143

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September 28, 1984

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FEDERAL EXPRESS

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Dr. A. Dixon Callihan
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4350 East West Highway
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Bethesda, Maryland 20814

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

Dear Administrative Judges:

Please find enclosed for your review copies of the following letters:

1. June 19, 1984 letter from Cordell Reed to R.C. De Young transmitting responses to March 23 and May 14, 1984 inquires.

2. August 16, 1984 letter from D.L. Farrar to James Keppler, transmitting responses to April 9 and May 2, 1984 inquiries.

3. August 30, 1984 letter from D.L. Farrar to R.C. De Young transmitting revised responses to findings and unresolved items.

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These letters followed up on the December 30, 1983 letter from Mr. Cordell Reed to Mr. R.C. De Young transmitting the "Byron IDI Response", which was served on the Licensing Board and the parties under my letter of September 25, 1984. As such, these letters are also relevant to Intervenor's pending "Motion to Reopen the Record...to Include the Byron Station Design as an Issue" and Commonwealth Edison Company's forthcoming response to that motion.

Please note the following letters, which have previously been served on the Licensing Board and the parties, are also relevant to the IDI Report:

Served April 20, 1984 (Board Notification No. 84-086):

1. March 23, 1984 letter from Nelson Grace to Cordell Reed, containing requests for further information on the IDI.

2. April 9, 1984 letter from Nelson Grace to Cordell Reed, requesting further information on High and Moderate Energy Pipe Breaks and Cracks.

Served May 25, 1984 (Board Notification No. 84-107):

1. May 2, 1984 letter from Nelson Grace to Cordell Reed, requesting further information on High and Moderate Energy Pipe Breaks and Cracks.

2. May 14, 1984 letter from Nelson Grace to Cordell Reed requesting further information on the IDI.

Very truly yours,

Victor Copeland

Victor G. Copeland

One of the Attorneys for
Commonwealth Edison Company

VGC:mbn

Enclosures

cc: Service List (with enclosures)

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

RELATED CORRESPONDENCE

DOCKETED
USNRC

'84 OCT -1 P1:41

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

June 19, 1984

Mr. R. C. DeYoung
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 Nos. 50-454/83-32
NRC Docket Nos. 50-454 and 50-455

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

(b): March 23, 1984 letter from J. Nelson Grace to Cordell Reed.

(c): May 14, 1984 letter from J. Nelson Grace to Cordell Reed.

Dear Mr. DeYoung:

This letter supplies additional information regarding Commonwealth Edison's responses to the findings, unresolved items, observations and general concerns which were identified during the Byron integrated design inspection.

Attachment A to this letter contains responses to the NRC questions contained in references (b) and (c) regarding issues not associated with pipe break analyses. The pipe break issues will be addressed in a separate letter in the near future.

Please address further questions regarding this matter to this office.

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

Very truly yours,

for Cordell Reed
Vice President

bs

cc: J.G. Keppler - w/Attachment

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Attachment A

BYRON IDI RESPONSES

Table of Contents

Section

Section I

Response to NRC Letter Dated March 23, 1984

General Item:

"Systematic review and corrective action program to assure that the necessary calculations in the mechanical systems discipline are identified, performed and updated as needed to support the current design.

Your response stated that all safety-related calculations in the Project Management Division calculation books were being reviewed in accordance with an approved instruction to determine that they were technically adequate to support the current Byron/Braidwood design and to determine if the format conformed to the applicable version of Procedure GQ 3.08. You also stated that no hardware changes had resulted from these reviews, which were about 80% complete. You are requested to provide the following additional information about this review program when completed:

1. Describe the provisions in the review program to determine that all necessary calculations in this discipline have been identified and performed. Indicate the number of new calculations, if any, that were performed.

2. Indicate the number of calculations, if any, where:

   (a) Hardware changes were made

   (b) Reanalysis was performed

   (c) Updated information was incorporated or documentation was improved

   (d) Incomplete calculations had been approved

   (e) Additional action 'as taken to correct root causes or generic deficiencies

In addition, with respect to the overall project, your response noted that Commonwealth Edison quality assurance audits have included design calculations and that problems identified in those audits were pursued to determine root causes and seek out generic deficiencies. You are requested to provide the following information with respect to previous Commonwealth Edison audits of Sargent & Lundy design calculations:

1. The number of calculations audited

2. The number of calculations, if any, where:

(a) Hardware changes were made

(b) Reanalysis was performed

(c) Updated information was incorporated or documentation was improved

(d) Incomplete calculations had been approved

(e) Additional action was taken to correct root causes or generic deficiencies."

RESPONSE

The review of all safety-related calculations in the Project Management Division (PMD) calculation books, performed by the Byron project engineers, as described in the responses submitted with the December 30, 1983 letter, has been completed. This review was initiated to address the stated IDI concerns about the adequacy of the calculations previously performed by Sargent & Lundy Byron project PMD engineers. The results of this review indicated that the current design was adequately supported by these calculations.

As stated in this request for additional information, the objectives of this review were to verify that the existing calculations met the standards and instructions in effect at the time the calculations were performed, and to verify that these calculations were technically adequate. This review program did not include any specific provisions to determine that all necessary calculations by Byron PMD engineers had been identified and performed. However, the following two considerations should resolve this particular issue:

1. The safety-related calculations in the Byron PMD calculation books represent the calculations that were determined to be the necessary and sufficient calculations required by the Project Manager and/or Mechanical Project Engineer, as required by Sargent & Lundy Quality Assurance Procedure GQ-3.08. To address the concern, the Mechanical Project Management Engineers initiated a survey to confirm that the necessary PMD calculations have been performed. Two additional calculations resulted from the survey. These calculations were performed to provide documentation for the size of the diesel oil day tank and the diesel oil storage tank. These calculations represent the compilation of the formally documented calculations performed by PMD engineers during the course of the project. As such, we believe that all necessary calculations were performed.

2. In order to provide additional assurance that Sargent & Lundy has adequately addressed this issue, Commonwealth Edison Company (CECo) has retained the services of Bechtel Power Corporation to perform an Independent Design Review (IDR) of three systems on the Byron/Braidwood plants. This systematic review will include an evaluation of the design adequacy and the design process on these systems, and will ensure that the output documents (e.g., calculations) meet the licensing commitments and safety-related design requirements.

The following tabulation summarizes the results of Sargent & Lundy's review of the existing PMD calculations:

Category	No. Of Calculations
a) Hardware changes were made	0
b) Reanalysis was performed	0
c) Updated information was incorporated or documentation was improved	73
d) Incomplete calculations had been approved	0
e) Additional action was taken to correct root causes or generic deficiencies	0
f) No changes to original calculations	39
Total Calculations Reviewed	112

The following clarifications to the categories presented above need to be made. Category (b) was defined as those instances where an existing calculation was found to be technically deficient or was not conservative relative to the existing design, and therefore, required a detailed analysis to verify the adequacy of the plant design. No calculations were determined to be included in this category. Category (c) was defined as those instances where the existing calculation was technically adequate and supported the current design; however, revisions to the calculation format, list of references, updated information, or other related areas were made in order to improve the documentation aspects of the calculation. In no instance did these changes result in a design change or hardware change.

3. In the above letter from the NRC, Commonwealth Edison was requested to respond and supply information regarding the number of calculations for the Byron Project which were examined during Commonwealth Edison Quality Assurance Department audits of Sargent & Lundy. Edison audits of design calculations began in early 1979; and during the period February 1979 through January 1984, Edison Quality Assurance conducted 22 audits of Sargent & Lundy which examined work pertaining to the Byron Project. During 12 of the 22 audits, approximately 325 calculations for the Byron/Braidwood project were reviewed. It should be noted that, for the most part, calculations are applicable to both projects; however, some calculations were unique to either Byron or Braidwood. All of the calculations were processed by the same team of Sargent & Lundy personnel because the Byron/Braidwood stations are designed under a single project group.

