U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

Division of Quality Assurance, Vendor, and Technical Training Center Programs

Quality Assurance Branch

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Facility Names:	Byron Generating Station Unit 2 Braidwood Generating Station Units 1&2
Inspection At:	Sargent & Lundy Engineers Chicago, Illinois
Inspection Conducted:	November 18 - 22, 1985
Inspection Team Members:	
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BYRON UNIT 1 AND BRAIDWOOD 1&2
DESIGN VERIFICATION ACTIVITIES
IMPLEMENTATION INSPECTION
NOVEMBER 18 THROUGH 22, 1985

I. BACKGROUND

D. Eisenhut's letter of January 14, 1985 to C. Reed of Commonwealth Edison Company (CECo) discussed previous design verification activities related to work performed by Sargent & Lundy Engineers (S&L), the architect engineer for Byron 2 and Braidwood 1&2. The Staff indicated, in particular, that the applicable findings from the Byron 1 Integrated Design Inspection (IDI) and the Independent Design Reviews (IDRs) of Byron 1 and Clinton 1 should be addressed for Byron 2 and Braidwood 1&2. A public meeting was held at the NRC offices in Bethesda, Maryland on April 10, 1985. During that meeting, CECo and S&L presented to the Staff the applicable findings from the previous IDI and IDRs and the methodology used to address them and develop any necessary corrective actions. Subsequently, on April 24, 1985, the applicant submitted a letter to the Staff (C. Reed to H. Denton) indicating the steps it was taking for corrective actions to remedy these applicable findings from the previous IDI and IDRs. The Staff concluded in a letter (H. L. Thompson to C. Reed, dated April 29, 1985), that the design verification activities performed should enable the Staff to arrive at a conclusion concerning the design process for Byron 2 and Braidwood 1&2 without the performance of additional third party design verification activities. An implementation inspection was performed by the Staff to assess the effectiveness of the implementation of the appropriate corrective actions. The result of this inspection is the subject of this report.

II. PURPOSE

The purpose of this inspection was to inspect implementation of applicable corrective actions on Byron 2 and Braidwood 1&2 from the the Byron 1 IDI and IDRs of Byron 1 and Clinton 1 to provide additional assurance that the design process used on Byron 2 and Braidwood 1&2 produced a plant design which meets FSAR commitments and NRC regulations.

III. NRC INSPECTION TEAM

The inspection was conducted by NRC personnel with support of contractor personnel as follows:

Assignment	Name, Position
Team Leader	E. Imbro, Section Chief, IE
Mechanical Systems	S. Klein, Consultant, WESTEC Services
Mechanical Components	J. Blackman, Consultant, WESTEC Services
	H. Wang, Inspection Specialist, IE
Elec. and I & C	S. Athavale, Electrical Engineer, IE
Civil/Structural	G. Harstead, Consultant, Harstead Eng.
	H. Wang, Inspection Specialist, IE
Quality Assurance	W. Belke, Quality Assurance Engineer, IE

IV. PERSONNEL CONTACTED

A large number of CECo and S&L personnel were contacted during the inspection. The following is a brief list of key personnel contacted during the inspection.

Name	Company, Title
H.S. Taylor	S&L, Manager, Quality Assurance Division
D.L. Leone	S&L, Project Director
B.R. Shelton	CECo, Project Eng. Manager
W.G. Cleff	S&L, Project Manager
B.A. Erler	S&L, Structural Design Director
P. Wattelet	S&L, Project Manager Director
K.T. Kostal	S&L, Braidwood Project Director
K.J. Hansing	CECo, Director of Quality Assurance
G.C. Jones	S&L, Braidwood Project Manager
A.D. Miosi	CECo, Nuclear Licensing Adm.

In addition, the NRC team members contacted the S&L lead engineers in the individual technical disciplines. The S&L lead engineers arranged for contact with the specific S&L engineers and designers at the request of the NRC team member.

