PRC

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

Docket Nos.: STN 50-454
and STN 50-455

MAY 25 1984

MEMORANDUM FOR: The Atomic Safety and Licensing Board for Byron:

Ivan W. Smith
Dr. Dixon Callihan
Dr. Richard F. Cole

The Atomic Safety and Licensing Appeal Board for Byron:

Alan S. Rosenthal
Dr. Reginald L. Gotchy
Howard A. Wilber

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

SUBJECT: FOLLOW-UP ON BYRON INTEGRATED DESIGN INSPECTION
(BOARD NOTIFICATION 84-107)

In accordance with the present NRC procedures for Board Notifications, the following information is being provided:

1. Letter from J. N. Grace (NRC) to C. Reed (CECo), Subject: "Byron Integrated Design Inspection - Report No. 50-454/83-32", dated May 2, 1984

2. Letter from J. N. Grace (NRC) to C. Reed (CECo), Subject: Byron Integrated Design Inspection - Report No. 50-454/83-32", dated May 14, 1984

Both letters discuss NRC follow-up to its September 30, 1983 report on the Byron Integrated Design Inspection. This report was transmitted as Board Notification 83-157, dated October 17, 1983; previous follow-up was transmitted as Board Notification 84-086, dated April 20, 1984.

As additional items concerning staff follow-up on the report become available, they will continue to be provided to the Boards.

Thomas . Novak, Assistant Director
for Licensing
Division of Licensing

cc: SECY (2)
OPE
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Parties to the Proceeding
See next page

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<u>Docket No. 50-454,455</u>

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

May 2, 1984

Docket No. 50-454

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
P. O. Box 767
Chicago, Illinois 6069^

Gentlemen:

SUBJECT: BYRON INTEGRATED DESIGN INSPECTION - REPORT NO. 50-454/83-32

My letter dated April 9, 1984 requested your response to concerns we identified in our inspection during the week of March 26, 1984 at Sargent and Lundy's offices concerning analyses of postulated failures to high and moderate energy piping at the Byron plant. On April 9 and 10, 1984 we conducted a site inspection in order to supplement our inspection in Sargent and Lundy's offices. The enclosure to this letter addresses concerns we identified in both inspections for which your response is requested.

The results of our inspection of Sargent and Lundy's analyses of postulated failures to high and moderate energy piping indicate the analyses are not complete enough to ensure the design is adequate and that additional work is required, as we have noted. We are concerned that the work performed to date may have been non-conservative by taking credit for continued operation of equipment items not covered by the pipe break analyses, e.g., piping, instrumentation lines and cables. In addition, while the specific items identified by us in item 8 of the April 9, 1984 letter and in the enclosure to this letter may be resolved by further analysis, they are indicative of the quality of the design process concerning breaks and cracks in high and moderate energy piping and the need for further review on your part to identify the root cause of the problems identified. You are requested to identify any other areas of these analyses where additional work may be required. Your review should consider the fact that the Sargent and Lundy analyses did not take advantage of field walkdowns; the enclosure indicates examples where field walkdowns would have been useful. In addition, you should provide an assessment, and justification if appropriate, for proceeding with fuel loading prior to resolving each open item (identified by us and by your review) associated with the piping failure analyses for Byron. We are also considering enforcement actions based on the deficiencies in Sargent and Lundy's analyses of postulated failures to high and moderate energy piping.

My letter dated March 23, 1984 identified other items where additional information is required or where additional review of your responses to the subject report is still being conducted. One of the items covered the necessity for your conducting audits of design implementation in areas other than those covered by the Integrated Design Inspection. We note that Commonwealth has

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taken action to have Bechtel conduct a review of Sargent and Lundy covering three systems at the Byron Station. We understand you will submit to us the plans for conducting this review. We plan to leave this IDI report item as an open item pending our review of this plan.

