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

OCT 19 1984

Docket No. 50-454

Commonwealth Edison Company
ATTN: Mr. Cordell Reed, Vice President
PO Box 767
Chicago, Illinois 60690

Gentlemen:

SUBJECT: BYRON GENERATING STATION INDEPENDENT DESIGN INSPECTION REPORT NO 50-454/83-32

Reference (a): October 1, 1984 letter from Mr. T. R. Tramm to Mr. R. C. DeYoung

Reference (a) provided additional information to address NRC staff questions raised during our review of your response to the NRC's Integrated Design Inspection Report and Bechtel's Independent Design Review Report. One of the items addressed in reference (a) is corrective action taken as a result of trends identified in the Bechtel Independent Design Review Report.

Provided below is your stated corrective action for each trend along with NRC comments and request for additional information.

Use of Undocumented Judgments

CECo Response: Standards have been issued by Sargent & Lundy in the Electrical, Structural, and Mechanical areas via standards ESI-253, SAS-22, and MAS-22. These standards require documenting engineering judgments.

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

Insufficient Control of the FSAR

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

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NRC Staff Comments: Please describe how it is assured that the FSAR is updated to reflect the actual design of the Byron plant when design changes are made.

Insufficient Review of Changes

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

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

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

NRC Staff Comments: Commonwealth Edison (and Sargent & Lundy) is committed to Regulatory Guide 1.64 and ANSI N45.2.11. Please describe measures taken to assure that your procedures and Sargent & Lundy procedures reflect provisions of ANSI N45.2.11 with respect to review of changes. Please describe training conducted to assure that personnel are aware of their responsibilities for design verification.

Additional Matters

Reference (a) also noted that additional information on the completed auxiliary building flooding calculation would be provided. On October 10, 1984, we received the information by telecopy. With respect to the flooding calculation please provide the completed flooding calculation and documentation of reviews noted in the October 10, 1984 telecopy.

Sincerely,

J. N. Grace

J. Nelson Grace, Director
Division of Quality Assurance, Safeguards and Inspection Programs
Office of Inspection and Enforcement

cc w/enclosure:

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Distribution
DCS
QASIP Reading
QAB Reading
NRC PDR
LPDR
SECY
OPE
OGC
WJDircks
HRDenton
NRC Resident Inspectors (Byron and Braidwood)
NRR Project Managers (Byron and Braidwood)
Regional Administrator Region 3
RCDeYoung
JMTaylor
JNGrace
JGPartlow
GTAnkrum
JMilhoan
RVollmer
DGEisenhut

IE:QAB	IE:QAB	IE:QASIP:DD	IE:QASIP:D
JMIlhoan	GTAnkrum	JGPartlow	JNGrace
10/15/84	10/15/84	10/15/84	10/15/84

Commonwealth Edison
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Address Reply to: Post Office Box 767
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October 1, 1984

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

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

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

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

Dear Mr. DeYoung:

This letter provides additional information to address NRC questions raised during the review of our response to the NRC's report on their Integrated Design Inspection (IDI) and to the report of the Bechtel Independent Design Review (IDR). Submittal of this information was requested in a meeting in Glen Ellyn on September 1? 1984 and in a conference call on September 21, 1984.

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

There are only three items which remain to be provided to resolve IDI/IDR concerns. 1 FSAR changes necessary to close IDR Observation 8.47 will be provided later this week. 2 Additional information on auxiliary building flooding will be provided to address IDI Finding 2-19 later this week. 3 A description of the methodology used to address pipe whip in the jet impingement study provided in reference (a) will also be provided later this week.

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Please address further questions regarding this matter to this office.

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

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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

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ATTACHMENT A

ADDITIONAL INFORMATION

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

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

Bechtel Response:

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

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

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

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

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

Bechtel Response:

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

Item 5 (Cont'd)