V. INSPECTION SUMMARY AND CONCLUSIONS

During the inspection, the NRC team reviewed the background documentation and implementation of corrective actions stemming from the Byron 1 and Clinton 1 IDRs and the Byron 1 IDI findings to ensure that they had been adequately addressed at Byron 2 as well as Braidwood 1&2, where applicable. Generally, it was determined that the corrective actions had been implemented in a manner consistent with the intent of the original finding. Not all corrective actions had been completed at the time of the inspection, since they are attributes of ongoing review programs. Items remaining open due to the ongoing nature of review programs will be verified and closed, upon the completion of these programs, by NRC Region III. The NRC team did review corrective actions that had been completed and determined that they were being implemented adequately. Consequently, it was concluded that completion of the ongoing review programs would assure adequate implementation of appropriate corrective actions.

The following are specific comments from each of the disciplines that participated in this inspection:

1. <u>Quality Assurance</u>.

The information provided in the response to Item 6 of the January 14, 1985 letter from Darrell Eisenhut to Cordell Reed was reviewed. Item 6 of this letter requested a description of the aspects of the QA program related to design which assure that the applicable design commitments were implemented at Byron 2 and Braidwood 1&2. To verify the information provided in the response, the programmatic aspects of the QA program implementation related to design were reviewed. This included a review of S&L and CECo QA manuals, audit procedures, audit reports, audit checklists, and QA program descriptions. Sufficient evidence was provided to indicate that the programmatic aspects of the CECo and S&L QA design commitments are being applied in accordance with 10 CFR 50 Appendix B and other FSAR commitments.

2. Mechanical Systems

Many of the findings in the mechanical systems area are directly applicable to Byron 2 and Braidwood 1&2 as they are related to specific technical issues which are independent of the type of NSSS supply. Several, involving FSAR revisions, were basically editorial in nature. The NRC team inspected documentation which supported corrective actions for the following findings:

2.1 Byron 1 IDR Observation Report 8.38 - CCW Design Pressure

This OR concerned a calculated peak pressure of 268.5 psig in an S&L piping calculation sheet, CC-VP-01, which exceeds the design pressure of 150 psig. As a result of this IDR Observation, Westinghouse determined that design pressures could be exceeded if a postulated letdown heat exchanger tube rupture resulted in pressurizing the surge tank to the relief valve setpoint. Westinghouse reported this condition to the NRC. The problem was resolved for Byron 1 by removing the relief valve and operating the surge tank at atmospheric pressure.

The Team inspected the documentation substantiating that the design change described for Byron 1 being implemented for Byron 2 and Braidwood 1&2.

2.2 Clinton IDR Observation Report 79 - Pump, Valve Testing vs. FSAR

This observation concerned inconsistencies between FSAR commitments to require shop testing, e.g., for valve seat leakage, and the lack of requirements for such testing in procurement specifications. S&L had verified that shop functional testing was performed by the vendor on the listed Essential Service Water (ESW) active valves and the other active valves listed in the FSAR. However, the response to this finding did not clearly indicate that the procurement specification had been revised to reflect the requirements for shop testing of valve seat leakage at design pressure. The Team was concerned that replacement valves purchased to these specifications in the future might not be tested as required.

The Team was advised that the procurement valve specification invokes MSS-SP-61, which is a standard valve specification. This in turn requires in shop valve seat leakage testing to be performed at 1.1 times design pressure corresponding to 100 degrees F. The valve procurement specification is applicable to all Byron and Braidwood units and therefore replacement valves would also be leak tested.

2.3 Byron 1 IDI Finding 2.2 - Flow Calculation Procedural Items

The Byron IDI identified several deficiencies in flow calculation AFJK-1 which were inconsistent with the requirements of S&L Procedure GQ-3.08, Revision 4. However, Revision 3 of this procedure was the procedure in effect at the time AFJK-1 was prepared and did not require documentation of assumptions or input data that must be verified. The team confirmed that the calculation was revised to be in accordance with Revision 4 of GQ-3.08 and that the calculation is applicable to Byron 2 as well as Braidwood 1&2.