A summary of the corrective actions resulting from the above audits is as follows (using the same definitions as in Item 2 above):

a) No calculations were found to have problems which resulted in hardware changes.

b) One calculation resulted in reanalysis.

c) 16 calculations were noted where updated information was incorporated or documentation was improved. A breakdown of the 16 calculations is as follows: 14 required documented calculations to be originated to support the design, 1 resulted in a Design Criteria change, and lastly, 1 calculation was updated to show the correct load value.

d) No cases were noted where incomplete calculations had been approved.

e) As a result of the 17 calculational deficiencies referred to in 3(b) and 3(c) above, Sargent & Lundy performed extensive reviews for two of the deficiencies. The action taken to prevent recurrence included revisions to procedures.

Finding 2-1: Diesel Engine Air Intake

"You are requested to provide for our review 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."

RESPONSE

A copy of the documented walkdown (dated 11-15-83) is enclosed.

ATTACHMENT TO FINDING 2-1

UNIT 1 AUXILIARY FEEDWATER DIESEL AIR INTAKE

LOCATION REVIEW

LINE NO.:

1DØB1A14, Piping Subsystem 1DØ25

MATERIAL:

A-106 GrB; 0.375 inch wall

ROUTING:

The air intake line runs a short distance in the Auxiliary Building and penetrates "L" line wall into the Turbine Building at EL 391'-10". The line turns upward immediately and penetrates level 401. At level 401 the line extends upward for approximately 6 feet with two 90° elbows and a debris screen welded onto the end of the pipe. Intake air for the Auxiliary feedwater diesel is taken from the Turbine Building air volume. The total length of intake pipe is approximately 30 feet.

PIPE SUPPORT:

The pipe is seismically supported in the Auxiliary and Turbine Buildings by an anchor provided at "L" line wall. One pipe support located in the Turbine Building is provided for support of the debris screen and double elbows.

TURBINE BUILDING INTERACTION:

Below El 401: The intake line runs vertically from El 391'-10" to level 401 and is located 2'-10" from the edge of "L" line wall. This area is free of larger components and no hazard to pipe integrity exists. The pipe is physically above most nearby components.

Above El 401: The intake line ends approximately 2'-10" from "L" line wall. In plan, the pipe is located West of the CO_2 storage tank and North of the Turbine Building elevator shaft. All piping within eight feet of the intake line is 6" or less in diameter.

The nearest larger diameter pipe is 1ØG21C24. This ØG line is located above the floor with centerline at El 413'-3". In plan the ØG line is located approximately eight feet East of the intake line. The elevator shaft structure would provide ample protection to shield the intake line from a swinging ØG line.

Two cable trays run vertically overhead but would not be capable of jeopardizing the integrity of the intake air line.

ATTACHMENT TO FINDING 2-1

(Cont'd)

CONCLUSION:

The Auxiliary feedwater pump diesel intake air line is a reasonably short run of pipe. The air intake pipe is routed to provide a minimum exposure to non-safety related components capable of damaging it. The air intake location is acceptable.

UNIT 2:

The Unit 2 air intake location is similar to Unit 1, however the installation status is incomplete at this time. An inspection of area indicated a similar piping and equipment arrangement.

REFERENCES:

1. Piping analytical drawing 1DØ25, sheets 1 and 2, Rev. 1
2. Composite drawing M-330, Rev. T
3. Photographs taken 11-15-83.

Prepared James Austin 11-15-83

Reviewed [illegible] 11-15-83

Approved Dennis Demons 11-15-83

Finding 2-3: Basis for Time Delay

"You are requested to describe the basis for determining that the hydraulic transient, considering worst case conditions such as minimum technical specifications condensate storage tank level, high condensate water temperature, simultaneous pump start and runout flows, does not result in pump trip or undesirable addition of relatively impure ESW system water to the steam generators."

RESPONSE

The auxiliary feedwater system preoperational test (2.3.10) included a section to determine whether a hydraulic transient on a simultaneous pump start would result in a sudden decrease in system pressure causing either a pump trip or the opening of the essential service water supply valves. The test was conducted under the following conditions:

a. Condensate storage tank level was verified to be at 197,500 gallons. (The test procedure requires the condensate storage tank level to be at the minimum technical specification level of 200,000 gallons, $\pm$ 25,000 gallons.)

b. Suction pressures initially recorded,

c. AFW pumps started individually and suction pressures recorded, and

d. AFW pumps started simultaneously and suction pressures recorded.

The maximum suction pressure transient recorded was a 10.2 psi drop in pressure for the motor-driven pump during the simultaneous start (initial suction pressure approximately 20 psig); however, the suction pressure stabilized around 17 psig. The setpoint for switchover to essential service water is approximately 14.1 psia (1.22 in. Hg. vac.). The ambient water temperature was not recorded since this parameter does not have any effect on the results as the difference in water vapor pressure between 40°F and 100°F is negligible. The pumps were not operated at a runout condition since runout orifices are installed in each auxiliary feedwater supply line to prevent this situation from occurring.

Based on the test results, it has been sucessfully verified that a simultaneous pump start does not induce a hydraulic transient that will either open the essential service water supply valves or trip the pump due to a sudden loss of suction pressure. The test also verified that a time delay on the pump trip circuit is not required.

Finding 2-13: Design Criteria Updating

"We recognize your statement that the design criteria are intended to guide design efforts in the initial phases of design. However, we do not understand how development of a status list, by itself, will provide an effective safeguard to assure that personnel performing safety-related activities will not be mislead by obsolete information, particularly in view of the following:

1. The design criteria appear to be important and useful documents with widespread distribution, including availability to the plant staff and design engineers.

2. They are controlled design documents and this creates a tendency to assume they are kept correct and current.

3. Since they are generally not being updated, many will contain obsolete and potentially misleading information.

Accordingly, you are requested to describe your plans for additional measures, such as stamping all copies, to assure that personnel performing safety-related activities are not mislead by obsolete information."

RESPONSE

Based on a review of the 30 Byron/Braidwood safety-related design criteria documents, the March 23, 1984 status report classified each design criteria under one of the following categories:

		Number of Design Criteria
Design -	The design criteria is correct, reflects the current engineering design, and can be used as a design document.	18
Information -	The design criteria is not 100 percent correct. It does, however, provide a design basis for reference but cannot be used as a design document.	12
Obsolete -	The design criteria does not reflect the current design and cannot be used as a design document.	0
	TOTAL	30

While we still believe that the identification of the document status through a status list is adequate, those design criteria classified as "information" (or if categorized as "obsolete" in the future) will be appropriately identified on each page and redistributed in accordance with the project distribution list. We believe we have addressed all of your concerns in regard to this item.

Finding 2-18: Flooding Analysis

"For the RHR heat exchanger cubicles (Item B in your response), you are requested to provide the load due to the maximum flooding level, the design live load for the floor and the ultimate capacity of the floor."

RESPONSE

In the RHR heat exchanger cubicles, the maximum flooding level is 101 inches, which results in a dead load of 0.525 KSF. The occupational live load used was 0.05 KSF and the total factored uniform design load for the SSE level load combination, of which the flooding is a part, is W_{SSE} = 4.91 KSF. The allowable ultimate capacity of the floor, based on the strength design method in ACI 318-71, is 6.65 KSF. Therefore, the floors in question have ample capacity for accommodating the flooding load.

Finding 3-2: Functionality Criteria

"Your response indicated that Class 1, 2, and 3 stresses are being utilized to evaluate for functionality. However, it is not clear how the analyst decides whether or not the Class 1, 2, or 3 stress results from PIPSYS are acceptable per the functionality requirements when certain stress indices and stress intensification factors (specified in Tables 1 and 2 of the General Electric report) are higher than those listed in the Winter 1979 Addenda of the ASME Code which is the basis for the PIPSYS calculations. You are requested to explain further how these decisions are made."

