adequacy of the U-bolt had not been verified. When a support on an as-built subsystem is found to be structurally inadequate, an ECN is prepared by Westinghouse site personnel to correct the identified problem. After the modification is completed, CECo issues a revised 100% as-built package. This package is transmitted to Westinghouse along with the field ECN. The package is reviewed by the design engineer to confirm that the field modification incorporated all necessary changes to confirm structural adequacy. In the event that the field ECN does not resolve the design deficiencies, the design engineer would again notify the Westinghouse site personnel and the process would be repeated. However, a review of the U-bolt calculation has since been completed and documented and no changes in the U-bolt are required.

Furthermore, in the above-cited finding, the revised 100% as-built package was issued to Westinghouse by CECo on October 24, 1983. Field ECN #53212 corrected the overstress in the cantilever angle section by replacing the angle with a tube steel section. To facilitate construction, the U-bolt was removed and replaced by a frame of angle sections around the pipe. The design engineer has confirmed that the field modification incorporates all necessary changes to confirm structural adequacy.

FINDING 3-17: PIPING RESPONSE FOR CHANGED SPAN LENGTH

It is the opinion of the IDI Team that the results of the engineering evaluation of the support relocation (1CS09004R) on 1CS09 is not adequate for seismic qualification.

RESPONSE

We do not agree with this conclusion, and based on the discussion below, we feel that the method to resolve the discrepancy between the analyzed configuration and the as-built configuration is fully acceptable. The analyst reviewed the analysis package and established, by calculation, a maximum support load increase of 40% to be appropriate due to the relocation of vertical support (004R). The assessment included load increases on the moved support as well as the two adjacent vertical supports (001 and 005). Although a precise prediction of the effect of this relocation would have required a computer reanalysis, the use of a hand calculation was deemed appropriate in this case, considering the limited changes from the analyzed configuration and the large margins in the pipe stress, tank nozzle loads, and valve accelerations. These actuals versus allowables are tabulated in the Tables F3.17-1 through F3.17-3.

To verify the adequacy of the assessment made in this reconciliation process, an SSE computer reanalysis was conducted with the support relocation included. This resulted in SSE support load changes of -15%, +20%, and +33% from the original analysis for supports 001, 004, and 005, respectively. Therefore, the +40% increase estimate for all three supports was indeed conservative. In addition, the displacements at the penetration 1AB38 are unchanged. Furthermore, the piping stresses were virtually unchanged. Reanalysis of 1CS09 with the support location deviation included, demonstrated that the estimated support load increases were conservative, and justified the simplified analysis used in the original evaluation.

The determination for performing computer reanalysis is based on the degree and number of as-built deviations, the margin in stress limits and equipment loads, and the adequacy of addressing the deviations using hand calculations and qualified engineering judgment. The validity of this approach has been demonstrated with respect to the piping system cited in the finding. The issue of the overstressed component of a support system was discussed in the response to Finding 3-16, and as stated there, the review cycle for the support modification would resolve any such overstress.

TABLE F3.17-1

PIPE STRESS SUMMARY

NC3650 EQUATION	ACTUAL STRESS (ksi)	ALLOWABLE STRESS (ksi)
8	1.9	18.6
9	4.5	22.3
10	8.2	28.1
9 Faulted	7.6	44.6

TABLE F3.17-2

VALVE ACCELERATIONS
(In g's)

LVE ...S. NO.	OBE						SSE					
	ACTUAL			LIMITS			ACTUAL			LIMITS		
	Ax	Ay	Az	Ax	Ay	Az	Ax	Ay	Az	Ax	Ay	Az
1CS017B	0.70	0.17	0.15	2.25	2.5	2.25	1.15	0.32	0.28	2.5	3.0	2.5
1CS040B	0.28	0.14	1.05	2.25	2.5	2.25	0.53	0.28	2.0	2.5	3.0	2.5
1CS018B	0.32	0.42	0.51	2.25	2.5	2.25	0.57	0.75	1.01	2.5	3.0	2.5

TABLE F3.17-3

SPRAY ADDITIVE TANK ALLOWABLE LOADS
(lbs, in.-lbs)

ACTUAL LOADS	LIMITS
Fx = 130	F1 = 3500
Fy = 270	F2 = 850
Fz = 100	F3 = 850
Mr = 5000	Mr = 25000

FINDING 4-1: TRANSVERSE WALL LOAD CRITERIA

In Section 11.0 of the Project Design Criteria, page 11-4 contains a listing of transverse loads to be considered in the design of walls. This list omits horizontal seismic inertial loads, wind loads, and tornado differential pressures. This is not appropriate. It is considered to be a failure to follow Procedure GQ-3.04 of the Sargent & Lundy Quality Assurance Manual (Reference 1.36). The preparer of the design criteria did not include all "applicable design inputs" in that numerous horizontal loading sources were not listed within the list of transverse loads to be considered for wall design.

In view of the inappropriate criteria, in our judgment, a systematic check of all walls to see that all loads were considered should be made in resolving this item.

RESPONSE

We do not agree with the conclusions of this finding. The Sargent & Lundy Structural Project Design Criteria, DC-ST-03-B/B (I.D.I. Reference 4.31) is appropriate. Item D Section 11.6.2.1.1 of Reference 4.31 states that "miscellaneous loads are considered where applicable" and since wind and tornado affect only periphery walls, they are considered miscellaneous. Also, Section 10.2.2.2 states specifically that shear walls shall be designed for wind load. Section 10.2.3.2 states that all safety Category I structures w'll be designed for tornado loading. Out of plane, horizontal self-weight seismic excitation is considered negligible in the design of shear walls. To demonstrate this to the IDI Team, a conservative calculation was done on a wall which would be sensitive to the effect of self-weight excitation, but the resulting increase in stress was only 2%. The span and thickness of this wall are representative of the shear walls in the plant. In order to maximize the effect of the self-weight excitation, conservative boundary conditions were used when calculating the increase in stress.

We believe that the foregoing information demonstrates that the criteria are appropriate. In addition, in the normal design process, we perform a final load check on all the walls, which will incorporate all loads in question. Therefore, we consider this finding to be resolved.

FINDING 4-2: SHEAR FRICTION METHOD

The Sargent and Lundy Structural Project Design Criteria (Reference 4.31) states that the shear friction concept shall be used to calculate the reinforcement required for transverse shear. This is contrary to Section 11.15.1 of ACI 318-71 (Reference 4.72) which the licensee committed to meet in FSAR Table 3.8-2. This is contrary to GQ-3.04 since the design criteria cited by the licensee in the FSAR was not incorporated within the project structural design criteria.

RESPONSE

We do not agree with the conclusions of this finding. The statement that the shear friction concept shall be used to calculate reinforcement for in plane and transverse shear was intended to apply only to the combined effect of these shears in the design of vertical reinforcement. Section 4.c of the Sargent & Lundy Structural Project Design Criteria, DC-ST-03-B/B (I.D.I. Reference 4.31) page 11-6, states that the steel required for transverse flexure and shear shall be combined with the other reinforcement. In the initial design, the walls were systematically checked for transverse shear as diagonal tension according to the ACI Code and shear ties were added as required. In most cases, ties were not required. Thus, the design criteria does agree with the FSAR, all required design considerations for transverse shear were met, and an additional systematic check is not required. However, the design criteria will be revised to clarify this item.

FINDING 4-3: TORSIONAL RESPONSE

Sargent & Lundy personnel indicated that the horizontal floor responses were calculated at the mass center of each floor. This amounts to ignoring the effect of torsion on response spectra for locations at the perimeter of the building.

During the IDI, Sargent & Lundy performed a study and determined that the horizontal component of torsion at the most unconservative location was 13% of the horizontal floor acceleration at the mass center. At this level, it would not be expected that neglect of torsion would adversely affect the design. However, this was an instance where a reasonably substantial effect associated with a specific FSAR commitment, was not considered or addressed in the documented design calculation. This was a failure to follow the FSAR commitment contained in Section 3.7.2.11. In this case, the assumption was made that the torsional effect at the periphery of the building was negligible, but the magnitude of this effect was not studied until the issue was defined by the Team.

RESPONSE

We agree with the IDI Team's observation that the effect of torsion for locations at the perimeter of the building was not included in the seismic response spectra. The response spectra have been generated at slab lumped mass locations, and these response spectra are considered representative for the entire slab. The decision to use such a methodology was based upon accepted industry practices in the field of dynamic analysis of shear wall structures. We agree with the IDI Team that this procedure does not adversely affect the design.

There are no commitments given in the FSAR to generate response spectra at the perimeter of the building. Subsection 3.7.2.11 states:

> "The floor slabs in building structures, along with the heavy equipment resting on them, have asymmetric mass-stiffness distribution. Therefore, the slabs will rotate about their vertical axes when these structures are subjected to lateral seismic loads. This torsional response was accounted for in the horizontal building model by including a torsional degree-of-freedom in each slab. However, the effect of the torsional component of ground motion during the earthquake on the response of building structures was considered insignificant and was not included in the analysis of Byron/Braidwood building structures."

The above FSAR description only addresses the modeling techniques used to account for torsion in the structural loading analysis of the building. It does not discuss the method of response spectra generation. The inclusion of a torsional degree-of-freedom for each slab in the building model does not imply that response spectra will be generated at all points on the slab.