2.4 Byron 1 IDI Finding 2.6 - Net Positive Suction Head (NPSH) for Auxiliary Feedwater (AFW) Pumps

The Byron 1 IDI determined that S&L Calculation AFJD-1 did not sufficiently document the basis for determining that adequate NPSH was available to the AFW pumps and violated Procedure GQ-3.08. A new calculation AFTH-01 which superseded Calculation AFJD-1 provided documented evidence that adequate NPSH was available to the AFW pumps. This calculation references Byron 1 pipe routing but is applicable to Byron 2 and Braidwood 1&2. The Team was concerned that the calculation did not document the effect of pipe routing differences between Byron 1 and the other units. During the inspection, S&L performed a calculation using the differences in pipe routing between Byron 1 and the Byron 2/Braidwood units for a typical AFW discharge flow path. The calculations demonstrated that these differences would result in a difference in discharge flow of the order of 0.5 gpm and could be neglected. Based on these calculations and the similarity of the suction piping between units, the Team agreed with S&L that these differences would have no effect on calculated results.

The Team also inspected completed documentation related to high and moderate energy line breaks (HELB/MELB). The Team found that break locations were identified, and differences in locations and pipe routing between Byron 1 and Byron 2/Braidwood were adequately documented. Also reviewed were S&L procedures PI-BB-38 and PI-BB-91 which were developed to define the division of responsibilities and channels of communication between the various technical disciplines involved in the HELB/MELB analysis. The Team's review of these procedural documents provided additional confidence that S&L had a systematic method for addressing the consequences of postulated high and moderate energy line breaks. However, the Team did note a distinct lack of documentation demonstrating the plant's capability to achieve safe shutdown following a postulated pipe break. In order to determine the consequences on a given piece of equipment required for safe shutdown or break mitigation, it is necessary to refer back to basic drawings specifying equipment and break locations and then reassess any potential interaction. In response to similar questions raised by IE during the Byron 1 IDI, Commonwealth Edison Company (CECo) submitted a document, "Byron 1, Confirmation of Design Adequacy for Jet Impingement Effects," to the NRC. It is our understanding that the Office of Nuclear Reactor Regulation (NRR) has requested the Applicant to provide further documentation demonstrating the capability to achieve safe shutdown following a postulated pipe break. Since the issue of HELB/MELB will be resolved between the Applicant and the NRC staff as part of the normal licensing process, IE plans no further action and considered this subject closed for the purposes of this inspection.

3. <u>Mechanical Components</u>

Many of the findings in the mechanical components area were directly applicable to Byron 2 and Braidwood 1 and 2 as they were related to specific technical issues which were independent of the type of NSSS supply. The team inspected documentation which supported corrective actions for the following findings:

3.1 Byron 1 IDI Finding 3.2 - System Functionality Requirements

This finding addressed the need to specify which systems must be evaluated for functionality requirements and the criteria used by S&L during the analysis of the piping. As a result of this finding, S&L did evaluate which system must be checked for functionality requirements and clarified the methodology used to evaluate the piping. The Team inspected documentation which substantiated the corrective action described above as well as several piping analysis calculations to verify that it had been implemented properly. Based upon this review, the Team concluded that the implementation had been satisfactorily performed on Byron 2 and Braidwood 1&2.

3.2 Byron 1 IDI Unresolved Item 3.2 - The Effects of Lateral Vibration on Struts and Rods

This unresolved item concerned the effect of lateral vibration on the axial load capacity of rods and struts. The general requirement in considering this phenomenon was contained in S&L's Engineering Mechanics Division, "Lesson Plan for Training Personnel in Piping Analysis," EMD TP-1, Revision 3. S&L performed an evaluation of the affected hardware for Byron 2 and Braidwood 1&2 and determined that few components were affected. In the cases where a reduction in load capacity was required, project documentation was revised accordingly. The Team reviewed all aspects of this item and concluded that it had been adequately implemented for all three plants.