Sincerely,

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

Enclosure:
Byron Followup Inspection

cc w/enclosure:
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4/26/84	4/ /84	4/26/84	4/26/84	4/26/84	5/ /84	4/27/84

C... (RIII)
concurred per telecon

BYRON REINSPECTION OF HIGH AND MODERATE ENERGY

PIPE BREAKS AND CRACKS

SITE INSPECTION

1. Report BB-J1-01 states for Zone 11.6-0 that a fire protection line is routed between Motor Control Center (MCC) 131 x 5 and MCC 132 x 5, and that a line break could at the most disable functions in one MCC only. We determined that the fire protection line is directly above MCC 132 x 5 and 17' from MCC 131 x 5. Water spray could be deflected by nearby ductwork to MCC 132 x 5 and simultaneously travel 17' to MCC 131 x 5. An analysis should be made of the potential for pipe cracks and, if any, the path of water spray.

2. Report BB-J1-01 states for Zone 11.4-0 that a wall separates MCC 131 x 3 from water lines in the area. We determined there are fire protection and other moderate energy lines within 5'-15' of MCC 131 x 3 which are not separated from the MCC by any wall and which would spray the MCC. A determination should be made why these were not identified in the Sargent and Lundy analysis, whether they are postulated to crack, and, if so, the impact on ability to reach safe shutdown.

3. Report BB-J1-01 states that CV lines are oriented away from MCC 131 x 1 and are separated by about 25'. We were unable to locate one high energy CV line (1 CV42E-2") shown on the composite drawing (M-228) used in Sargent and Lundy's analysis. Therefore there is uncertanty as to the effect of breaks in this high energy CV line on equipment in this area. It is noted that item 8.a of our April 9, 1984 letter indicates concern as to jet impingement upon essential service water lines in this area. Analyses should be made of effects of failure to CV lines upon these essential service water lines and other equipment required for safe shutdown. This includes MCC 131 x 1 for which our April 9, 1984 letter raised a question on single active failure of a redundant MCC (item 8.b).

4. We inspected a 1-1/2" boron injection line (1RC30AA-1-1/2) in the cold leg of Loop A. Based on a terminal end break postulated by Sargent and Lundy, we determined that there could be jet impingement upon a 3/4" sample line in the hot leg. This is contrary to Westinghouse requirements (SS 1.19) for limiting small line LOCA's to the affected leg. This relates to the concern expressed in our April 9, 1984 letter (item 5) where the Project Management Division of Sargent & Lundy has not reviewed the Westinghouse design criteria for protection against pipe rupture.

5. We inspected a 12" RHR line (1RC04AB-12) connected to the hot leg of loop C at a location where the FSAR had postulated breaks B10A and B10B. Sargent & Lundy had not determined whether the breaks were circumferential or longitudinal, so we postulated longitudinal breaks and identified potential targets. The targets were loop B and C drain lines, loop B crossover leg flow instrumentation lines, loop B 1-1/2" boron injection line and incore instrumentation lines. It is noted that some of these targets, if impacted and damaged, would result in violation of Westinghouse criteria, e.g., for confining damage to the affected loop.

6. Due to the unavailability in the FSAR of intermediate break locations for the pressurizer spray line, we could not assess compliance with Westinghouse criteria for protection against the effects of such breaks. This area should be evaluated.

7. Calculation 3C8-1083-001 makes statements as to separation of instruments required for safe shutdown. Based on our field walkdown, we were unable to confirm that this separation also existed for the cabling and instrumentation lines associated with these instruments. Specific cases reviewed were the source range neutron detectors and pressurizer pressure transmitters.

INSPECTION AT SARGENT & LUNDY OFFICES

1. Calculation 3C8-1083-001 defines "single train" zones as zones containing safe shutdown components or cables from only one train of the respective systems contained in these zones. The report states, that following any initiating high energy line break event in a "single train" zone, the additional failure by fluid jets of a safe shutdown component within the zone of this line break would be no worse than the initiating line break, i.e., either would disable that train. For each "single train" zone, you should verify there is no other piping except for that associated with the specific train of the specific system in the zone. If there is other piping, you should evaluate the effects upon the equipment in the zone resulting from jet impingement and/or water spray due to failure of that piping. This evaluation should consider that jets from piping breaks in nearby zones may reach components in the specific "single train" zone being evaluated. (See item 3 of our April 9, 1984 letter with respect to integrity of walls surrounding equipment cubicles.)