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

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

Item 14

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

Bechtel Response

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

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

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

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

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

Code — Procedure was qualified

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

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

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

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

3 of 7

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

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

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

OR File #	Subject	Reasons for No Concern Elsewhere
8.39	Power Cables in Cable Spreading Room	The resolution of the observation pointed out that the uncovered power cables actually was not included within the defined area of the cable spreading areas. Therefore, this was not a deficiency. But design change initiated
8.40	Cable Separation	OK Other manhole drawings were checked and did not contain any conflicting lines or any lines at all. Therefore, this appeared to be isolated to the subject drawings. Field inspection showed that the cables are installed correctly.
8.41	Motor Operated Valve Operators	To resolve this observation S&L performed a calculation to verify that MOVs required to function upon a safety signal will perform their safety function. In this calculation S&L included MOVs of all safety related systems in the plant. S&L expanded the scope of this observation to assure that the design of power supplies to 460V motors and MOVs of other systems are adequate with regard to this concern.
8.42	Cable Saddles in Manholes	Since the design of the cable saddle was proven as adequate, use of these saddles elsewhere would also be acceptable. Were they used elsewhere?
8.44 Need to look at OR8.43	CCW Nozzle Loads	The Observation dealt with a question of whether the designer's judgment that the effects of thermal growth produced insignificant stress levels and nozzle loads was justified. After extensive review within S&L and by Bechtel, the IDR team judged the configuration in question to be unique, and concluded that there was no reason for concern with similar judgments elsewhere in the plant.

TABLE-1
OBSERVATION REPORT SUMMARY
OF LIMITS TO CONCERNS

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

#0127C

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

Item 1

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

Sargent & Lundy Response

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

Item 3

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

Sargent & Lundy Response

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

Item 4

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

Sargent & Lundy Response

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

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

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

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

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

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

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

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

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

Item 7 (Observation Report 8.21)

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

Sargent & Lundy Response

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

Item 8 (Observation Report 8.29)

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

Sargent & Lundy Response

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

Item 9 (Observation Report 8.32)

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

Sargent & Lundy Response

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

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

Loadings

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

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

Analysis

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

Conclusion

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

Item 10 (Observation Reports 8.34 and 8.35)

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

Sargent & Lundy Response

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

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

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

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

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

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

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

Piping loads on the support drawing are:

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

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

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

OR 8.35, Item 4.1, Pipe Support 1CC01010X

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

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

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

OR 8.35 (4.2) Piping Support 1CC01051X

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

OR 8.35, Item 4.3, Pipe Support 1CC01012R

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

Documentation of Engineering Judgements

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

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

Item 13 (Observation Reports 8.23 and 8.27)

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

Sargent & Lundy Response

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

Item 15

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

Sargent & Lundy Response

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

Use of Undocumented Judgements

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

Insufficient Control of the FSAR

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

Insufficient Review of Changes

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

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

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

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

Noncompliance with Code Requirements

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

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

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

Item 16

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

Sargent & Lundy Response

The required FSAR change is attached.

HIGH ENERGY LINE BREAK REPORT

Item 1

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

Sargent & Lundy Response

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

Item 2

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

Sargent & Lundy Response

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

Item 3

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

Sargent & Lundy Response

Copy attached.

CAW-7732
CBW-4754

Westinghouse Electric Corporation
Water Reactor Divisions
Nuclear Operations Division
Box 355
Pittsburgh Pennsylvania 15230

August 1, 1984

Ref: SLWC-3121, 7/26/84

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

Attention: K. J. Green

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

Dear Mr. Leone:

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

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

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

J. L. Tain Jr.

W. E. Kortier, Manager
Commonwealth Edison Projects

JLT/bms/3645D

D. L. Leone, 30L

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

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

Item 6 (Observation Report 8.16)

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

Commonwealth Edison Co. Response

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

Item 11

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

Commonwealth Edison Co. Response

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

Item 12 (Observation Report 8.9)

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

Commonwealth Edison Co. Response

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

ATTACHMENT B

10/1/84

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

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

Need revs for 8.13 & 8.15?

3.9.3.2.1 Pumps

Balance of Plant

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

NSSS

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

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

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

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

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

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

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

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

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

3.9.3.2.2 Valves

Balance of Plant

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

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

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

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

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

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

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

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

a. stress analysis including the seismic loads where applicable,

b. in-shop hydrostatic tests,

c. in-shop seat leakage tests, and

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

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

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

TABLE 3.9-8

DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS

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

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

TABLE 3.9-9 (Cont'd)

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

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

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

9.2.2.4.1 System Availability and Reliability

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

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

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

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

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

9.2.2.4.2 Leakage Provisions and Activity Release

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

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

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

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

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

9.2.2.4.3 Incident Control

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

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

a. letdown heat exchanger CCWS flow controllers,

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

c. component cooling surge tank radiation control valve.

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

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

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

9.2.2.4.5 Electrical Power Supply

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

9.2.2.5 Tests and Inspections

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

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

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

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

9.2.5.4 Tests and Inspections

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

9.2.5.5 Instrumentation Requirements

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

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

QUESTION 010.8

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

RESPONSE

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

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

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