3.3 Byron 1 IDR Observation Report 8.29 - ASME Stress Criteria for Non-Pressure Retaining Parts

This observation report concerned a lack of stress criteria for non-pressure retaining parts of passive valves. In the Byron I IDR, three (3) passive valve configurations were found to be evaluated with incorrect criteria. The criteria used was for active valves, which is more conservative. Bechtel, however, was concerned that the lack of specific criteria might lead to a failure to properly evaluate stresses to the correct criteria. Therefore, review of calculated stresses against higher allowables could fail to detect an overstressed condition and degrade the safety-related function of the valve. To preclude such a situation, S&L did an equipment qualification review for vendor reports in which the equipment was evaluated to assure that it conformed to all required specifications, which include the correct criteria. The Team reviewed the implementation of vendor report reviews for Byron 2 and Braidwood 1&2. The team concluded that all the necessary technical items were correctly addressed and that the valves met the design specifications.

3.4 Clinton IDR Observation Report 20 - Allowable Deflection for Pipe Supports

This observation concerned the calculation of allowable deflection for auxiliary steel pipe supports, which is limited to 0.25 inches. Based on commitments made to the NRC, this calculation was to consider dead weight, thermal, self-weight excitation, and dynamic as well as seismic excitation loadings. However, it was determined that S&L was only considering deflections due to seismic loads. To remedy this situation, S&L revised Structural Design Standard E37.0 (Revision 4, dated April 15, 1985) to require consideration of all design loadings, and performed a review of the applicable Byron/Braidwood calculations to ensure that all applicable loadings had been considered. The Team inspected the documentation substantiating the actions discussed above and concluded that it had been properly implemented for Byron 2 and Braidwood 1&2.

3.5 Clinton IDR Observation Report 25 - Extended Valve Mass Flexibility Effects

This observation report identified a potentially unconservative modeling technique used to predict valve acceleration levels in piping analysis. The extended valve mass was represented as a more rigid assembly than its actual behavior indicates. To account for the additional resulting amplification in flexible valves, a factor of 1.5 was applied to the valve acceleration levels. This factor was substantiated based on a representative study of typical valves. The amplified acceleration levels were compared to allowables and found to be acceptable. The Team reviewed the substantiating documentation regarding this observation report and concluded that it had been properly implemented for Byron 2 and Braidwood 1&2.

4. Civil/Structural

The Byron 1 IDI and IDR dealt, in general, with generic issues such as Structural Design Criteria, Project Instructions, Seismic Analysis Calculations, Details of Vendor Supplied Equipment, Anchorage Details, and the Load Check Program. All of these were common to Byron 2 and Braidwood 1&2. Except for some isolated errors in the design of pipe supports in the Byron 1 IDI and concerns in the soil structure interaction analysis of the Byron river screen house, the resolutions and revisions to documents are applicable to all units.

In contrast, the Clinton IDR dealt with plant specific issues not in common with Byron/Braidwood, and, therfore, not directly applicable to Byron/Braidwood.

4.1 Byron 1 IDI Finding 4.1 and 4.2 - Ambiguities in the Structural Design Criteria

These findings addressed apparent ambiguities in the Structural Design Criteria, DL-ST-03-B/B. Specifically, horizontal seismic inertia loads, wind loads, and tornado created differential pressures did not appear to have been considered in the design of walls. Also shear friction was inappropriately used to calculate reinforcement requirements for transverse shear. These findings were resolved by revisions to the criteria, and a review of the revised calculations indicated that the calculations were performed correctly. The Team verified that these design approaches were implemented in an identical fashion on Byron 2 and Braidwood 1&2. Hence, it was concluded that the resolution of these findings had been satisfactorily implemented.

4.2 1 IDI Unresolved Items 4.1 and 4.2 - Required Substantiation for Reinforcement Design

These unresolved items concerned judgments made regarding the development of local reinforcement requirements at openings and at the tops of non-continuous slab supports. The items were resolved by developing additional calculations needed for reinforcement design and by clarifying how judgments were substantiated. The Team reviewed the documentation regarding these items to confirm that they were directly applicable to Byron 2 and Braidwood 1&2 and concluded that the matter had been satisfactorily resolved.