2. Item 1 in our April 9, 1984 letter states that there should be an evaluation of jet impingement effects on piping. This evaluation should consider that, in some cases, jet impingement may not cause breaks or cracks to piping within the target zone, but it will bend, crimp or otherwise deform the pipe. Analyses should be made as to the effects upon pipes due to jet impingement and whether such effects will cause loss of

functionality such that credit cannot be taken for their use in establishing safe shutdown.

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

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

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

May 14, 1984

Docket No. 50-454

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

Gentlemen:

SUBJECT: BYRON INTEGRATED DESIGN INSPECTION - REPORT NO. 50-454/83-32

My letter dated March 23, 1984 addressed responses contained in your December 30, 1983 letter where additional information was required or where additional review of your responses was still being conducted. The enclosure to this letter requests information necessary for us to complete our review.

Sincerely,

J. N. Grace

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

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

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REQUEST FOR ADDITIONAL INFORMATION

BYRON INTEGRATED DESIGN INSPECTION

Finding 2-1: Diesel Engine Air Intake

Our March 23, 1984 letter requested a copy of the documented walkdown which concluded that there are no additional non-safety-related components that will impair the function of the intake line. Please indicate how the turbine building crane was assessed relative to potential failure during a seismic event and consequential damage to the diesel intake line, unless this is covered in the documented walkdown.

Finding 2-4: Time Delay on Logic Diagram

(1) What system ensures that logic diagrams will be revised when the associated schematic diagram is revised?

(2) Please indicate the systems associated with each drawing referenced in FCR No. F21, 265.

Finding 2-8: Missing Calculation for Containment Spray

W. believe that the FSAR statements are design bases and are licensing co itments. Our letter dated March 23, 1984 (page 1 of enclosure) requested you to describe the provisions in your review program (of Project Management Division's calculations) to determine that all necessary calculations have been identified and performed. Please indicate how you ensured that necessary calculations were identified and performed relative to FSAR statements.

Unresolved Item 3-1: Rod Hanger and Pipe Rest Supports

The following outline is provided to clarify the team's intent:

(Use of infinite support stiffness met the licensing commitment in the sense that there was no specific commitment to use realistic stiffness in piping analyses.

(2) Our sample problem indicated that calculated piping stresses varied somewhat when realistic stiffnesses were employed, but not enough to matter with respect to the piping stress.

(3) Our sample problem indicated that calculated seismic support loads varied when realistic stiffnesses were employed. The maximum increase in a support load was 70 percent. This result is shown in Table 2 of the EG&G report at Sargent & Lundy Node 98A:

609 lb - EG&G calculated SSE load using reasonable stiffness

358 lb - S&L calculated SSE load using infinite stiffness

251 lb - 70 percent increase over the S&L calculated load

(a) In the sample problem, this type of variation was not considered to matter with respect to support strength in view of the large margins typically provided.

(b) However, we were concerned about your up-lift check procedures for non-linear supports such as pipe rests and rod hangers. When the seismic loads exceeded the dead weight and thermal loads further checking was performed to assure that unloading did not cause problems, e.g., checking of pounding action and of increased loads on adjacent supports. Our concern was as follows:

(i) If reasonable support stiffnesses were used, the predicted seismic loads would be substantially greater in some cases.

(ii) Some non-linear supports which were not originally predicted to unload and thus were not checked would be expected to unload.

(iii) We, therefore, intended to suggest that you check additional non-linear supports for unloading - for example, those where seismic loads exceed about half of deadweight and thermal loads.

You are requested to describe your plans to assure that seismic unloading of non-linear supports, where that can be expected, will not cause overstress due to pounding or increased loads on adjacent supports. It is noted that this request was provided to you by telecopy on April 2, 1984.

Finding 6-12: Equipment Status Display Criteria

Please inform us of the date that we can review the final design of the Equipment Status Display System.