FINDING 4-4: SHEAR STIFFNESS OF WALLS

The shear stiffness of the walls was calculated on the basis of the average thickness of the wall including pilasters. This overestimates the stiffness of the wall and therefore, can be unconservative. Where this occurred, the wall stiffness was increased by only a few percent. Therefore, we would not expect this to have any effect upon the design since the small overall difference is well within the margins provided by other conservatisms. This, however, constitutes a failure to follow the procedures provided in GQ-3.08, Design Calculations of the Sargent & Lundy Quality Assurance Manual (Reference 1.36). The Reviewer in this instance did not properly execute 'a review to determine if the input data ... is (sic) consistent with the design input' in that the actual design input was modified to become input data for the computer-aided design calculation in a manner incorrectly reflecting the actual physical configuration.

RESPONSE

We do not agree with the conclusions of this finding. The Byron and Braidwood models do include pilasters in shear wall stiffness and the assumption regarding pilasters will be documented as part of the calculation documentation upgrade covered under Finding 4-5. The area of the pilaster is loaded by the slab during an earthquake and thus should be included in the model. However, neglecting their area would not have a significant effect on design. We agree with the IDI Team that we would not expect this to have any effect on design since the overall difference is well within the margins provided by other conservatisms.

In addition, it is our judgment that the S&L reviewer of the original calculation did a proper review because he did agree with the treatment of pilasters in the modeling. Therefore, we consider this finding to be resolved.

FINDING 4-5: BUILDING CALCULATIONS

In Calculation 4.1.1 (Reference 4.25), we found a number of documentation deficiencies. These included failure to correct (update) calculations, lack of calculations, and statements indicating incomplete checking.

These items are violations of Sargent & Lundy Procedure GQ-3.08 (Reference 1.36). Sections 3.0 and 4.0 require indexing of computer runs, revisions to calculation sheets, statement of objectives, listing of data and assumptions used in the calculation, calculations recorded in neat and orderly manner, and verification that computer program is suitable. Although these types of deficiencies render the calculations confusing and difficult to follow, we did not find any instances where they had an adverse effect on the actual design and in our judgment, these specific items are not likely to have any significant effect on design.

RESPONSE

Although we believe that the original Calculation 4.1.1 can be followed, perhaps with some difficulty in some areas, this calculation is being upgraded to current documentation standards. S&L agrees with the IDI Team that the document deficiencies have no significant effect on the design.

FINDING 4-6: SUPPORT WELD ANALYSIS

A review of support 1AF14009R indicated that a field change request, FCR F-53681, required certain changes including a reduction in the weld lengths at the embedment plate from 6 inches to 4 inches This was not reflected in the calculation made at the time. This reduction was made more significant because bending moment is applied longitudinally to the weld. This change, made through the use of field change request, did not receive proper engineering disposition since the design calculation did not reflect the actual as-built conditions. This was contrary to the requirements of Sargent & Lundy project instructions contained in PI-BB-13, 'Procedure for Processing Commonwealth Edison Company Field Change Requests (FCR's)' (Reference 1.14). Section 4.4.2.2 of that instruction defines Sargent & Lundy as being responsible for the revision of all design documents necessary to complete action on the field change request. In this case, the revisions to the weld were not included in any new calculations.

RESPONSE

We agree that a weld calculation is required. The omission of this calculation is an isolated incident, because in similar calculations, weld adequacy is checked and a calculation is provided. During the IDI, calculations were performed for the as-built conditions. The results of those additional calculations demonstrate the adequacy of the weld.

Subsequent to these calculations, changes in design necessitated that the load on this hanger be increased and new calculations were prepared, reviewed, and approved on September 1, 1983 for revision D of drawing 1AF14009R. The new calculations no longer require consideration of "bending moment ... applied longitudinally to the weld ..." because the free end of the main member is now designed with an expansion anchor assembly. Thus, the capacity of the weld group is no longer controlled by stresses induced by bending moments.

FINDING 4-7: SUPPORT WELD ANALYSIS

A review of support 1AF14010X indicated that as a result of field change request, FCR F-53726, a calculation was performed. An error in the calculation of weld stresses was made caused by transposing values for member stress rather than member moments and forces. This random error constituted a violation of Sargent & Lundy Procedure GQ-3.08 (Reference 1.36) in that the reviewer of the hand calculation did not discover the technical inadequacy of the specific calculation.

RESPONSE

We agree that the calculation for FCR F-53726 written against support 1AF14010X did contain an error. The original calculation was reworked during the IDI and the support was verified to be adequate as detailed in the FCR. We also agree with the IDI Team that the stresses in the weld were nonetheless acceptable, and that this was a random transposition error, because other previous calculations by the same preparer showed that weld stresses had been calculated correctly. In addition, a review was made of previous work by the same reviewer, and this showed that he had not made similar oversights, indicating his error was not systematic.

FINDING 4-8: ELEVATION OF CONDUIT SUPPORT

This finding concerns the elevation of Conduit Support FC-3. The finding states that conduits were installed about two (2) feet higher than shown on electrical installation drawings.

RESPONSE

The elevation deviation was identified and submitted for engineering review to ascertain if there was an impact on design.

The IDI Report states as follows:

"After the licensee was made aware of this condition, Hatfield Electric Company, the installing electrical contractor, performed a routine quality control inspection. A discrepancy report (Reference 4.38) was prepared as a result and the information incorporated with other corrected as-built information into a field change request (Reference 4.39). These documents contained the correct information in that the actual centerline elevations were judged to be in agreement with those determined by the team."

The above is a true statement.

The IDI Report, however, fails to mention that the discrepancy report which was prepared, discrepancy report DR #2151 (Reference 4.38), was dated 05-19-83 and the field change request, FCR F23154, was dated 05-24-83. Both of these documents were prepared prior to the IDI being conducted.

As a result, this item does not represent a finding, but merely the agreement of the IDI Team that the site contractor had correctly prepared and submitted the required documents per Commonwealth Edison's Procedure QP No. 3-2, Section 4.4.

FINDING 4-9: STARTING BATTERY SUPPORT RACK DRAWING

This finding concerns a drawing detail in which an end view of the battery racks and anchorage details was inconsistent with the plan view of same.

RESPONSE

We agree that the end view of the battery racks and details were inconsistent. The drawing has been revised to delete the extraneous end view.

We also agree with the IDI Team that:

A. this finding represents an isolated minor drawing error resulting from the selection of a new vendor for the batteries and racks,

B. the installation was properly made and is in conformance with the design, and

C. the minor drawing error is considered to be a random oversight.

FINDING 4-10: CONDUIT PIPE STRAP

This finding concerns an uninstalled conduit strap.

RESPONSE

We do not agree with the conclusions of this finding. This item represents the identification of a previously identified discrepancy for which the proper documents had been processed but which had not as yet completed their review and closure cycle.

The missing conduit strap identified during the IDI was originally identified by as-built FCR F8569 on 01-08-82. It was also identified on the contractors QC Inspection Checklist (form HP203 dated 08-09-82), for the conduit run.

To date, FCR F8569 has not been closed out.

During the IDI, when the IDI Team identified the missing conduit strap, the contractor generated discrepancy report DR #2375. The requirement for the conduit strap has been deleted and DR #2375 was closed on 07-06-83.

FINDING 5-1: ANALYSES CONCERNING LACK OF SEPARATION

This finding concerns the adequacy of analyses in Interface Review Reports which are used to justify lack of separation between Class 1E and Non-Class 1E cables within panels. The finding states that some analyses did not address all of the potential means by which Non-Class 1E cables might degrade Class 1E cables and alleges that such analyses are contrary to licensing commitments contained in FSAR Appendix A. The finding also states that some Interface Review Reports were not updated to reflect licensing commitments that had taken place subsequent to the preparation of the Interface Review Report.

RESPONSE

We do not agree with the conclusions of this finding. Interface Review Reports only represent a portion of the analyses done to justify the independence of electric systems as a licensing commitment in the FSAR. We agree that some of the analyses associated with Interface Review Reports did not document all of the potential means by which Non-Class 1E cables might degrade Class 1E cables. This does not mean that all of the potential means by which Non-Class 1E cables might degrade Class 1E cables were not considered in the analysis. Certain conservative and generic design practices are used in preparing the Interface Review Reports, and are documented elsewhere. Following are a few examples of such conservative and generic design requirements as practiced by Commonwealth Edison Company:

(1) All power and control cables are qualified to Class 1E requirements including conformance to IEEE-383 (e.g., fire tests) even though most are used in Non-Class 1E applications. Thus, the Non-Class 1E cables are qualified to environmental requirements which are much more stringent than necessary. This conservative design provides assurance that Class 1E cables will not be degraded as a result of fire propagation by the Non-Class 1E cable.

(2) All control cables are nominally rated (insulated) for 600 volt applications but are applied in 120-Vac and 125-Vdc control circuits - thus providing a nominal (insulation) design margin of almost 500 percent. This conservative design provides assurance that the Class 1E cable will not be degraded by voltage transients in the Non-Class 1E cable.

(3) All control cable conductors are not less than No. 14 AWG, with a continuous current rating (design ampacity) of approximately 7 A. The majority of these control cables are utilized for a-c motor control applications which are powered by control transformers having a rated current of approximately 1 A and a short circuit capability of approximately 10 A; i.e., the short circuit current is only slightly above the continuous current rating of the control cable. This conservative design provides assurance that the Class 1E cable will not be degraded by current transients in the Non-Class 1E cable.

Thus, this finding appears to represent a judgment by the IDI Team that there is insufficient information delineated in the documentation on each individual Interface Review Report. Based on the above points, we do not believe that it represents incomplete analysis or a failure to meet licensing commitments.