RESPONSE

The following is a detailed discussion to illustrate that the qualification methods used by Sargent & Lundy to satisfy functional capability requirements are acceptable. This is true regardless of which addenda of the code is utilized for stress indices or stress intensification factors.

I. DIFFERENCES IN CLASS 1 STRESS INDICES:

A comparison of the stress indices (Table F3-2.1 attached) illustrates that the stress indices specified in Winter 1979 Addenda of the code (basis for PIPSYS) are equal to, or more conservative than, those specified in Table 1 of the General Electric report (NEDO-21985) except for the B_1 and B_2 indices for branch connections and butt welding tees.

As stated in our original response of December 30, 1983, a portion of the Class 2 and 3 piping is evaluated for functional capability using Class 1 analysis rules as delineated in Section 2.2e of NEDO-21985. (Namely: Eq. 9 of NB-3600.) For all cases where this approach was utilized, the allowable stress limit was considered to be 1.5 S_y. The other two methods which can be used for evaluating Class 2 and 3 piping for functional capability are the scanning method and detailed hand calculations per NEDO-21985. All three qualification methods are outlined in EMD TP-2, Rev. 4 (EMD-046032), as stated in our original response.

General Electric report (NEDO-21985) allows the use of 2.0 S_y as the allowable stress limit for the calculated stresses for the branch connections and butt welding tees using the higher values of B_1 and B_2 indices.

The ratio of allowable stress limit recommended by the General Electric report (NEDO-21985) to the allowable stress limit used by Sargent & Lundy $\left[\frac{2.0\ S_y}{1.5\ S_y}\right]$ is equal to 1.33.

The ratio of B_1 indices $\frac{B_1\ (S78)}{B_1\ (W79)}$ and B_2 indices $\frac{B_2\ (S78)}{B_2\ (W79)}$

for the worst case, is also equal to 1.33. Therefore, the use of 1.5 S_y, by the analyst, as the allowable stress limit will assure functional capability, in accordance with our licensing commitment regardless of which addenda of the code is utilized for the stress indices.

II. DIFFERENCES IN CLASS 2 & 3 STRESS INTENSIFICATION FACTORS

A comparison of the stress intensification factors (Table F3.2-2 attached) illustrates that the stress intensification factors specified in Winter 1979 Addenda of the code (Basis for PIPSYS) are equal to or more conservative than those specified in Table 2 of the General Electric report (NEDO-21985) except for welding elbows or pipe bends and welding tees.

As stated in our original response of December 30, 1983, most of the Class 2 and 3 piping is evaluated for functional capability using the stress scanning method, which is a very conservative approach. This means that the Service Level C PIPSYS stresses are scanned to assure that they do not exceed the Service Level B allowable stress limits. (Namely 1.2 S_h). The other two methods which can be used for evaluating Class 2 and 3 piping for functional capability are the detailed hand calculations per NEDO-21985 or Class 1 analysis rules. These qualification methods are outlined in EMD TP-2, Rev. 4 (EMD-046032), as stated in our original response.

The General Electric report (NEDO-21985) allows the use of 1.5 S_y as the allowable stress limit for calculated stresses for all piping components.

The most conservative ratio of allowable stress limit recommended by the General Electric report (NEDO-21985) to the allowable stress limit used by Sargent & Lundy

$\left[\frac{1.5\ S_y}{1.2\ S_h}\right]$* is equal to 2.0. This ratio addresses the

most conservative assumptions of material properties. The ratio of the stress intensification factors for welding

elbows and pipe bends $\left[\frac{\frac{1.3}{h^{2/3}}\ (S78)}{0.75 \times \frac{0.9}{h^{2/3}}\ (W79)}\right]$ is 1.93

*S_h is the allowable stress limit at 557° F. S_y is the yield stress value at 557° F. The ratio was determined for the worst case.

and the ratio of the stress intensification factors for welding tees $\frac{.90i \quad (S78)}{.75i \quad (W79)}$ is 1.28.

Therefore, the use of the stress scanning method will assure functional capability in accordance with our licensing commitment regardless of which addenda of the code is utilized for the stress indices.

We believe that we have addressed all of your concerns in regard to this item.

TABLE F3.2-1

COMPARISON OF CLASS 1 STRESS INDICES BETWEEN
THE 1979 WINTER ADDENDUM OF THE 1977 ASME CODE
AND THE 1978 SUMMER ADDENDUM OF THE 1977 ASME CODE

FITTING	1979 WINTER ADDENDUM OF 1977 ASME CODE (BASIS FOR PIPSYS)	1978 SUMMER ADDENDUM OF 1977 ASME CODE (BASIS FOR NEDO-21985)
Straight Pipe	$B_1 = 0.5$	$B_1 = 0.5$
	$B_2 = 1.0$	$B_2 = 1.0$
Curved Pipe or Butt Welding Elbows	$B_1 = 0.5$	$B_1 = 0.5$ max
	$B_2 = \frac{1.46}{h^{2/3}}$	$B_2 = \frac{1.3}{h^{2/3}}$ max ≥ 1.0
Branch Connections	$B_1 = 0.5$	$B_1 = 0.5$ unless B_{2r} or $B_{2b} = \frac{4}{3}$ then $B_1 = 0.67$
	$B_{2r} = 0.75C_{2r} \geq 1.0$	$B_{2r} = 0.75C_{2r} \geq \frac{4}{3}$
	$C_{2r} = 0.8\left(\frac{R_m}{T_r}\right)^{2/3}\left(\frac{r'_m}{R_m}\right) \geq 1.0$	$C_{2r} = 0.8\left(\frac{R_m}{T_r}\right)^{2/3}\left(\frac{r'_m}{R_m}\right) \geq 1.0$
	$B_{2b} = 0.5\ C_{2b} \geq 1.0$	$B_{2b} = 0.5\ C_{2b} \geq \frac{4}{3}$
	$C_{2b} = 3\left(\frac{R_m}{T_r}\right)^{2/3}\left(\frac{r'_m}{R_m}\right)^{1/2}\left(\frac{T'_b}{T_r}\right)\left(\frac{r'_m}{r_p}\right) \geq 1.5$	$C_{2b} = 3\left(\frac{R_m}{T_r}\right)^{2/3}\left(\frac{r'_m}{R_m}\right)^{1/2}\left(\frac{T'_b}{T_r}\right)\left(\frac{r'_m}{r_p}\right) \geq 1.5$
Butt Welding Tees	$B_1 = 0.5$	$B_1 = 0.5$ unless B_{2r} or $B_{2b} = \frac{4}{3}$ then $B_1 = 0.67$
	$B_{2b} = 0.4\left(\frac{R_m}{T_r}\right)^{2/3} \geq 1.0$	$B_{2b} = 0.4\left(\frac{R_m}{T_r}\right)^{2/3} \geq \frac{4}{3}$
	$B_{2r} = 0.75\left(\frac{R_m}{T_r}\right)^{2/3} \geq 1.0$	$B_{2r} = 0.5\left(\frac{R_m}{T_r}\right)^{2/3} \geq \frac{4}{3}$
Butt Welding Reducers	$B_1 = 1.0$	$B_1 = 1.0$
	$B_2 = 1.0$	$B_2 = 1.0$
Girth Fillet Weld to Socket Weld Fittings, etc.	$B_1 = 0.75$	$B_1 = 0.5$
	$B_2 = 1.5$	$B_2 = 1.0$

TABLE F3.2-2

COMPARISON OF CLASS 2 AND 3 STRESS INTENSIFICATION FACTORS
BETWEEN THE 1978 WINTER ADDENDUM OF THE 1977 ASME CODE
AND THE 1978 SUMMER ADDENDUM OF THE 1977 ASME CODE

FITTING	1978 WINTER ADDENDUM OF 1977 ASME CODE (BASIS FOR PIPSYS)*	1978 SUMMER ADDENDUM OF 1977 ASME CODE (BASIS FOR NEDO-21985)*
Straight Pipe Butt Weld	1.0	1.0
Welding Elbow or Pipe Bend	$0.75 \left(\frac{0.9}{h^{2/3}}\right)$	$\frac{1.3}{h^{2/3}}$
RFT	0.75i	0.75i
UFT	0.75i	0.75i
WDT	0.75i	0.90i
Fillet Weld Joint, Brazed Joint, Etc.	0.75(2.1) = 1.58	1.0
Reducer	0.75i	0.75i

*All Values Must be $\geq$ 1.0

Finding 3-6: Pipe Support Added Mass

"Your response stated that one example cited in the report did not, in fact, violate your criteria with respect to added mass and the other example exceeded the criteria by an insignificant amount (53 lb vs. 52 lb criterion). However, your response did not address the overall concern. You are requested to confirm on a systematic basis that your procedures for added mass are being uniformly followed or, if not, there is no significant effect on the analysis results."