4.3 Byron 1 IDR Observation Report 8.5 - Reevaluation of River Screen House

This observation report questioned whether or not the reinforced concrete portions of the river screen house had been properly reevaluated as a result of additional licensing considerations related to seismic analysis for soil structure interaction affects. S&L prepared additional calculations which confirmed the adequacy of the original judgments made regarding the reinforced concrete portions of the building. Since the river screen house is common to both Byron 1&2, the corrective action is applicable to both. It is not applicable to Braidwood 1&2 because soil structure interaction was not an issue at these units. The Team reviewed the substantiating documentation and concluded that the matter had been satisfactorily resolved.

4.4 Byron IDR Observation Report 8.36 - Allowable Capacity of Expansion Anchors

This observation report dealt with the treatment of allowable capacity of expansion anchors due to the relocation of the bolts. It appeared that the reduction in capacity due to the relocation was not considered in S&L document SDS-E11, Rev. 0, entitled "Concrete Embedments and Expansion Anchors." Paragraph E.1(c) in fact permitted a 28% increase in the load capacity. The justification for allowing the increase was that the published bolt capacity had already been reduced 35% due to bolt angularity and attachment location. Further, the design standard was developed on the basis of 3500 psi concrete, while the average concrete strength at Byron 1 was 5265 psi, and if an elliptical shear-tension interaction equation is used, increases of 23% and 18%, respectively, in bolt allowables would result. These additional factors more than account for the 28% increase in allowable capacity. The Team reviewed the documentation and concurred with the approach utilized. Since the same standard was invoked at Byron 2 and Braidwood 1&2, the Team confirmed that resolution of similar problems had been treated identically.

4.5 Clinton IDR Observation Report 61 - Seismic Stability of Structures

This observation report addressed a concern that the FSAR commitment of structural stability design was not implemented properly. The FSAR specifies a foundation stability check for overturning, sliding and flotation, which include load combinations with OBE, SSE, wind, tornado and probable maximum flood. The structural stability calculation for the Clinton river water screen house foundation did not consider all load combinations mentioned above. S&L performed additional calculations to show that the load combinations did not reduce the required factors of safety below acceptable limits. The Team reviewed the basis for resolving the observation and concurs with the resolution. In addition, the team reviewed comparable areas in Byron 1&2 and Braidwood 1&2 and found no other such discrepancies.

5. Electrical Power

5.1 Byron 1 IDI Finding 5.1 - Analysis Concerning Lack of Separation

This finding identified many deficiencies related to lack of separation. The following are a few selected situations that are applicable to all Byron and Braidwood units. The NRC team found that most of the separation analyses had not yet been performed. However, the Team reviewed the appropriate procedures for the analysis and available representative calculations. In all cases, the Team concluded that the S&L calculational procedures were acceptable as were the specific calculations reviewed. S&L stated that these analyses would be completed prior to fuel load as part of the normal design process. Based on our review, Items 1, 2, 4 and 5 below are considered closed. IE has requested RIII to follow-up and close out Items 3 and 6 prior to fuel load as IE was unable to review any of the analyses in these areas since none were available during our inspection.

1. For non-class 1E cables which are in close physical proximity to class 1E cables, S&L had generated a file 4391/27 containing a list of such cables for each of the Byron and Braidwood units. S&L indicated that each cable will be analyzed to ensure that the class 1E cable(s) will not be damaged. The Team reviewed generic analysis/calculation procedure 4391/Q-4, which was used for the above mentioned cable analyses, and found that the attributes considered for such analyses were: 1. energy level (voltage, current transients), 2. physical separation, 3. electrical separation, 4. insulation, 5. environmental conditions and, 6. fire retardant characteristics. The Team reviewed two sample calculations, 2FW020-1 and 2FW314-1, and found that the attributes used, and the methodology and results were acceptable.

2. For the situation where a cable in air is in contact with the class 1E or non 1E cable trays, the Team found that S&L had addressed this concern in its analysis 4391/19Q-14. This analysis was based on a Wyle Laboratory Report 17769-1 that pertains to cables tested under similar conditions. The Team concurs with the analytical approach using the Wyle Lab test report.