Those Interface Review Reports that were not updated to reflect licensing commitments that have taken place subsequent to their preparation, were in the process of being revised at the time of the IDI. This was shown to the IDI Team.

In order to make the analyses more clearly understood by personnel outside of the Electrical Project Engineering Division, we have taken the following actions:

All Interface Review Reports have been reviewed. Those reports containing incomplete documentation have been revised to document all potential means by which a Non-Class 1E cable might degrade Class 1E cables, as stated in Appendix A (Regulatory Guide 1.75) to the FSAR. Those reports affected by licensing commitments made subsequent to the preparation of the report have been revised to address those commitments (e.g., addition of second circuit breaker). The battery charger (reference IDI Sample 7) is included in a list with many other Class 1E items, in a non-harsh environment, to be qualified by March 1985. If it should not become qualified, the Interface Review Report will be revised to justify the use of an unqualified battery charger. Interface Review Reports have been revised to reference calculations where, in the opinion of the analyst, such references are appropriate and necessary to understand his conclusion. In addition, Project Procedure BBP-7 has been revised to list the elements which should be included in Interface Review Report analyses.

There was no failure to meet licensing commitments. There is no case (either prior to, or subsequent to, the IDI) in which the analysis (Interface Review Report) indicates that there is inadequate separation between Class 1E and Non-Class 1E cables.

FINDING 5-2: FAULT CURRENT CALCULATION

This finding concerns the assumption of 14,000 Ampere available fault current in Calculation 19AQ-17. The finding states that the reference for the 14,000 Ampere assumption was not listed in the calculation as required by Sargent & Lundy's Procedure GQ-3.08.

RESPONSE

We agree that the source of the assumption should have been listed.

We also agree with the IDI Team that the omission did not affect the validity of the calculation. Calculation 19AQ-17 has been revised to reference the source document (Calculation 19AJ-1).

FINDING 5-3: HEAT DISSIPATION CALCULATION

This finding alleges three minor procedural errors associated with Heat Dissipation Calculation 19AI-18. The first error was that pages were not numbered 1 of 80, 2 of 80, etc., nor was the final page identified with a 'final page' designation as required by Sargent & Lundy's Procedure GQ-3.08. The second error was that the review method used was not documented in accordance with Sargent & Lundy's Procedure GQ-3.08. The third error was the use of an assumed motor efficiency of 95 percent when the actual vendor data on the motor efficiency was available at the time the calculation was performed.

RESPONSE

We do not agree with two of the alleged errors cited in this finding. We do agree that the review method should be documented. The following response to this finding addresses each of the alleged errors identified in the finding.

1. There was no error in page numbering. Calculation 19AI-18 is an 81 page calculation and clearly states "final" on page 81 of the calculation. No correction is required.

2. Sargent & Lundy agrees that the review method should have been documented in accordance with GQ-3.08. The review method used for Calculation 19AI-18 has now been documented.

3. There is no error in motor efficiency. The calculation clearly states that assumed motor efficiencies of 95 percent are used for all 4000- and 6600-volt motors. The use of assumed motor efficiencies for all motors simplifies the calculation. This assumption is valid due to the conservative factors and approximate nature of the calculation. As the IDI Report states, the use of the actual motor efficiency (94%) has no effect on the design. Furthermore, as acknowledged in the Report, there is a Departmental Standard (ESI-253) that requires that the input data and assumptions used in safety-related calculations be subjected to a final check. This (final check) program is in process; the subject calulation is on the list of calculations to be checked, and the final check will be complete prior to fuel load.

   A draft revision (never formally issued) of Sargent & Lundy Electrical Engineering Reference ESC-525 (Reference 5-25) is mentioned in the Report as requiring that efficiency figures for the motor given by the manufacturer should be used for the calculation when available. This reference does not govern the preparation of the calculation and is identified in the calculation references as only applicable for additional heat dissipation information not available in the governing Sargent & Lundy Standard. No correction is required.

FINDING 5-4: ALARM LIGHT LOGIC DIAGRAM

This finding addresses a logic diagram drafting error wherein a flow setpoint was shown as 100 GPM when it should have read 160 GPM.

RESPONSE

We agree that the 100 gpm logic diagram flow setpoint was a drafting error.

We also agree with the IDI Team that the error is an isolated case. We have further verified that other documents which specify instrument requirements and have been released for construction and purchase, specifically the flow switch and indicating light instrument data sheets, have the correct setpoint. The logic diagram has subsequently been corrected to indicate the correct flow data.

FINDING 5-5: ELECTRICAL DRAWING

This finding identifies a minor transcription error on a design drawing in which a conduit number was incorrectly listed.

RESPONSE

We agree that a transcription error on a design drawing was made. However, the IDI Report also contains a transcription error i.e., the correct Conduit Number is "57" (rather than "51", as listed in the report). This drawing error has been corrected to show Conduit No. "57".

We also agree with the IDI Team that the error was minor and "in general" the methods, procedures, and documents associated with manual routing of cables in the field (Cable 2AR190) are in good order.

FINDING 6-1: OPENING OF FLOW CONTROL VALVES

This finding states that the auxiliary feedwater air-operated flow control valves should, if closed, open on receipt of a safety injection signal as described in the auxiliary feedwater system design criteria (DC-AF-01-BB). This has not been implemented in the system design.

RESPONSE

We agree with the conclusions of the finding. However, the intent of the design criteria statement is to preclude leaving any of the eight flow control valves (normally open) closed after testing or maintenance. As an alternate design to supplying a safety injection signal to the flow control valves to assure that they are open, the design is such that closure of a flow control valve produces a signal to the Equipment Status Display indicating an inoperable status of an auxiliary feedwater train. The operator would reopen the valve(s) to return the auxiliary feedwater system to a normal standby and ready condition.

The current design more than meets the intent of the design criteria to assure that failure to reopen the flow control valves after testing or maintenance does not jeopardize the function of the auxiliary feedwater system. Please refer to the response to Finding 2-13 for a discussion concerning update of the design criteria.

FINDING 6-2: RELAY DESIGNATION NUMBER

This finding concerns an inconsistency in the bus undervoltage relay designation between an AF System schematic diagram and an EF System schematic diagram.

RESPONSE

We agree that there was an inconsistency on the drawing. We also agree with the IDI Team that this is an isolated (random) error.

The EF System schematic diagram was revised to eliminate the inconsistency.

FINDING 6-3: BASES FOR SETPOINTS

This finding states that no calculations exist for the auxiliary feedwater pumps suction pressure switches' setpoints. The lack of a calculation or other documented design basis for the setpoints is a violation of IEEE 279-1971.

The finding stated that this appeared to be a systematic problem that should be addressed.

RESPONSE

We do not agree with the conclusions of this finding. The suction pressure switch setpoints were based on the existing head loss calculations AFJD-1 and AFJK-1 and approved design drawings. A separate documented calculation specifically for the suction pressure switches was not performed prior to the IDI nor was it deemed necessary. A documented calculation (AFTH-02) was performed to verify to the IDI Team that the setpoint is consistent with the system design requirements. No design error in the setpoints was found which could be attributed to the lack of a separate and specific calculation.

In general, instrument setpoint bases are existing engineering calculations, design drawings, and/or vendor-supplied component design data. Separate setpoint calculations are performed for only those instrument setpoints that cannot be determined from existing data. Instrument setpoints determined from these sources are documented on prepared, reviewed, and approved instrument data sheets. This method of establishing instrument setpoints does provide a documented design basis in compliance with IEEE 279-1971.

The auxiliary feedwater pump suction pressure switch is probably the most complex and most safety significant switch in the Sargent & Lundy (S&L) scope. Discussions were held between the instrumentation personnel and the responsible engineers to determine the requirements for the switch.

To address the IDI concern, S&L will make a documented assessment of the safety-related C&I instruments in the S&L scope to iden-tify instruments that are judged to be complex in application and scope. A documented calculation will be provided for those instruments identified, if it does not already exist, to verify the adequacy of the setpoint, and will include a verification of the setpoint accuracy (Finding 6-7), and the reset valve (Finding 6-8), if applicable.

FINDING 6-4: CALIBRATION INTERVAL

This finding addresses a discrepancy between the test reports (References 6.28 and 6.29) and the proposed technical specifications regarding the calibration test interval for the auxiliary feedwater pump suction pressure switches (1PSL-AF051 and 1PSL-AF055). The test reports specify a test interval of 208 weeks, while the technical specifications specify a test interval of 18 months.

RESPONSE

The instrument maintenance calibration test reports for 1PSL-AF051 and 1PSL-AF055 were incorrectly marked with a calibration frequency of 208 weeks (4 years) instead of 18 months. The test reports have been revised to reflect the appropriate calibration frequency.

Byron Instrument Procedures 2000-003 and 2000-004 provide administrative controls for calibration intervals. Additionally, calibration requirements are being verified during the review of Byron technical specifications.

FINDING 6-5: SWITCH CONTACT OPERATION

This finding addresses an apparent inconsistency between the contact action specified on the instrument data sheet for pressure switch 1PSL-AF055 and the corresponding electrical schematic diagram where the pressure switch low-low interlock is shown.

RESPONSE

We agree that the instrument data sheet incorrectly indicated the contact action. Discrepancies between instrument data sheets and schematics occur during the design process and instrument data sheets, as a general practice, are revised as required.