RESPONSE

A criteria for inclusion of the support added masses in piping analysis does exist as stated in our previous response. Deviations from the existing criteria will not adversely affect the validity of the analysis results based on the inherent conservatism in the total design process.

This has been demonstrated by conducting a technical evaluation of previously completed piping analyses. The sample for this evaluation was determined utilizing the military standard statistical sampling scheme (MIL-STD-105D). The evaluation has been completed and was documented on June 15, 1984.

Furthermore, to ensure uniform application in the use of support added masses in the future, a retraining program is being established. Detailed classroom and hands-on instruction are being conducted with emphasis placed in this area.

Finding 3-7: Overlap of Plate

"You are requested to provide a copy of the following documents for our review:

1. FCR F-9079

2. The backup calculation

3. Revision C to Support Drawing 1AF03009R"

RESPONSE

A copy of each of the requested documents is contained in the attachments, and each of these is discussed below:

1. Attachment A, FCR F-9079:

   Byron FCR F-9079 (dated 4-16-81) was written against Revision B of support drawing 1AF03009R to have a 1/4-inch overlap on its embedded plate. This FCR was picked up on Revision C of the drawing and issued on 12-23-81. Subsequently, this FCR was closed out on 1-9-82 as shown on Part D of the first page of the FCR.

2. Attachment B, Structural Division Calculation:

   A copy of Calculation No. 13.1.2, Pages 521, 522, and 523, is enclosed.

   Page 521 is the approval page of the calculation for Drawing 1AF03009R, Revision C, which incorporates FCR F-9079. This page contains the preparer's and reviewer's signatures and dates, and also shows approver's signature and date (top right side of page).

   Page 522 shows that design load has been changed per Revision C of the drawing from 1753 pounds to 1056 pounds and the review method has been identified (see "Remarks" column and notation for "Remarks" column).

   Page 523 is for backup calculations for the expansion anchor plates and bolt as indicated on Page 522. The supporting calculation for Revision C is shown on lower portion of this calculation sheet with Revision C indication in a box. The calculations shown on upper portions of this sheet are for previous revisions.

3. Attachment C, Drawing 1AF03009R, Revision C:

   Drawing 1AF03009R was revised per FCR F-9079 and to reflect the formal analysis loads.

ATTACHMENT A TO RESPONSE
TO FINDING 3-7

FCR F-9079

CC-HUNTER-JEC

Commonwealth Edison Company
Field Change Request

FCR NO. F-9074
DATE: 4-16-81

PART A] REQUEST CLASS: ☒ Initial Construction ☐ Plant Modification ☐ Major ☒ Minor

PROJECT: BYRON STATION UNIT 1 & 2

P.O. NO: 207010 Scope of P.O. MECHANICAL ERECTION

SYSTEM: AUX FEEDWATER Component: HANGER

Doc/Dwg No: 1AF03009 Rev. B Doc/Dwg. Title: SUPPORT

Description of Change Request: REVISE AND INSTALL PER AB 2889

Reason for Change Request: TO FIT AS BUILT PIPE LOCATION

Does This FCR Result From A NCR ☐ Yes ☒ No

If Yes Give NCR No. Date:

Request Originated By: ☐ Construction ☒ Other

If Other Give Name of Organization: HUNTER

Prepared By: D.STANGEL Date: 4-16-81

Engineering Disposition: ☐ CHED ☐ PROJ. ☒ S&L ☐ NONE

Request Approved By: [signature] Date: 4-21-81

SITE CONSTR. SUPT., MAINT. ASSIST. SUPT. OR DESIGNEE

Logged By: [signature] Date: 4/24/81

CONSTRUCTION DEPT. OR MAINT. DEPT.

ADVANCED VERBAL CONCURRENCE - TELEPHONE VERIFICATION:

Telecon By: V DePesia & R Sampson Date: 4-17-81

Resolution or Approvals: Approved

Recorded By: John DeSario Date: 4-17-81

PART B]

ENGR'G. DISPOSITION: ☐ Approved ☐ Approved With Comment ☐ Rejected

Engr'g. Comment/Instruction:

For Information Only

Engr'g. Approval: Date:

PART C]

A-E DISPOSITION: ☒ Approved ☐ Approved With Comment ☐ Rejected

A-E Comment/Instruction:

A-E Approval: [signature] Date: 12-1-81

Part D] AFFECTED DESIGN DOCUMENTS REVISED AS LISTED: 1AF03009(C)

- LIST HERE OR ATTACH LIST OF REVISED DOCUMENTS -

DESIGN DOCUMENTS ISSUED BY: [signature] Date: 1-9-82

PART E] FINAL DISPOSITION:

REVIEWED BY REQUESTOR: Date:

CONSTRUCTION OR MAINTENANCE

APPROVED BY: Date:

SITE CONSTR. SUPT. OR MAINT. ASSIST. SUPT., OR DESIGNEE

COMPLETION VERIFIED BY: Date:

SITE QA SUPT. OR STA. QA DESIGNEE

88-1235 12-80

Q.P. Form 3-2.2 ...
(Rev. ...

Part F

FIELD CHANGE REQUEST
Contractor Notification

To: Aunter

Action to or taken: INSTALL Per Request

AB 2889

Routing

Construction Supt. or Designee for Signature.
Const. Clerk to route as follows:
Copy part F to contractor
Copy all parts to Q.A. and Project Sta. Const.
Route orig. to P.E. Mgr. or SNED Mgr. or A-E with transmittal letter for appropriate engineering disposition (See part A).
Copy to P.E. Mgr. or SNED Mgr. or A-E where orig. is not transmitted.

~~For Information~~ Only

John D. Pearsin 9-17-81
G.E.Co. Field Engineer Date

Q.P. Form 3-2.2
Page 2 of 2
(Rev. 3)