3. For the case where conduits of different classes are in contact with each other, S&L generated a Project Instruction (PI), PI-BB-53, Rev. 1, providing guidance to perform walkdowns for a random sample of conduit systems to identify such situations. The PI also provides guidance to perform analyses to determine if the safety-related cables are degraded by any fault in the nonsafety-related cables. The Team agrees with the methodology provided in the PI. RIII will close this item on completion of the walkdown and resolution of any deficiencies detected during the walkdown (control nos. 455/85-047-01; 456/85-060-01; 457/85-056-01).

4. In cases where non-class 1E loads are connected to a class 1E power supply, S&L generated a generic interface review report (IRR) #2LL044-1, Rev. 1 that requires two qualified circuit breakers, in series, or a combination of a circuit breaker and fuse in series and an analysis to determine the trip setting coordination. The Team finds this approach is acceptable.

5. For the two cases where the distance between the bottom of the top tray and the top rail of the bottom tray (carrying either power or control cables) violated separation criteria, S&L indicated that their solutions were as follows:

The first case involved cables 11336BC1E & 11336GC1B shown on drawing 6/20E-0-3051. The team found that S&L had performed an analysis as required by cable separation criteria violation (CSCV) #168 and resolved the issue by covering the bottom tray with a solid metal cover for the entire length of separation violation plus three feet on each side. The Team finds this solution acceptable.

The second case involved cables 11715K2B & 117711JC2E shown on drawing 6/20E-0-3062. The Team found that S&L had performed analysis CSCV #521 which determined the installed configuration to be acceptable for the following reasons:

a. The cables are qualified to IEEE-383 and therefore will not propagate fire.

b. All cables are control cables in this area. No trays carry power cables.

c. The nonsafety-related cables are seismically supported in seismic category I areas and are segregated into two divisions similar to safety-related circuits.

d. The protective devices used for nonsafety-related circuits are the same as used for the safety-related circuits. These devices were purchased under the same specification (Westinghouse Specification #F/L-2755).

e. All instrument and control (I&C) cable trays are covered with solid metal covers.

The Team found results of analysis CVCS #521 acceptable.

6. For situations where actual cable lengths (pulled lengths) are different than those used for voltage drop calculations, the NRC team found that S&L had generated an Electrical Department Instruction ESI-253, dated 8/6/85 for reviewing the final calculations using the as built cable lengths as noted on the cable pull cards of the contractor. The Team finds the newly generated ESI-254 provides a reasonable solution. However, this item will be considered open pending the completion of implementation prior to fuel load. Region III will close this item when installation is completed. (control nos. 455/85-047-02; 456/85-060-02; 457/85-056-02).

5.2 Byron 1 IDI Finding 5.2 - Fault Current Calculations

Fault current calculation 19AQ-17 did not provide a reference for the 14000 amp available short circuit current. This finding is applicable to all Byron and Braidwood units. The Team reviewed Rev. 1 of calculation 19AQ-17 and found calculation AJ-1 Rev. 1 listed as the reference for the 14000 amps available short circuit current. The Team finds this resolution acceptable.

5.3 Byron 1 IDI Finding 5.3 - Heat Dissipation Calculation

Heat dissipation calculation 19AQ-18, which is applicable to all Byron and Braidwood units had three procedural discrepancies. The team found these had been resolved by S&L in the following manner:

a. Revision 1 of calculation 19AQ-18 had the page number discrepancies corrected.

b. Revision 1 of calculation 19AQ-18 and the QA Manual GQ-3.08 Rev. 5 Section 3.4.1 includes review methodology for calculations.

c. The original calculation used the assumed value of motor efficiency as 95% instead of the actual value of 94% shown in vendor supplied data. S&L indicated that the calculation will be revised using actual data from the vendor catalog during the final calculation check. Region III will verify the corrective action that has been taken and close this item at the completion of the final calculation check (control nos. 455/85-047-03; 456/85-060-03; 457/85-056-03).