Since the actual contact type specified for the pressure switch in question is an SPDT, it is not critical to specify the contacts as opening or closing on decreasing pressure. Specifying the contact as opening or closing in this case would be equally correct. It is important, however, to specify the contact action to occur at a decreasing or increasing setpoint and this was done. We would further note that the contact operation specified on the instrument data sheet does not control or jeopardize the contact implementation required on the electrical schematic diagram for the correct equipment function.

For the particular case noted in the finding, we have, since the IDI, revised the auxiliary feedwater pump suction pressure switch instrument data sheet to be in agreement with the electrical schematic diagram.

FINDING 6-6: TEST REPORT

The Commonwealth Edison test report for 1PSL-AF055 incorrectly states the direction of pressure change for the low suction pressure setpoint (Reference 6.29). In addition, this test report has a typographical error in the number for wire AF1BLA5 in that it was listed as AF1BLA3. At the time this test report was inspected, Commonwealth Edison actions to correct the Sargent & Lundy instrument data sheet and Commonwealth Edison test reports had not been initiated using the instrument discrepancy report form. Subsequent Commonwealth Edison actions to correct the test report and the data sheet discussed in Finding 6-5, while technically correct, appear to introduce unnecessary complexity in the designation of vacuum setpoint values. On the basis of our examination of other test reports (References 6.125, 6.126, 6.127, and 6.129), these errors did not appear to represent systematic weaknesses.

RESPONSE

As noted on page 6-10 of the IDI Report, actions were taken to correct the instrument data sheet and the test report using approved procedures. Also, as noted in the IDI Report, no evidence was found to suggest a systematic weakness in this area.

FINDING 6-7: SETPOINT ACCURACY REQUIREMENTS

This finding addresses an alleged need for setpoint accuracy to be specified on the instrument data sheet for pressure switches 1PSL-AF051 and 1PSL-AF055. The finding states that the absence of the setpoint accuracy indicates a systematic weakness and is not in compliance with IEEE 279-1971.

RESPONSE

We do not agree with the conclusions of this finding. The instrument data sheet is the documented design basis for the instrument in compliance with the general requirements of IEEE 279-1971.

The instrument data sheet contains the specific information to establish the requirements the instrument vendor must meet and therefore establishes the design basis.

In general, the information on the data sheet is obtained from an engineering assessment of information contained in system calculations, design drawings, and/or vendor supplied component design data. Instrument vendor specification sheets and/or catalog information are generally used as part of the assessment.

Setpoint accuracy is determined in a similar way. The setpoint accuracy required is then used (along with other instrument requirements) in the review of instrument vendor catalog information to establish instrument selection. Instruments selected and documented on the data sheets are vendor standard designs which are selected to envelop the system operating requirements as opposed to specifying special design characteristics (i.e., setpoint accuracy) which would meet a specific system requirement. See the response to Finding 6-3.

The instrument data sheet records specific information about the instrument, as well as, the manufacturer and model number. Vendor setpoint accuracy specifications, in addition to other design data and specifications, exist for that model number. This information supplements the instrument data sheet design information. We conclude this is consistent with IEEE 279-1971.

FINDING 6-8: BASIS FOR RESET VALUE

This finding addresses an alleged need for a pressure switch reset value design basis for 1PSL-AF051 and 1PSL-AF055. Compliance to IEEE 279-1971 is cited as the requirement.

RESPONSE

We do not agree with the conclusions of this finding. As stated in the response to Finding 6-7, the instrument data sheet is the documented design basis for the instrument in compliance with the general requirements of IEEE 279-1971.

The instrument data sheet contains the specific information to establish the requirements the instrument vendor must meet and, therefore, establishes the design basis.

In general, the information on the data sheet is obtained from an engineering assessment of information contained in system calculations, design drawings, and/or vendor supplied component design data. Instrument vendor specification sheets and/or catalog information is generally used as part of the assessment.

Reset values are determined in a similar way. The reset value required is then used (along with other instrument requirements) in the review of instrument vendor catalog information to establish instrument selection. Instruments selected and documented on the data sheets are vendor standard designs which are selected to envelop the system operating requirements, as opposed to specifying special design characteristics (i.e., reset value) which would meet a specific system requirement. Special design characteristics which would meet exactly the specific system parameter are not specified. In this particular case, the pressure switch reset was specified as the "manufacturer's standard" which enveloped the requirements. Therefore, specific calculations to determine an exact reset value were not required. See the response to Finding 6-3.

Finally, the instrument data sheet records specific information about the instrument, as well as, the manufacturer and model number. Vendor reset specifications, in addition to other design data and specifications, serve to supplement the instrument data sheet design information. We conclude this is consistent with IEEE 279-1971.

FINDING 6-9: PREOPERATIONAL TESTING

A commitment on FSAR page 14.2-31 states that during the auxiliary feedwater system preoperational test, 'motor and diesel-driven pumps will be verified to start under any safeguard situation under any possible control lineup, including restart capability, from any control station, following a protective trip.' Contrary to this commitment, no plan exists to test the actual operation of the pressure switch at its reset point under conditions where the pump is required to automatically take suction from the essential service water system. Commonwealth Edison and Sargent & Lundy personnel indicated understandable reluctance to test the auxiliary feedwater system using essential service water because that system's relatively impure water would necessitate cleanup of the auxiliary feedwater system and the steam generators. However, the test could be performed using water of better quality. Commonwealth Edison and Sargent & Lundy also indicated a belief that testing the individual components separately is adequate. However, we could find no planned test to demonstrate that the piping from the essential service water system will carry full flow without excessive restriction. In addition, no test of the entire integrated system response to a need for switchover to the safety-related water source is planned. The inconsistencies with the FSAR commitment should be resolved.

RESPONSE

The lack of planning to actually switch the suction of the auxiliary feedwater pumps to the essential service water system on low suction pressure does not constitute a violation of the FSAR commitment quoted on page 6-11 of the IDI Report. The suction alignment valving consists of simple open/close valving and, as such, does not constitute control lineups in the sense of the FSAR discussion. However, to clarify and put on record our intentions in this regard, page 14.2-31 of the FSAR has been revised via Amendment 43 (Sept. 1983) (copy attached). The IDI Team did indicate that they felt the test in question could be performed using "water of better quality". We wish to point out that the essential service water system has been filled for some time with strained river water. Even should extensive flushing and cleaning be done, it would be difficult to ensure that hideout contamination would not result in water chemistry that does not meet the stringent requirements for steam generators. Secondary side water chemistry has been a matter of significant interest in recent times, as its importance has gained wider recognition. Moreover, the piping from the essential service water system consists of a fairly short run of 6-inch pipe, containing two gate valves, fed from a 14-inch header. (The pump suction headers are six inch lines). This system is pressurized by the essential service water pumps. The small confirmatory gain resulting from the test desired by the IDI Team is wholly outweighed by the potential risk to the steam generator chemistry associated with such a test.

TABLE 14.2-19

AUXILIARY FEEDWATER SYSTEM

(Preoperational Test)

Plant Condition or Prerequisite

Prior to core load, prior to and during integrated hot functional testing, and during ECCS full flow testing, RCS will be at various temperatures and pressures.

Test Objective

To verify the ability of the auxiliary feedwater system to respond to a feedwater demand under any plant condition.

Test Summary

The auxiliary feedwater system will be tested prior to hot functional testing to verify pump performance over extended periods on recirculation, and at various flow rates. Motor and diesel-driven pumps will be verified to start under any safeguard situation under any possible control lineup, including restart capability, from any control station, following a protective trip. Control logic and interlocks for both manual and automatic operation and protective features for motor- and diesel-driven pumps and all power-operated valves will be verified for setpoint, indication, and alarms.

All motor- and diesel-driven pumps will be tested for five consecutive, successful cold starts per pump.

All motor operated valves will be verified to position or reposition to the required lineup from any plant condition, safeguard situation or suction requirement. Essential service water booster pumps attached to the diesel prime movers will be verified for flow and cooling requirements of the engine and cubicle cooler. All flow limiting devices will be verified by line flow checks and identification tab data.

Acceptance Criteria

The auxiliary feedwater system supplies feedwater in accordance with Subsection 10.4.9.3.1.

FINDING 6-10: STAMPING OF SAFETY RELATED DRAWINGS

This finding addresses the absence of a "Nuclear Safety Related" stamp from a control logic diagram (M-4037-1AF04) which shows safety related equipment.

RESPONSE

We agree with the conclusions of the finding that the control logic diagram M-4037-1AF04 was not correctly stamped "Nuclear Safety Related". We have corrected this minor error and reissued the drawing showing the "Nuclear Safety Related" stamp.

In addition, Sargent & Lundy has taken action to review the complete set of logic diagrams and the control and instrumentation diagrams for correct stamping and has revised, as required, those drawings which are not in conformance with the stamping requirements.

FINDING 6-11: IDENTIFICATION OF SAFETY-RELATED COMPONENTS

This finding addresses an alleged omission in classifying certain safety related controls cabinet modules as safety related.

RESPONSE

We do not agree with the conclusions of this finding. The channel test and relay card modules in question are contained in Control System Cabinets 1PA33J and 1PA34J.

Because of project practices which require instrument numbers only for those instruments which need the documentation of setpoint and/or calibration data, these modules were not numbered. In this case, instrument numbers are not required in order to provide a classification for the modules since the classification is provided by virtue of the cabinet classification in the project Equipment Li t, classification of the instrument loop in the procurement specification and classification of devices on wiring and schematic diagrams. We believe the current Sargent & Lundy documentation provides adequate classification of the modules as safety related hardware.