COMPONENT SUPPORT/C.E.A. DISCREPANCY REPORT

HUNTER CORPORATION

D.R. NO. AB-2889 LOCATION AUX BLDG. UNIT 1 AREA 2

COMPLETE THIS SECTION FOR PIPING SUPPORT DISCREPANCIES

CONST. COPY SUPPORT DWG. NO. 1AE03009R REV. 11B QUALITY CLASS C

LINE 1SXB7AB-8" CONST. COPY PIPING DWG. NO. SX-55 REV. 3

FLOOR ELEVATION BELOW SUPPORT 383' INTERFACE DWGS. ______

REMARKS: ASBUILT PER F.C.R. Delete Prev. Revs

COMPLETE THIS SECTION FOR C.E.A. DISCREPANCIES

PROBLEM: ☐ RE-BAR HIT ☐ NUT WON'T TIGHTEN ☐ C.E.A. WON'T SET ☐ C.E.A. ABANDONED ☐ HOLE ABANDON

C.E.A. ELEVATION ______ C.E.A. SIZE ______ long x ______ diameter

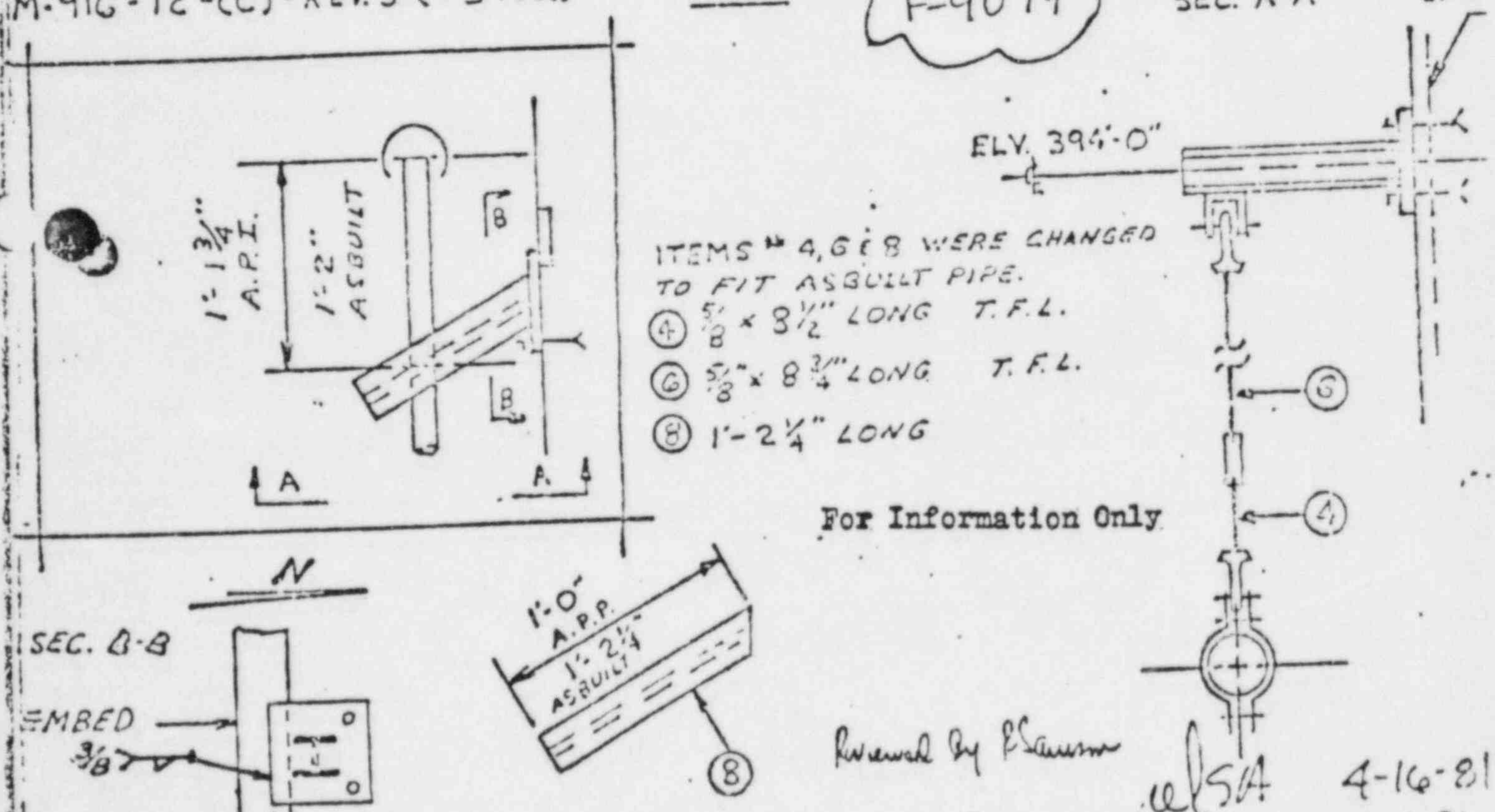
DIMENSIONS AND DIRECTIONS FROM NEAREST COLUMN LINES ______

CONCRETE SURFACE ☐ CEILING ☐ BEAM SIDE ☐ BEAM BOTTOM ☐ COLUMN ☐ FLOOR ☐ WALL

☐ HOLE IS MARKED WITH D.R. NO. ☐ ABANDONED HOLES GROUTED

REMARKS:

M-916-12-(C)-REV.3-(± 3 TOL.) SKETCH F-9079 SEC. A-A



Initiated By [signature] Date 4-9-81

Production Supervisor

ENGINEERING DISPOSITION

RESOLUTION TO COMPONENT SUPPORT DISCREPANCY ACCOMPLISHED THRU: ☐ DESIGN DWG. REV. ☐ CECO F.C.R.

F.C.R. NO. ______ R.F.A. NO. 6925 REMARKS: ______

By ______ Date ______

NOTE: 1. SEND ONE COPY OF C.E.A. DISCREPANCIES TO CECO SCD STRUCTURAL DEPARTMENT.

ATTACHMENT B TO RESPONSE
TO FINDING 3-7

Supporting Calculation for Revision C of
Component Support Drawing No. M-1AF03009R

SARGENT LUNDY ENGINEERS CHICAGO	Calcs. For MECHANICAL COMPONENT SUPPORT NUMBER: M-1AF03009R	Calc. No. 13.1.2
	X Safety-Related / Non-Safety-Related	Rev. 2 Date
		Page 521 of

Client COMMONWEALTH EDISON COMPANY	Prepared by SEE BELOW	Date
Project BYRON/BRAIDWOOD UNIT 1	Reviewed by	Date
Proj. No. 4391-00/4683-00 Equip. No.	Approved by DF Pool	Date 12/21/81

BB I REV SR

MECHANICAL COMPONENT SUPPORT DOCUMENTATION SHEET 2

SD&DD REVIEW RECORD FOR

MECH. COMPONENT SUPPORT DRAWING # M-1AF03009R

DWG.REV.	STATUS	PREPARED BY	DATE	REVIEWED BY	DATE
C	3	M Pinto	12/15/81	[illegible]	12-18-81

NOTATION FOR "STATUS" COLUMN

"1" - PREVIOUS SD&DD COMMENTS INCORPORATED: DOCUMENTATION SHEET 1 SHOWN FOR DRAWING REV. _____ REMAINS UNCHANGED

"2" - THIS DRAWING REVISION DOES NOT EFFECT DOCUMENTATION SHEET 1 FOR DRAWING REV. _____.

"3" - THIS DRAWING REVISION EFFECTS DOCUMENTATION SHEET 1 FOR DRAWING REV. B. CHANGES IN DOCUMENTATION ARE SHOWN NEXT TO SCRATCHED OUT INPUTS.

"4" - OTHER: EXPLAIN:

Form GQ431 Rev. 1

SARGENT & LUNDY ENGINEERS CHICAGO

Calcs. For MECHANICAL COMPONENT SUPPORT NUMBER: M-1AF030U5R REV. 1/2		Calc. No. 3.1.2
X Safety-Related	Non-Safety-Related	Rev. 0 Date
		Page 522 of

Client COMMONWEALTH EDISON CO.	Prepared by	Date 8/15/80
Project BYRON/BRAIDWOOD UNIT 1	Reviewed by D.E. Burris	Date 8-21-80
Proj. No. 4391-00, 4683-00	Approved by	Date

BB I SR

MECHANICAL COMPONENT SUPPORT DOCUMENTATION SHEET

CONTROLLING DESIGN LOAD: [X] BY/BR [ ] MARBLE HILL

DESIGN LOAD: ~~[illegible]~~ 704X1.5 = 1056#

REV. C

LOAD TYPE: [X] DESIGN, [ ] OPERATING, [ ] HYDRO, [ ] EMERGENCY, [ ] OTHER; ____________

ITEMS REVIEWED	SECTION OF REVIEW MANUAL USED	REMARKS
[X] NON-VENDOR CATALOG COMPONENT SUPPORT ELEMENT	5.0	A
[ ] STIFFENERS		
[X] CONNECTION OF NON-VENDOR CATALOG SUPPORT ELEMENTS	5.0	A
[ ] WELDS OF COMPONENT SUPPORT ATTACHMENTS		
[X] EXPANSION ANCHOR PLATES		C
[ ] OTHER:		