5.4 Byron 1 IDI Finding 5.4 - Alarm Light Logic Diagram

This finding addressed a discrepancy on Drawing M-4037-1AF03 for Byron 1. The discrepancy was that the flow setpoint was 100 gpm rather than the correct value of 160 gpm. S&L indicated that this discrepancy had been corrected for all Byron and Braidwood units. The Team reviewed the following drawings and found that the discrepancy had been corrected.

Byron 2	-	DWG # M4122-2AF03 Rev. C
Braidwood 1	-	DWG # M4035-1AF03 Rev. D
Braidwood 2	-	DWG # M4122-2AF03 Rev. C

5.5 Byron 1 IDI Finding 4.9 - Starting Battery Support Rack Drawing 6E-0-3391C Rev. AC

This finding identified an error on the Byron 1 battery rack and anchorage detail drawing 6E-0-3391C Rev. AC. S&L indicated that the error had been corrected on all the Byron and Braidwood units. The Team reviewed the following drawings and concurred with S&L.

Byron 2	-	DWG # 6E-0-3391C Rev. AE
Braidwood 1	-	DWG # 20E-0-3391C Rev. AT
Braidwood 2	-	DWG # 20E-0-3391C Rev. AT

5.6 Clinton IDR Observation Report 60 - Lack of Voltage Drop Calculations

This OR is applicable to all Byron and Braidwood units. It concerns the operation of ESF loads and motor operated valves (MOV) during block start of the ESF loads. This OR also addresses timing for operation of the MOV overloads. S&L indicated that all required voltage drop/voltage profile calculations had been completed. The Team reviewed calculations AU-7 & AU-8 and found that during starting/accelerating period of the 4KV ESF loads, the 4KV system voltage dropped to 84% of the rated voltage and the voltage at the 4KV motor terminals was 82-83%. S&L stated that these 4KV motors were capable of developing sufficient torque at a minimum terminal voltage of 80% to accelerate the ESF load in the required time. The Team noticed from the calculations that the time taken by the ESF loads to accelerate was two seconds and during this period the MOV motors are stalled and experience locked rotor current (LRC). Once the ESF loads are accelerated, the 4KV bus voltage rises to 99% of its normal value, which translates to a voltage equivalent to 78.8% of normal voltage (480v) at the motor terminal of MOVs. This starting terminal voltage for the MOVs is below the minimum required (80% specified by Westinghouse) terminal voltage for developing the full design rated torque. S&L indicated that torque requirements to operate the globe and butterfly valves within the design required time were 69% and 62%, respectively. The Team found that at 78.8% terminal voltage, the starting torque developed by the motor was 97% of the design rated value which was higher than the actual required torque; therefore, voltage of 78.8% at the MOV motor terminals is acceptable.

The MOVs start moving after ESF loads are accelerated and completed their travels (open or close) in three seconds. Since the rotor current continues to fall after the MOVs start moving, S&L assumed an additional two seconds of 100% LRC instead of three seconds for the review of the heater curves for the thermal overload heaters. Therefore, the total time during which the MOV motor would experience LRC is four seconds (two seconds for ESF loads acceleration and two seconds for MOV movement). The Team found that the operation of overload trip for LRC was 5.5 seconds. This is more than the four seconds of LRC conservatively assumed by S&L for 100% travel of the MOV and therefore, is acceptable.

5.7 Clinton IDR Observation Report 51 - Battery Rack Gap

This OR addressed the concern of the battery rack gap of 1" to 1-1/2" which was larger than the 1/4" to 1/8" gap used for the qualification testing setup by GNB Battery, Inc. Commonwealth Edison Company (CECo) indicated that its Engineering Department had instructed the field staff to adjust the gaps to between 1/4" and 1/8". This solution is applicable to all Byron and Braidwood units and is acceptable to the Team.