FINDING 6-12: EQUIPMENT STATUS DISPLAY SYSTEM CRITERIA

This finding describes a disagreement between the design criteria and the logic diagrams for deactivation of the Equipment Status Display.

RESPONSE

We disagree with the IDI Team's statement that the Equipment Status Display (ESD) system design criteria's statements regarding deactivation are contrary to IEEE 279 or the implication that FSAR commitments have not been met. Regulatory Guide 1.47, Section B (Discussion), specifically states that "certain safety-related functions of a nuclear power may be bypassed or made inoperable during the performance of periodic tests or maintenance". The ensuing regulatory guide discussion describes requirements for an indication system based on bypasses or inoperability during testing and maintenance. The ESD system is designed to meet these requirements and to meet IEEE 279 requirements in specific cases.

However, the generality regarding deactivation of the ESD after receipt of a safety injection signal, as listed in the design criteria DC-ME-07-BB, will be revised to reflect the actual implementation as noted on the logic diagrams (i.e., the auxiliary feedwater system is not deactivated after receipt of a safety injection signal). This specific design criteria will be revised since it provides important documentation for the ESD system.

Section II

Detailed Responses to Unresolved Items

UNRESOLVED ITEM 2-1: DIESEL ENGINE EXHAUST PIPE

The licensee should assure that the vulnerability of the auxiliary feedwater pump diesel exhaust pipe and its hinged cap has been considered. The revisions made to the exhaust piping of the emergency diesel generators should be reviewed for applicability to the auxiliary feedwater pump diesel exhaust piping.

RESPONSE

Rupture discs were added to the exhaust lines of the emergency diesel generators since a postulated tornado missile could potentially disable the exhaust line for each of the redundant diesels causing both trains to be inoperable. Only one of the two auxiliary feedwater pumps is diesel driven and potentially vulnerable to tornado generated missiles. The motor-driven auxiliary feedwater pump and components are entirely inside the auxiliary building.

Approximately 7 feet of auxiliary feedwater pump diesel exhaust piping is exposed above the auxiliary building roof. It was our engineering judgment that this short length of 16-inch Schedule 40 pipe would not crimp completely closed from a tornado missile and incapacitate the auxiliary feedwater pump diesel. Sargent & Lundy has performed a detailed review to calculate the affect of the tornado missiles on the auxiliary feedwater pump diesel exhaust stack. The review has confirmed our engineering judgment that the auxiliary feedwater system will not be incapacitated.

UNRESOLVED ITEM 2-2: MAXIMUM PIPING PRESSURE

All systems shall be verified to have calculations or documentation in place to assure the maximum pressure is within the pipings capabilities.

RESPONSE

The maximum pressure for the safety-related systems are being documented in the individual system piping design specifications. The maximum pressures are checked against the latest design minimum wall calculations (4-9-82) and Sargent & Lundy Mechanical Standard MES-2.5, Maximum Allowable Working Pressures to verify the adequacy of the affected piping.

We trust this additional information resolves this item.

UNRESOLVED ITEM 2-3: PIPE TIP DEFLECTIONS

Sargent & Lundy Project Instruction PI-BB-38 states that subsequent to pipe whip restraint design the final pipe tip deflections are transmitted to Project Management Division for review. This had not been done prior to the inspection and criteria for this review are not stated.

RESPONSE

Final pipe tip deflections became available during the IDI and were transmitted to the Project Management Division (PMD) as noted in the Inspection Report (IDI Report Reference 2.150). This is consistent with the schedule for finalization of pipe whip restraint design. Work in this area at late stages of construction is not abnormal because as-built dimensions can affect restraint calculations and design.

The Sargent & Lundy project instructions are written to standardize repetitive design functions and to define departmental interfaces. Project instructions are not written to define all design work. Final disposition of the tip deflections is being coordinated by the Project Management Division within the mechanical department. The final report will fully document the procedures and results.

This review is underway. The tip movements inside containment have been superimposed on composite drawings to check for interference with mechanical and electrical equipment. An additional field check is being made to insure that no field routed systems are within the tip deflection envelope.

Auxiliary Building breaks are treated by checking the location of the break to ensure that no safe shutdown systems are in the same area. Because of the compartmentalized design of the Auxiliary Building, no field check is necessary.

UNRESOLVED ITEM 3-1: ROD HANGERS AND PIPE REST SUPPORTS

The licensee should either consider the use of realistic support stiffnesses in piping analyses, or develop more conservative acceptance criteria for rod hangers and pipe rest supports.

RESPONSE

There is a contradiction in the IDI Report, since on Page 3-9, the Report states "the use of infinite support stiffness met the licensing commitments and our audit calculations did not indicate any adverse effects for the sample we selected."

Thus, the IDI Team basically agrees with the procedure used as an acceptance criteria for rod hangers and pipe rest supports. In light of the many conservatisms introduced into the design by the development of seismic inputs for the piping analysis, Sargent and Lundy believes that the "uplift check" method utilized is a more than an adequate engineering approach to justify the use of rod hanger and pipe rest supports.

The alleged unpredictability of the seismic support loads is something that was never discussed or brought up during the IDI. Sargent & Lundy subsequently reviewed Reference 3.113, "Piping Stress Evaluation For Independent Verification of Sargent & Lundy Design SAFF-261-83," as performed by EG&G and could not substantiate "the 70% unpredictability of weight, thermal, and seismic loads" statement in the IDI Report attributed to Reference 3.113 on Page 3-6 of the Report.

Sargent & Lundy agrees with the IDI Team's conclusion on Page 3-9 of the IDI Report and the conclusion arrived at in Reference 3.113 which states, "...results of the study indicate that the analysis procedures used by Sargent & Lundy adequately examine piping stresses per ASME Code Criteria." We therefore believe that adequate conservativeness exists and that all licensing commitments have been met such that this item is considered resolved.

UNRESOLVED ITEM 3-2: LATERAL VIBRATION OF STRUTS AND RODS

S&L employs no criteria to evaluate the possibility of lateral vibrations of struts and rods and no criteria was available to evaluate the frequency of supports in the unrestrained direction.

RESPONSE

S&L's practice for designing supports is considered more than adequate for the following reasons:

1. In conjunction with the applied piping loads, the self-excitation of the auxiliary steel provided for the purpose of supporting the piping is accounted for in sizing and designing the structural steel members.

2. The criterion outlined in Paragraph 6.4.5 of Engineering Mechanics Division's "Lesson Plan for Training Personnel in Piping Analysis," EMD TP-1, Revision 3 requires an evaluation to include the mass effect of support hardware in piping system dynamic analyses. By doing so, the generated loads contain the additional loads due to hardware self-excitation.

Based on the above discussion, S&L believes that a consistent support design practice does exist and the approach that S&L has adopted in addressing this issue is adequate. We trust this explanation resolves this item.

UNRESOLVED ITEM 3-3: DIESEL DRIVE STIFFENER

The seismic qualification test determined that the coupling between the diesel engine and the gear box was loose due to excessive flexibility in the engine mounts. Additional gussets were added to the gear box to resolve the problem, however, vendor drawings were not updated and issued to Sargent & Lundy. It should be assured that this design change had been incorporated in the other diesel drive units (Byron 2, Braidwood 1, 2).

RESPONSE

The diesel engine supplier, Stewart and Stevenson, has confirmed in writing that the gussets were added to all four of the auxiliary feedwater pump diesels' gear boxes. Stewart and Stevenson in-house production drawings were revised to show the subject gussets on 10-5-79. Due to the small detail, it was impossible to show these gussets on the general outline and installation drawings submitted to Sargent & Lundy. Sargent & Lundy and Commonwealth Edison have verified that the gussets are installed on the diesel engines at the site, which should resolve this item as requested.

UNRESOLVED ITEM 3-4: PIPE SUPPORT ADDED MASS

Westinghouse analysts do not consider the additional mass or weight that might be attached to piping by trapeze hangers, and we found no instructions for considering this effect. It should be verified that there are no such cases in the Westinghouse scope or that any cases that might exist are being properly considered.

RESPONSE

Westinghouse analysis criteria does not require the inclusion of the weight of trapeze hangers in the piping model. The effect is considered small, and furthermore, this type of support design is used infrequently by Westinghouse. Clearly there has been no violation of any Westinghouse criterion or FSAR commitment associated with this unresolved item.

Westinghouse has reviewed a random sample of 377 small bore supports selected from a total Westinghouse scope of 6291 in the containment and auxiliary building. The review found two trapeze hangers, one in Subsytem 1SD01 and one in Subsystem 1CV25. The weight of the trapeze in Subsystem 1CV25 is less than one foot equivalent run of pipe, and therefore would have a negligible effect on the piping.

The weight of the trapeze in Subsystem 1SD01 corresponds to approximately 3.5 feet of pipe. It is one support of a total of 31 on this subsystem. To demonstrate the effect of the trapeze weight, Subsystem 1SD01 was analyzed with the additional mass of the trapeze support. The anlaysis results show that all piping stresses remain within code allowables and that the changes in support loads result in interaction ratios less than code allowable stresses. The subsystem designs for the two cases where trapeze supports were found would be the same with or without trapeze weights in the piping model.

Based on the results of the sampling which indicate a very small number of trapeze supports are used and the results of the test analyses that show the inclusion of trapeze weight did not change design, we conclude that the criteria as presently followed is sufficient for the piping analysis.

We trust that this additional information is sufficient to resolve this item.