NOTATION FOR "REMARKS" COLUMN

"A" - Review Manual utilized with no supplemental hand calculations

"B" - Review Manual utilized with supplemental hand calculations which follow

"C" - Non-Standard: hand calculations follow

Form GQ 3.08.1 Rev. 2

SARGENT & LUNDY ENGINEERS CHICAGO	Calcs. For MECHANICAL COMPONENT SUPPORT NUMBER: M-1AF03003R REV. F SHT 2 of 2	Calc. No. 13.1.2
	X Safety-Related / Non-Safety-Related	Rev. 0 Date
		Page 523 of

Client	COMMONWEALTH EDISON CO.	Prepared by	P. S. Gurad	Date	8/15/80
Project	BYRON/BRAIDWOOD UNIT 1	Reviewed by	B.E. Burns	Date	8-21-80
Proj. No.	4391-00, 4683-00	Approved by		Date	

BB I SR

ITEM 9 2 10

$P_{V_1} = 2103 + 5 \times \cancel{41.9}\ 28.9 = \cancel{2313}\ 2248\#$ ✓

$P_{H_1}\ (28.9) = 57.8\#$

$P_{V_2} = 5(13) = 65\#$

$P_{H_2} = 2(13) = 26\#$

FORCE DUE TO MOMENT $= \dfrac{\cancel{2.313}\ 2.248 \times 7.33 + \cancel{.084}\ .058 \times 8.38 + .065 \times 6 + .026 \times 6}{2 \times 9} \times 1.0$

$= \cancel{1.21}\ 1.17^K$

FORCE DUE TO SHEAR FRICTION $= \dfrac{2.455}{4 \times .7} = 0.88^K$

NET FORCE $= \cancel{2.1}\ 2.06^K < 2.09^K$

REV. C we L.S.F 1.25 OK

CHECK ANCHOR BOLTS.

$P_{support} = 68\#$ $P_y = (5 \times 68) + (704 \times 1.5) = 1396\#$

$P_x = P_z = 2 \times 68 = 136\#$

$M_y = (136 \times 3.75) + (136 \times 7.5) = 1.530$ in-k

$M_z = 1396 \times 8.39 = 11.712$ in-k.

$T = \dfrac{1.530}{2 \times 10.5} + \dfrac{11.712}{9 \times 2} + \dfrac{.136}{4} + \dfrac{1.396 + .136}{2 \times .7} = 2.16^K$/bolt < 3.9 K/bolt OK.

Since bolts are weaker weld is Ok.

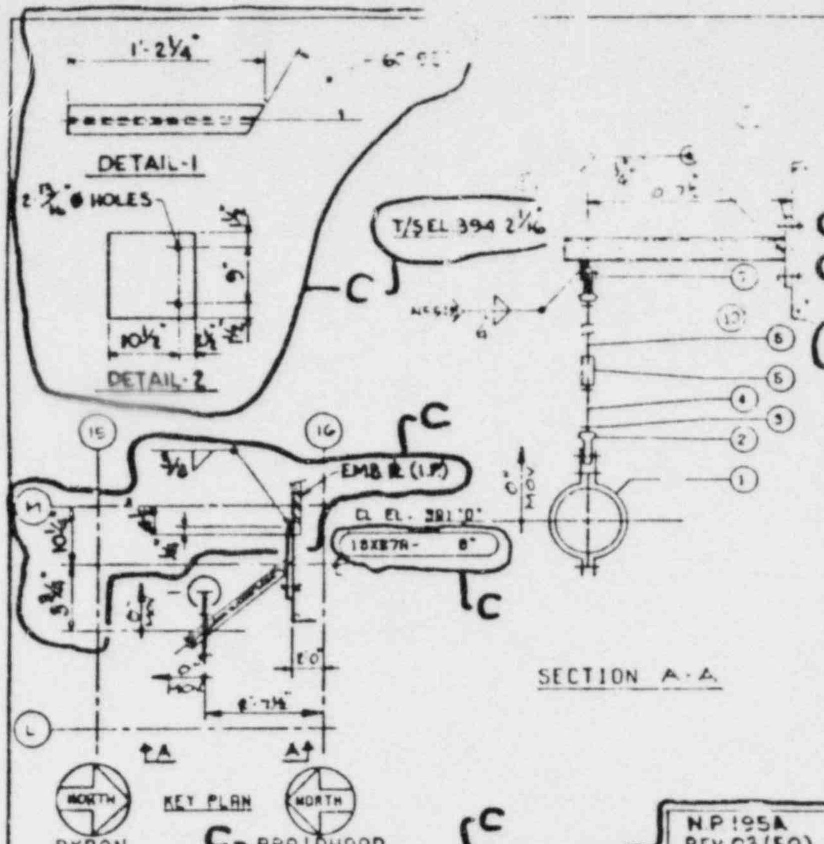
check Anchor Plate:

$M = 2.16 \times \dfrac{10.5}{2} \times 2 = 22.68$ in-k

$f_b = \dfrac{22.68}{\dfrac{12(.75)^2}{6}} = 20.16^{ksi} < 27$ksi OK.

ATTACHMENT C TO RESPONSE TO FINDING 3-7

Support Drawing M-1AF03009R,
Revision C



BILL OF MATERIAL

ITEM	QTY	FIG NO. OR PART NO.	DESCRIPTION
	1	——	COMPONENT SUPPORT ASSEMB.
1	1	295 N	8" PIPE CLAMP (13.1)
2	2	290	5/8" EYE NUT (1.2)
3	3	——	5/8" HEX NUT, SA-307 GR.B 1.3
4	1	140 N	5/8" X8¼" W/TFL (0.74)
5	1	230	5/8" TURNBUCKLE (1.0)
6	1	253 N	5/8" X 8¼" W/5¼"BE (0.76)
7	1	86	5/8" STEEL ATTACHMENT WITH LOAD PIN (1.5)
8	1	——	W4×13, (SA-36) 1'-2¼" LG. SEE DETAILS (15.44)
9	1	—	[illegible] PLATE [illegible] SEE DETAILS (X 6)
10	2	—	¾" [illegible] ANCHOR PER FORM [illegible] (1.88)
	1	——	BUNDLE & TAG FC-09
			TOTAL SUPPORT ASSEMBLY WT 67.53

WELD NO.	WELD PROCEDURE	EXAMINATION PROCEDURE

REFERENCE DRAWINGS	DWG. NO.	REV.	DWG. NO.	REV.	DWG. NO.	REV.	DWG. NO.	REV.
	M-850	L	S-690	AL	[illegible]	K	N.P 195A REV 03 (FO)	
	M-544-95	M	A-827	AB	[illegible]	M		

LOADS (LB)				
DESIGN	OPERATING	HYDRST TEST	EMERGENCY	FAULTED
704	420		828	

DRAWING RELEASE RECORD

REV.	DATE REL'SE	PREPARED	REVIEWED	APPROVED	FILM
A	3-26-79				
B	7-8-80				
C	11-22-81				

EMD REVIEW

EMD 034501

SUPPORT CLASS C

PURPOSE

F-2771 UNAFFECTED, L-2771 FOR FABRICATION, F-2739 FOR RECORD, L-2739 FOR CONSTRUCTION, PERFORMAL ANALYSIS. FOR F-9079 BY AS BUILT

WILLIAM CARL CLEFF 62 31871 REGISTERED PROFESSIONAL ENGINEER OF ILLINOIS

P.E. SEAL

BYRON/BRAIDWOOD STATION
UNIT-1
COMMONWEALTH EDISON COMPANY
CHICAGO, ILLINOIS

NUCLEAR SAFETY RELATED

NO SCALE		SUPPORT NO.	REV.
PROJECT NO. 4391-00 4685-00	SARGENT & LUNDY ENGINEERS CHICAGO	M-1AF03009R	C
		SHEET 1 OF 1	

Finding 3-16: U-bolt Analysis

"You are requested to describe the criteria employed for U-bolt design and confirm that they have been uniformly applied."