5.8 Byron 1 IDR Observation Report 8.9 - Isolation Device in 125 Vdc System

The 125 Vdc control center, which is safety-related, has two nonsafety-related components (the under voltage relay and the ground detector voltmeter) connected to the bus with only one breaker rather than two qualified breakers in series as committed to in FSAR Table 8.1-1 Appendix A.1.75. S&L indicated to the Team that analysis 19-AI-122 Rev. 0 was performed for this situation. The analysis recommended the addition of a qualified fuse in series with the breaker. The Team reviewed the calculation and found the solution acceptable. The Team also reviewed the associated drawings to ensure that the qualified fuses were incorporated in the circuits for all Byron and Braidwood units.

6. Instrumentation and Control

6.1 Byron 1 IDI Finding 6.3 - Bases For Setpoints

This finding indicated that there were no calculations for the AFW pump suction pressure switch settings. This finding is applicable to all Byron and Braidwood units. During the NRC inspection of the week of November 18, 1985, S&L indicated to the Team that a setpoint calculation was performed for the Byron and Braidwood units. The Team reviewed the setpoint calculation for instrument AF-TH02 and found the calculation contained the correct base data such as instrument accuracy, drift, effects of environment on accuracy, and calibration errors. The Team also reviewed other S&L documents such as I&C Procedure SM-AF051, Rev. 0 which provided the methodology to review I&C data related to setpoints, and Project Design Calculation Listing DIT-BB-CID-001, Rev. 0 which contained a list of all calculations related to system instrumentation. The Team finds that these documents satisfactorily address the attributes of IEEE-279 for all the Byron and Braidwood units.

6.2 Byron 1 IDI Finding 6.5 - Switch Contact Operation

This finding indicated that the AFW pump suction pressure switch 1PSL-AF055 contact status specified on the instrument data sheets was inconsistent with what was on the electrical drawings. S&L reviewed the instrument data sheets PS-180, Rev. E of 1PSL-AF055, for Byron 2 and Braidwood 1&2 and found that the switch contact status was shown correct on the instrument data sheets and the drawings were incorrect. S&L corrected the drawings to be consistent with the instrument data sheets. The Team reviewed the following schematics and found that the status of switch contact shown on those schematics was consistent with what was on the instrument data sheets. Drawings reviewed were:

Byron 2 - 6E-2-4030-AF01
Braidwood 1 - 20E-1-4030-AF01
Braidwood 2 - 20E-2-4030-AF01

The Team considers this finding satisfactorily resolved.

6.3 Byron 1 IDI Finding 6.11 - Identification of Safety-Related Components

This finding addressed an alleged omission in identifying certain safety-related control cabinet modules on the safety-related component list. During the NRC inspection of the week of November 18, 1985, S&L indicated to the Team that a safety-related component list is under preparation in accordance with Project Instruction PI-BB-68, Rev. 0. This list will assign a number to each component located in safety-related cabinets (such as PC cards) which do not have unique ID numbers at the present time. The list will be prepared for each of the three units. The Team considers this finding satisfactorily resolved.

6.4 Byron 1 IDI Finding 6.12 - Equipment Status Display System Criteria

This finding cited a discrepancy between the design criteria and the logic diagrams for status displays deactivated equipment. During the implementation inspection of S&L during the week of November 18, 1985, S&L informed the Team that design criteria DC-ME-07-BB had been revised to correct this discrepancy. The Team reviewed Revision 3 of that criteria and found that the criteria reflected the actual implementation and the discrepancy cited in the finding was satisfactorily resolved. This resolution is applicable to all Byron and Braidwood units.

6.5 Byron 1 IDI Finding 6.10 - Stamping of Safety-Related Drawings

This finding addressed the absence of the "nuclear safety related" stamp from control logic diagram drawing M-4037-1AF04 for Byron 1. This error was found on drawings M-4122-2AF04 for Byron 2, M-4037-1AF04 for Braidwood 1, and M-4122-2AF04 for Braidwood 2. S&L indicated that these errors had been corrected and the "nuclear safety related" stamp was shown on these drawings. The Team reviewed the revised prints of these drawings and concurred with S&L. The Team also randomly selected other safety systems drawings from the Byron and Braidwood units for review to ensure that the "nuclear safety related" stamp was on every one of them. The Team found that all drawings reviewed had the necessary stamp.