UNRESOLVED ITEM 3-5: FUNCTIONAL CAPABILITY CRITERIA

Unresolved Item 3-5 addresses the difference in functional capability criteria for NSSS and BOP piping, stating that Westinghouse is using criteria approved for NSSS piping for analysis of BOP piping.

RESPONSE

Westinghouse agrees with the IDI Team that this unresolved item does not represent any significant technical problem. Westinghouse disagrees with the IDI Team that this unresolved item is an instance of Westinghouse meeting the licensing commitment appropriate to the NSSS rather than BOP. The Westinghouse functional capability criteria has been approved by the NRC, as stated in the IDI report. It is an acceptable method for addressing the NRC concern on the issue of demonstrating Class 2 and 3 functional capability. By definition, the Westinghouse criteria are BOP criteria since the NSSS (the reactor coolant loop and primary equipment) contains only Class 1 piping. Furthermore, Westinghouse has been using this criteria as a design basis for all Class 2 and 3 stainless steel piping and not limiting it to a sampling process which would have satisfied the original NRC request.

To resolve this item, the FSAR will be amended to clarify the use of this criteria as defined in IDI Report Ref. 3.190 for Westinghouse scope of piping.

UNRESOLVED ITEM 3-6: FORCE AND MOMENT DIRECTIONS

The unresolved Item concerns the assurance of the correct application of the orientation of allowable nozzle loads received by Westinghouse from Sargent and Lundy for the spray additive tank (1CS01T).

RESPONSE

The procedures applied to verify coordinate orientation of equipment loads (as described below) and the specifics pertaining to the loads in the cited unresolved item (as described below) assure that the correct orientation is used in the equipment orientation.

The S&L coordinate system for allowable nozzle loads has been documented in S&L letter SLWC-5447, dated 12-1-83. For all S&L-supplied equipment, the actual loads are sent to S&L for final approval. These loads are transmitted in a well defined coordinate system. The coordinate system has been documented in Westinghouse letter CAW-6625, dated 11-14-83. The correct application of the proper coordinate system is assured by the final S&L verification.

The coordinate system of S&L-supplied allowable loads for the spray additive tank was not defined for the three (3) force (Fx, Fy, Fz) components and one moment (resultant moment, Mr) limits provided. (Note that the coordinate system definition is required for only forces since the resultant moment limit is independent of orientation.) The actual forces on the tank were shown in the reviewed calculation to be less than the minimum force component limit; in fact, the resultant of the actual forces is less than the minimum force component limit (see Table F3.6-1). Therefore, the S&L limits were met independent of the coordinate system. The coordinate system for the allowable loads for the spray additive tank is consistent with the coordinate system identified in SLWC-5447.

In the design process, the receipt of nozzle load allowables for the piping analysis is not mandatory and is not always available for all equipment. The evaluation of the nozzle loads from the piping analysis by Sargent & Lundy is the basis for establishing final acceptability of the equipment.

We believe this item is now resolved.

TABLE U3.6-1

SPRAY ADDITIVE TANK ALLOWABLE LOADS

ACTUAL LOADS		LIMITS
F_x = 130 lbs	316 lbs	F_1 = 3500 lbs
F_y = 270 lbs		F_2 = 850 lbs
F_z = 100 lbs		F_3 = 850 lbs
M_r = 5000 in-lbs		M_r = 25,000 in-lbs

UNRESOLVED ITEM 4-1: OPENINGS IN WALLS

There are openings in the shear walls along Column Lines 10 and 26 which appear to be substantial in size. It appeared that the designers had made a judgment that the openings were not sufficiently large so as to change the distribution of forces in these walls. However, we could find no evidence of how this judgment or other similar judgments were made. The licensee should determine whether the judgment was valid for this opening and, in general, the bases upon which similar judgments have been made.

RESPONSE

For openings in walls, all interrupted reinforcing bars are replaced at the edges of the opening, and additional reinforcing is placed around openings to reduce any load stress concentrations which could be present.

We agree that an engineering judgment was made that the openings in Column Lines 10 and 26 were not sufficiently large so as to change the distribution of forces in these walls. This same type of engineering judgment was used in the design of other shear walls in the plant. Also, there are cases where the original design engineer judged the redistribution of stresses around openings may have been significant. In these cases, such as along the base of the shear walls along Column Lines 6 and 30, an in-depth analysis of the redistribution of forces around large openings was made, and the reinforcing steel was located accordingly.

In order to document the above-noted engineering judgment, additional calculations will be performed to show that redistribution of forces is not a significant effect for various cases of wall openings.

UNRESOLVED ITEM 4-2: TOP REINFORCING FOR SLABS

In the design of slab 4AS53, the boundary condition where the slab framed into a wall was considered hinged, while the actual boundary conditions are such that a fixed support would be more appropriate for design. Negative moment steel equal to that at the continuous support was provided, and the potential problem was avoided since the designer supplied more than adequate reinforcing steel. However, the Team was concerned since it found no criteria addressing this situation and this could lead to a situation where insufficient reinforcement would be provided. In the Team's judgment the licensee should verify that adequate top reinforcement was provided for all non-continuous slab supports.

RESPONSE

As shown on Sargent and Lundy Drawings S-690 and S-790 (I.D.I. References 4.64 and 4.77), negative moment steel, equal to that at the continuous support, was provided at the junction to the wall of slab 4AS53. This negative steel is not required by design according to Calculation 7.43 (I.D.I Reference 4.78), but is provided as a standard detail on Sargent & Lundy drawings S-690 and S-790 to increase the factor of safety for slabs. The design conservatively assumes that the junction of the slab and wall is hinged.

By assuming a hinge with no moment capacity, the maximum possible positive moment in the center of the slab is considered. Steel is provided for this positive moment which is all that is required for a safe design. Any negative moment steel added at a wall or other noncontinuous support provides an additional factor of safety. We trust this additional explanation resolves this item.

UNRESOLVED ITEM 4-3: ANCHOR BOLT DIAMETER

In November 1981, a Field Change Request was initiated by Hatfield Electric Company (Reference 4.53) which stated that no mounting detail was available and proposed a location for the two sets of racks as well as a mounting detail which was identical to that used in the calculation except it utilized 1/2-inch diameter expansion anchors instead of 1/4-inch anchors. The drawing (Reference 4.52) was revised (Revision L) to provide for the use of 1/2-inch anchors on Unit 1 and 1/4-inch anchors on Unit 2. A field check by the team indicated that for Byron, 1/2-inch anchors had been used. Upon questioning, it was determined that the 1/2-inch diameter, 7-inch long expansion anchors had been substituted in Unit 1 because of the need to have an anchor with sufficient length to penetrate through a 3-inch thick topping floor into the base concrete. Currently, the drawing continues to call for 1/4-inch diameter anchor bolts for Unit 2. Since this need may also exist for Unit 2, the licensee should resolve what will be required on Unit 2 and make any necessary revisions to the drawing.

RESPONSE

The design calculations for this mounting detail show that a 1/4-inch diameter expansion anchor is adequate to transfer the design loads. Because the installation in question was made to a finish slab, a longer anchor was used as required by Form BY/BR/CEA. This form requires the contractor to set expansion anchors into rough concrete. Therefore, at the time of installation an anchor with the proper length was used. If a similar installation is made to a finish slab in Unit 2, the substitution of a longer anchor will also be made. However, for purposes of clarity, drawings will be revised where applicable to specify the longer anchor. We consider this item to be resolved.

UNRESOLVED ITEM 4-4: STARTING BATTERY SUPPORT RACKS CLEARANCE

This item concerns a field inspection of the Auxiliary Feedwater Diesel Starting Battery Rack, which identified a space between the end of the battery and the end of the rack. The Seismic Qualification Report for the batteries and racks had not yet been submitted, so no final resolution could be made as to whether the space was acceptable.

RESPONSE

Sargent & Lundy agreed at the time of the IDI to ensure that the final battery/rack installation is consistent with the arrangement used for its seismic qualification.

Subsequent to the IDI, the Seismic Qualification Report for the battery/rack was submitted. The report indicated a spacer installed between the battery and the end of the rack. Sargent & Lundy has contacted the vendor to provide revised installation drawings and additional mounting hardware to make the installation agree with the Seismic Qualification Report.

UNRESOLVED ITEM 4-5: AS-BUILT REVIEW

With the data developed from the execution of these procedures, Sargent & Lundy will undertake a program to review the structural design adequacy based on the as-built conditions. It is noted that at the present time Sargent & Lundy envisions a 100% recheck program on loads from all types of hangers except for loads less than the limits provided for in PI-BB-34. No separate procedure or instructon had been developed to address how the data being gathered are to be utilized and integrated to make the final as-built loading checks. Considering the nature of the work to be done, the licensee should assure that such guidance is developed in the near future to support completion of the checks.

RESPONSE

A separate procedure exists in Calculations 18.1.2.5 (Revision 0 dated 3-16-83) and 18.2.9.5 (Revision 0 dated 3-14-83) which was developed to address how the data being generated via PI-BB-34 is to be utilized and integrated to make the final as-built loading checks. This information was readily available at the time of the IDI but the IDI Team did not indicate that this was an item of concern. The procedures are included in the Design Control Summaries of each item being checked. There is no PI for final load check since once the loads are gathered according to PI-BB-34, there is no interdisciplinary work required. The final load check is performed solely by the Structural Department. We trust this additional information resolves this item.