RESPONSE

The U-bolt review procedure is defined in the Westinghouse Byron Pipe Support Design Reference Manual, Byron 1 and 2 (Revision 0, dated 11-22-83), and is summarized below.

The criteria for evaluating U-bolts, for pipe sizes up to 2 inches, is based on results of tests on U-bolt samples. The vendor allowables, as published in applicable load capacity data sheets, have been shown to be conservative by results of failure tests for sizes of 1/2 inch to 2 inches for tension and side loads. Westinghouse has reviewed these test results and established allowable U-bolt loads by applying factors of safety to the test results. Safety factors of 4.0 (normal/ upset) and 2.0 (faulted) are used to define test-based allowables. This test data was collected for ITT Grinnell Figure 137H U-bolts. Since the Byron Project uses both the ITT Grinnell U-bolts and Elcen Figure 68A U-bolts, a comparison was made of U-bolts for various pipe sizes and shows the Elcen U-bolt to be identical in both material and bolt dimensions. The test data is therefore equally applicable to the Elcen U-bolt. For U-bolts for pipe sizes greater than 2 inches, the manufacturer's load capacity data sheets are the basis for the acceptance criteria.

The acceptance criteria for U-bolts for piping 2 inches and smaller, is based on test data. It is Westinghouse policy for the Byron Project to select U-bolts in the design phase using the vendor supplied load capacity data sheets. Allowables based on the test results discussed above may be used in the as-built reconciliation phase.

Westinghouse reviewed pipe support calculations to verify that the criteria was correctly applied. This review covers analyses that apply to 62% of the 820 U-bolts in the Unit 1 containment and applicable scope in the auxiliary building. In all cases, the reviewed U-bolts met the specified criteria.

Finding 4-1: Transverse Wall Load Criteria

"For the original Byron design, you are requested to describe how the design of peripheral walls, perpendicular to the transverse load, actually considered the loads coming on to the outside of the walls from wind and tornado (wind and differential pressure).

For a given wall loaded with a transverse wind load, describe how the transverse load at the center of a wall between two supporting orthogonal walls (shear walls) is carried out through the wall to the orthogonal walls which carry the shear force as in-plane shear in a normal shear wall concept. Describe how Sargent & Lundy determined the load capacity of the peripheral wall loaded transversely and converted these forces to stresses and then to reinforcing areas, if required. Describe how these same loads are addressed in the final load check. The use of diagrams to illustrate the details is recommended."

RESPONSE

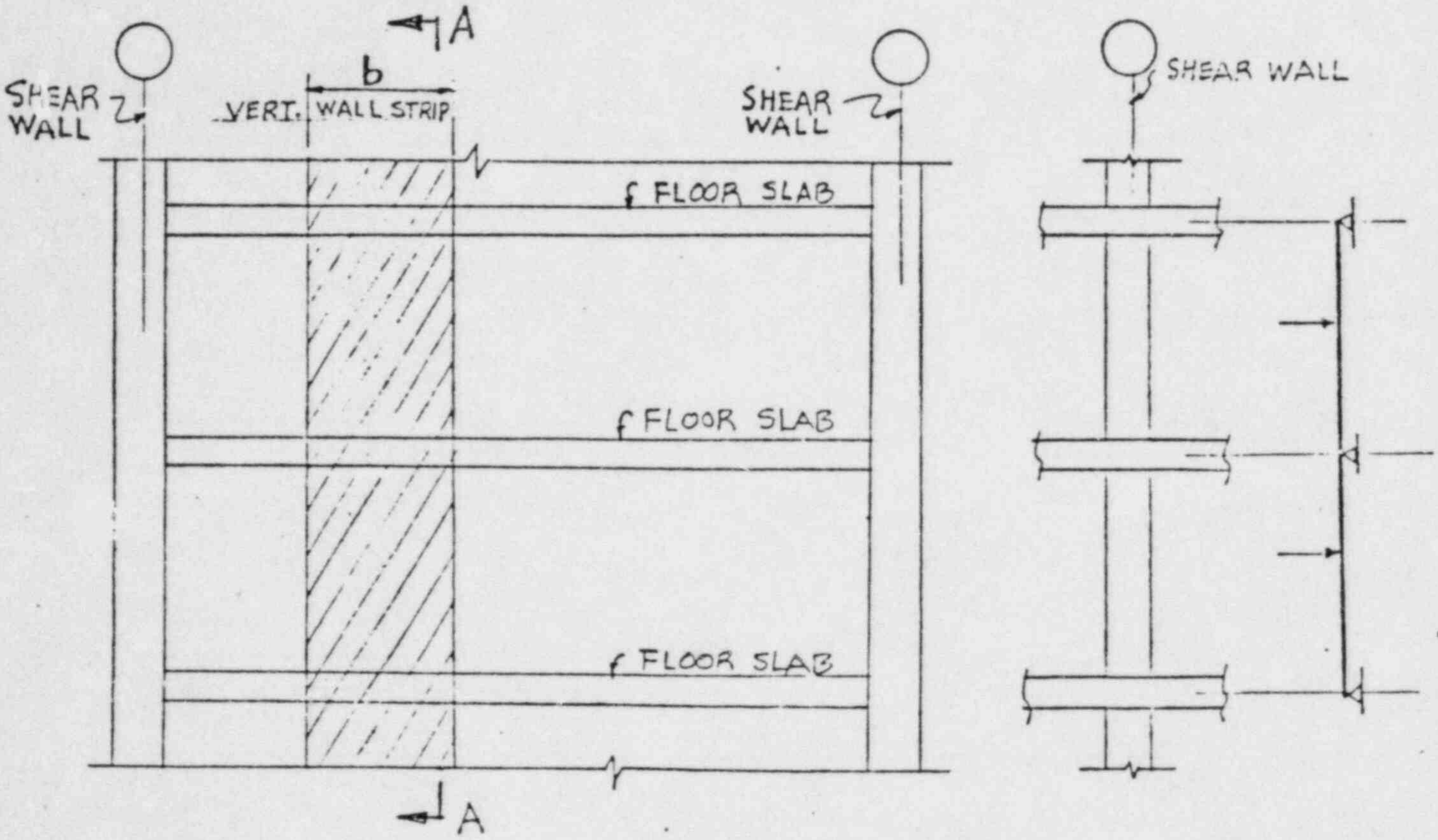
As noted in Subsection 11.6.2. of the Structural Project Design Criteria DC-ST-03 B/B (IDI Reference 4.31), the tornado loads in Safety Category I structures were obtained by static analysis utilizing the DYNAS lumped mass model. Various positions of the tornado were investigated to determine the maximum shear wall forces; and, for each position, the location and magnitude of the resultant tornado surface pressure was determined and transferred at the mass center at each elevation. The maximum shear wall forces thus obtained were compared to those forces due to seismic loading and the walls were designed for the most critical loading combination. Seismic load governed over wind and tornado in the shear wall design.

For a wall loaded with a given transverse load, the wall was analyzed using a "strip method" (i.e., the most critically loaded strip of the wall was isolated and treated as a "wall strip" which spans either vertically or horizontally and is supported at the intermediate floor slabs or adjacent walls as applicable, see Figures F4.1-1 and F4.1-2). In those cases where a vertical strip was used, the transverse load was applied to the wall strip and transferred to the supporting intermediate slabs. The load is then transferred through the slabs to the orthogonal shear walls and down to the foundation. In those cases where a horizontal strip was used, the transverse load applied to the shear wall was transferred directly to the supporting orthogonal walls.