UNRESOLVED ITEM 5-1: ANALYSIS CONCERNING LACK OF SEPARATION

This item concerns specific open questions described in Notes 1, 2 and 3, and in Samples A, D and E of the report. Note 1 concerns cable tray separation. Note 2 indicates that a cable touching the sidewall of a cable tray is a violation of Hatfield Procedure #10. Note 3 concerns touching of safety-related and non-safety-related conduits. Sample A concerns support of a tubing bundle in the remote shutdown panel. Sample D concerns safety-related and non-safety-related cables in close proximity in panel 1PA11J. Sample E concerns safety-related and non-safety-related cables in close proximity in Switchgear Bus 141 Cubicle 16.

RESPONSE

The response to this item is divided into six parts which address each of the open questions:

Note 1. The cable tray separations identified with Note 1 were identified on Cable Separation Criteria Violation Reports prior to the IDI. The IDI Team chose not to acknowledge the existence or specific resolution in cable separation conflict reports.

Note 2. IEEE Std-384 does not specify a separation distance from a cable-in-air to a cable in a raceway. In regard to Class 1E/-Non-Class 1E separation, Hatfield Procedure #10 requires a 12-inch distance from one cable-in-air to another cable-in-air. This distance does not apply for a cable-in-air to a cable in a raceway (e.g., cable tray). Hatfield Procedure #10 allows a cable-in-air to touch the sidewall of a cable tray.

Note 3. There are safety-related and non-safety-related conduits in the auxiliary feedwater tunnel that touch each other. There is in progress a program to address all potential instances of safety-related and non-safety-related conduits installed with less than 1-inch separation. The detailed procedures for this program are still being developed but will consist essentially of (a) an onsite physical walkdown of a statistical sample of all safety-related conduits, (b) an engineering analysis of all identified instances where the conduit separation is less than 1 inch, and (c) a modification of the conduit arrangement in any instance where the less than 1-inch separation cannot be justified by analysis. This program did not result from the IDI. The analysis will (where possible) show that the separation of less than 1-inch does not degrade the safety-related circuit. The detailed procedures for the walkdown will be completed by December 15, 1983.

Sample A. The tubing bundle has been reviewed by Westinghouse (Responsible for Seismic Qualification of the panel), and an additional support will be installed.

Sample D. All non-safety-related cables which could be in close proximity to safety-related cables in panel 1PA11J were identified in the Interface Review Report Index (i.e.: scheduled for analysis) at the time of the IDI. The design provides for the maximum cable separation attainable, considering the limitations imposed by the panel size and the terminal block arrangement; where the "required" separation can not be attained, the lesser separation is analyzed.

Sample E. It is noted that the Equipment Number "1AP95ED" in the IDI Report is in error. The correct equipment number is "1AP05ER". All non-safety-related cables which could be in close proximity to safety-related cables in Switchgear Bus 141 Cubicle 16 were identified in the Interface Review Report Index (i.e.: scheduled for analysis) at the time of the IDI.

UNRESOLVED ITEM 6-1: LOSS OF OFFSITE POWER TEST

Commonwealth Edison Byron Station personnel stated that a complete test of each automatic initiation signal was scheduled during the engineered safety features preoperational test. A number of preoperational test procedures (References 6.99, 6.100, and 6.101) and the loss of off-site power startup test procedure (Reference 6.102) were inspected at the Byron Station to determine how the engineered safety feature actuation system interface would be tested. The loss of off-site power test will automatically initiate auxiliary feedwater and numerous other engineered safety feature systems when circuit breakers are opened to simulate a loss of off-site power; however, Step 9.1.20 of this procedure only requires verification "that at least one train of the above (listed) equipment is running." This may have been an inadvertent omission from the test procedure acceptance criteria as the procedure was not yet signed. Since both trains are required to start automatically under these conditions, the licensee should ensure that the test procedure requires verification that both trains start during the test.

RESPONSE

The engineered safety features actuation system tests and various tests of individual safety systems will provide full, and, in some cases, redundant testing of individual automatic initiation signals. These will be done during preoperational testing. The loss of offsite power startup test does not need to reconfirm each of these signals. The requirement in Step 9.1.20 of this test is intended to ensure that adequate flow is maintained to the steam generators. It is not intended to duplicate the preoperational test program.

However, the Chapter 14 FSAR commitment for the loss of offsite power test involves verification of diesel generator starting, and load sequencing. Therefore, the test procedure should verify that the motor-driven auxiliary feedwater pump is properly sequenced onto the vital 4160V bus. The procedure will be modified to conform to FSAR commitments.

We believe that this response serves to resolve this matter.

UNRESOLVED ITEM 6-2: PRESSURE SWITCH QUALIFICATION

This unresolved item describes the current, incomplete status of the equipment qualification for the auxiliary feedwater low suction pressure switches and the fact that complete documentation was not available for the incomplete test.

RESPONSE

Within the context of the whole equipment qualification program, it is agreed that the pressure switch qualification is incomplete; however, we disagree with the inclusion of this particular component's qualification program as an unresolved IDI item. The status of the qualification program for the pressure switches was fully explained to the IDI Team. A brief discussion of the status of the pressure switch qualification follows:

Equipment qualification for the auxiliary feedwater pumps suction pressure switches was initiated in October 1982 for a previously specified instrument with a range of 0-100 psig. Subsequent to the beginning of this test (which includes aging), the pressure range of the switch was revised to 30 inches of Hg to 20 psig. The testing was allowed to proceed on the previously specified pressure switch model, with the intention of providing qualification of the current pressure switch model by similarity. If this cannot be done, a new test would be required. Since, at the time of the IDI, the test of the previous model was not complete, a final disposition for the current pressure switch model's qualification was not available as explained to the auditor. As of October 1983, the test report has been received and is being evaluated. Sargent & Lundy was fully aware of the status of the qualification program for the auxiliary feedwater pump suction pressure switches and will complete the program to meet the licensing requirements for the Byron/Braidwood Stations.

We consider this item to be resolved.

UNRESOLVED ITEM 6-3: CONTAINMENT ISOLATION REQUIREMENTS

The Team found that FSAR Table 3.2-1 did not reflect a number of safety-related containment isolation requirements that were currently reflected in the design criteria document of March 1983 (Reference 6.37). This did not represent a finding because it is not uncommon for FSAR changes to follow design changes by a few months and the team did not determine whether or not an FSAR change was planned. The licensee should assure that the inconsistency is corrected.

RESPONSE

The revisions to FSAR Table 3.2-1 to bring it into agreement with Revision 6 of CC-ME-01-BB ("Classification Criteria of Structures, Systems and Components"), dated 3/16/83 were included in FSAR Amendment 42. Amendment 42 is dated May 1983 and was transmitted to the NRC by letter dated 5/31/83.

Had the IDI Team checked at the time they conducted their IDI, they would have found that an FSAR amendment which incorporated the changes of Revision 6 of CC-ME-01-BB into FSAR Table 3.2-1 was in fact in preparation at that time and was issued for use to all FSAR holders shortly thereafter. The "inconsistency" has been corrected. This item has no safety significance and in fact is characteristic of the normal FSAR revision process.

We consider this item to be resolved.

UNRESOLVED ITEM 6-4: MANUAL START-STOP SWITCH

This finding concerns the location of local start/stop controls for the motor driven auxiliary feedwater pump. The controls are provided at the Remote Shutdown Panel. The Westinghouse functional requirements specify local manual start/stop controls local to the motor driven pump. The team did not determine the degree of need for the switch location recommended by Westinghouse or the licensee's reason for the actual switch location.

RESPONSE

We agree with statements in the Report that the design meets current regulatory requirements and that the licensee is not required to follow every Westinghouse recommendation. The Westinghouse recommendation was considered in determining the design location of the switch. Sargent & Lundy and Commonwealth Edison Company determined that the local start/stop controls should be located at the Remote Shutdown Panel and that a control switch local to the motor driven pump is not required. We consider this issue to be resolved.

Section III

Detailed Responses to Observations

OBSERVATION 2-1: CONDENSATE STORAGE TANK

This observation recommended that a sizing calculation on the condensate storage tank be prepared to preclude the possibility of switchover to the relatively impure essential service water source for events of moderate frequency.

RESPONSE

Proposed Commonwealth Edison Technical Specifications require verification every 12 hours that the condensate storage tanks are maintained such that a minimum of 200,000 gallons of water are available for the exclusive use of the auxiliary feedwater system. Each condensate storage tank holds 500,000 gallons of water which provides sufficient capacity for normal makeup and the minimum 200,000 gallons reserved for the auxiliary feedwater system. No additional calculations are required.

We consider this item closed.

OBSERVATION 4-1: SPREADING LIVE LOADS TO ADJACENT BAYS

Section 10.0 of Design Criteria DC-ST-03-B/B (Reference 4.31) contained some details beyond those specified in the FSAR on procedures and assumptions to be made in considering various load effects. The Team considered one item as constituting an unconservative assumption. This involved a statement in Section 10.2.1.1.3.1 regarding the input live loads from piping and cable tray loads within a given bay being spread onto two bays. However, since the actual loading was scheduled to be checked in the final load check, the statement would not lead to deficiencies. Accordingly, this was not considered a Finding or an Unresolved Item. It is mentioned as an item for licensee consideration with respect to revising the wording.

RESPONSE

The design method in the Design Criteria for cable tray and piping loads was apparently misinterpreted by the IDI Team. Loads shown are for all cable trays or pipes in a given direction in a given area. They are shown in one bay for simplicity, however, they represent loads in the entire area. Also, the final load check will reevaluate the slabs for the actual, as built, loading condition to verify the adequacy of the original of loading assumption. We trust this additional explanation resolves this matter.

OBSERVATION 4-2: WORDING OF CRITERIA

On page 11-6 of Section 11 of Design Criteria DC-ST-03-B/B (Reference 4.31), the radical sign in equation 4a exends over the last term. The terms of equations 4a and 4b are apparently total required steel area for shear and flexural while N_o/f is that steel area due to gravity load which can be deducted. It is not explained that the total areas for shear and gravity loads should be proportioned. In Section 29.0 on page 29-3, spans l_1 and l_2 are referred to as being defined by ACI 318-71 in which these spans are center- to-center spacing. The method of analysis is specified as being from ACI 318-63 which defines clear spans rather than center-to-center-spans. The above items are editorial in nature and are not considered findings or unresolved items. They are mentioned as items for which we recommend that the licensee consider clarifying the wording.

RESPONSE

A. The radical sign in Equation 4a which extends over the last term is a typographical error. The proper equation was used in the design because the shear wall Design Control Summary and shear wall design in Calculation 7.12.6 use the correct formula. The design criteria, page 11-5 will be revised to show the correct Equation 4a.

B. The definition of N_o/f will be clarified in the design criteria to indicate that the total dead load is proportioned linearly to the respective sections in the wall (flange and web). The shear wall design was performed using the correct method.

C. The use of l_1 and l_2 in the design of slabs was used as a selection criteria to define the analysis method to be used. Once the criteria was established, the applicable values for length were used depending on the criteria selected.

We trust this additional information resolves this item.

OBSERVATION 4-3: EDITING POSITION DESCRIPTION MANUAL

The Sargent & Lundy Position Description Manual provides a job description for each job classification. Position descriptions were reviewed from the department manager down into five of the nine divisions of the department. In general, it was observed that Sargent & Lundy places high educational and experience requirements on engineering positions and that there is a graded increase in those requirements as the job responsibility increases. Several items listed below were noted during the review.

(1) None of the positions utilize professional engineering registration as a requirement.

(2) The department manager is not required to have more experience than three of the key division heads.

(3) The Structural Design Director, who is responsible for four divisions is not required to have experience equivalent to three of the key division heads not under his direction.

(4) The heads of two of the divisions under the Structural Design Director do not have equivalent experience requirements as three of the other key division heads.

(5) A supervisor in the Analytical Division, and a Supervisory Structural Engineering Specialist in the Specialist Division are not required to have supervisory experience.

These items are observations and do not represent Findings or Unresolved Items, but are mentioned as items that Sargent & Lundy may wish to consider with respect to editing its Manual.

RESPONSE

(1) Although the requirement for professional engineer registration is not explicitly stated in the Position Description Manual, the Department Manager and Assistant Department Manager do have professional registration. The position descriptions for the Department Manager and Assistant Department Manager will be revised to indicate that professional registration is required.

(2) The experience requirement of 10 years is an appropriate limit for requiring a minimum experience level. Beyond that, other qualities and capabilities which are not based on years of experience become more important. Therefore, these positions all have the same limit.

(3) The Structural Design Director position description has design experience equivalent to or more than that of the Engineering Division Heads he is responsible for. The other two Divisions (Drafting and Architectural Design) do not require a college degree and therefore require stronger emphasis on years of experience rather than college education, due to the nature of their responsibilities.

(4) Due to the different responsibilities between these Divisions, it is felt it is not necessarily appropriate to have equivalent experience.

(5) The requirement for a Supervisor in the Analytical Division and the Engineering Specialist Division does not include supervisor experience because this could be the first supervisor's position this person is holding.

OBSERVATION 4-4: HANGER LISTS AND LOAD SUMMARIES

Instruction PI-BB-34 places the responsibility of preparing and updating hanger location drawings, hanger load summaries and the mechanical hanger lists on the Structural Department for the project. Based on the team's judgment, the project instruction as written is believed to create the potential for errors and omissions since the Mechanical Department personnel would generally be more familiar with the piping hangers they designed than would the personnel from the Structural Department. This is not a finding or an unresolved item but an item which the licensee may wish to consider.

RESPONSE

The present system for preparing and updating hanger load summaries is well established, is in accordance with existing procedures, and in our judgment, the most suitable way to perform the tasks. The hanger drawing contains all load and location information required for design and installation. This information constitutes the information required by the Structural Department for the load checks of the plant structures. The Mechanical Department generates an up-to-date listing of every hanger and its current revision, which is revised and issued on approximately a monthly basis (Mechanical Hanger List).

The Hanger Load Drawing (HLD) is a consolidation of load location information controlled by the Structural Department. It includes loads from HVAC, cable tray, conduit, junction box, Westinghouse, NPS, and mechanical hanger loads.

The preparation of the HLD and hanger load summary (HLS) is an interim step in the overall structural design process. Familiarity with the structure is the essential element in formulating the HLD and HLS since both are structural design tools. Production of an HLD and HLS require certain judgments that the Structural Department is qualified to make. For these reasons, we believe the actual transfer of load and location information is best handled within Sargent & Lundy through the hanger drawings and drawing list. We consider this item to be closed.

OBSERVATION 4-5: HANGER LOAD SUMMARY INSTRUCTION

In the definition of the hanger load summary in Section 2.2 of instruction PI-BB-34, it appeared that the intent is to also include those component supports whose load is applied to concrete slabs, walls, and framing. However, this was omitted from the instruction. This is not a finding or an unresolved Item but is mentioned as an item recommended for licensee consideration in structuring the procedures.

RESPONSE

Instruction PI-BB-34 (I.D.I. Reference 1.31) does include component supports whose load is applied to concrete slabs, walls, and framing. It is only the Hanger Load Summary (HLS) of Section 2.2 which applies only to steel members. The HLS form is not required in the final load check of slabs and walls. Section 4.1 of PI-BB-34 states that loads attached to slabs, concrete beams, walls, and steel are required. Also, Table I Indicates what information is required for each of these items; thus, nothing has been omitted from the instruction. Once the loads have been received by the Structural Department, the method of summarizing loads, other than those attached to steel, for the final load check is explained in the Design Control Summary for final load check of slabs, concrete beams, and walls. Therefore, we do not consider a procedural change to be necessary.

OBSERVATION 4-6: FLOW DIAGRAMS

The team noted that flow diagrams would improve instructions PI-BB-34, 35 and 37. This is not a Finding or an Unresolved Item, but is mentioned as an item recommended for licensee consideration from the standpoint of improving procedural guidance and control.

RESPONSE

PI-BB-34 will be updated such that under Section 6, "Flow Chart", the word "none" will be replaced by "none required." A flow chart is not required in this PI since all flow of information is a simple, one-step process. Various parties send the information outlined in the PI to the Structural Department. A flow chart would be simplistic and of no value.

For PI-BB-35 and PI-BB-37, a flow chart will be added at a future revision to further clarify these instructions. After this is completed, this item will be closed.

OBSERVATION 5-1: PHYSICAL SEPARATION

As noted in Section 7.7.2.2, there were instances where there was sufficient room to provide separation inside of cabinets rather than using analyses to justify the lack of separation. It appeared, however, that virtually all instances may be resolved with analyses to justify the existing conditions. This does not represent a finding or an unresolved item because such analyses, if they are adequate, satisfy the licensing commitment and accepted industry standards. Accordingly, the matter is mentioned as an item recommended for licensee consideration.

RESPONSE

It is Edison's position that two basic criteria must be satisfied relative to separation inside cabinets. The first involves satisfying licensing commitments. The second involves satisfying Edison's own concepts of safety, reliability and maintainability. In specific instances, analyses are employed to satisfy both basic criteria as opposed to literal application of separation distances. Edison finds either approach acceptable.

OBSERVATION 6-1: EQUIPMENT STATUS DISPLAY POWER SUPPLIES

This observation concerns the design of power supplies for the ESD System. The power supplies are provided in accordance with the design criteria. The Team observed that both power supplies are connected to the same division rather than from different divisions.

RESPONSE

The ESD System is classified non-safety-related. Although the Byron design has two non-safety-related divisions, the physical location of the Equipment Status Display panel in the control room requires that cable entry be from the bottom, and only one division is located below the control room. Both power supplies are taken from the same division because all cable in the lower cable spreading room is associated with one division.

The use of two power supplies from the same division provides more than very limited benefit as stated in the Report. There are multiple sources of power to the 4-kV bus (including multiple system auxiliary transformer and access to the diesel generator) that ultimately feed both power supplies. The use of two power trains greatly reduces the possibility of loss of power due to a single component failure.

OBSERVATION 6-2: EQUIPMENT STATUS DISPLAY QUALITY STANDARDS

This observation recommends that additional emphasis be given to the documentation of the design for the Equipment Status Display System.

RESPONSE

The Equipment Status Display System is non-safety related, Category II. However, we will, as part of normal power plant design practice, provide a complete and formal documentation package.

The documentation package for the Equipment Status Display design will include both the hardware and software aspects of the system.

OBSERVATION 6-3: PUMP INSTRUMENTATION LOCATION

This observation recommends that the as-built installation of the auxiliary feedwater pump suction pressure switches be reviewed for ease of calibration and maintainability.

RESPONSE

Sargent & Lundy has reviewed the concerns of the IDI Team with the Byron Station Project Construction Department Lead Instrument Engineer and the Byron Station Instrument Foreman. After inspection of the instrumentation associated with both the auxiliary feedwater pumps 1A and 1B, it has been determined that the instruments are accessible for maintainance and calibration. We consider this issue closed.