For the case of transverse wind and tornado loading on walls, the controlling condition was tornado pressure plus a tornado-generated missile. The original wall design included an analysis which demonstrated that any peripheral wall was adequate to absorb this load and transfer it via a controlling strip to the lateral load resisting system.

The transverse flexural steel area required for each wall strip was calculated using the flexural strength provisions of Chapter 10 of ACI 318-71. The transverse shear friction steel area required was calculated using the shear-friction provisions of Chapter 11 of ACI 318-71. Transverse loads are addressed in a similar manner in the final load check.

FIGURE F4.1-1

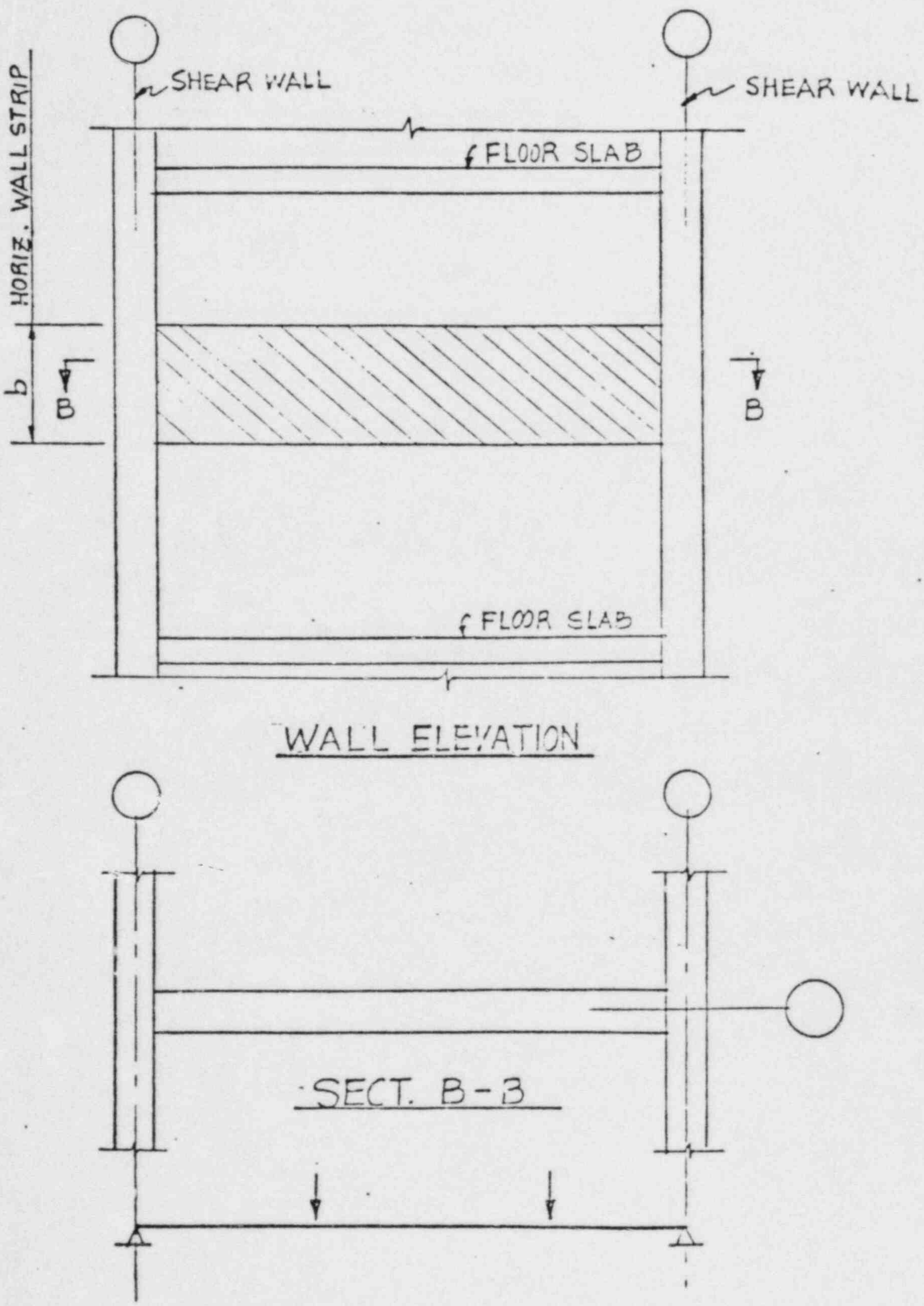


WALL ELEVATION

SECT. A-A

WALL STRIP: SPAN VERTICALLY

FIGURE F4.1-2



WALL STRIP: SPAN HORIZONTALLY

Finding 4-2: Shear Friction Method

"Our understanding of your response is as follows:

In the design of walls, the transverse shear stresses were computed from the transverse loads such as dynamic soils and water pressure, wind loads, tornado effects and horizontal seismic and compared to a value of V_c such as determined by Section 11.4 of ACI 318-71. If the actual shear stress was less than or equal to the allowable, no ties were added. If the actual shear stress was greater than the allowable stress, ties were added to carry the stress exceeding that carried by the concrete such as required in Section 11.6 of ACI 318-71.

In either case, when the vertical reinforcement was sized, the value of the actual transverse shear was combined with the in-plane shear and the resultant used as the total shear to be carried. The area of steel was computed from the resultant shear value by using the shear-friction concept.

The result is that in all cases, there is a margin in the vertical reinforcement relative to carrying in-plane shear loads.

You are requested to verify that this understanding is correct or, if it is not, describe how the actual design was executed. You are also requested to provide the details of how the transverse shear was combined with in-plane shear. The use of diagrams to illustrate details is recommended."

RESPONSE

Your understanding of our response is correct relative to the treatment of transverse shear stress, but differs somewhat relative to the method used for sizing vertical reinforcement.

In the design of the walls, the transverse shear stresses were computed from the transverse loads and compared to a value of v_c determined using Chapter 11 of ACI 318-71. If the actual shear stress was less than or equal to the allowable, no ties were added. If the actual shear stress was greater than the allowable stress, ties were added to carry the stress exceeding that carried by the concrete.

In the design of the vertical reinforcement, however, the steel area required for transverse shear loads was added directly to the steel area required for in-plane shear loads. Both the transverse shear steel area and in-plane shear steel area were determined using the shear friction concept.

Finding 6-3: Bases for Setpoints
Finding 6-7: Setpoint Accuracy Requirements
Finding 6-8: Basis for Reset Value

"IEEE 279-1971, which is invoked by 10 CFR 50.55a(h) requires 'A specific protection system design basis shall be provided .... The design basis shall document as a minimum .... (2) the generating station variables .... required to be monitored in order to provide protective actions .... (4) prudent operational limits for each variable .... (5) the margin .... between each operational limit and the level considered to mark the onset of unsafe conditions .... (6) the levels that, when reached, require protective actions, (7) the range of transient and steady state conditions .... throughout which the system must perform, (8) the malfunctions .... for which provisions must be incorporated to retain necessary protective actions and (9) minimum performance requirements including .... response times .... accuracies .... ranges .... of the magnitude and rate of change of sensed variables ....'

Our understanding of the intent of your responses is as follows:

1. For all safety-related instruments in the Sargent & Lundy scope, you will assure that documented bases have been provided as required by IEEE 279.

2. For cases judged to be complex, you will assure that calculations have been provided to support the selection of setpoints.

You are requested to indicate whether or not this understanding of your intent is correct. If it is not, please explain what is different in your intent."

RESPONSE

I. The instrument data sheets for safety-related instruments document the bulk of the design basis information and, in conjunction with other information denoted below, comply with our interpretation of IEEE-279 as follows (item numbers correspond to referenced sections above):

(2) The generating station variables that are required to be monitored in order to provide protective actions are determined during the design and review process of a particular system as documented on the system's Piping and Instrument Diagrams (P&ID), Control and Instrument Diagrams (CID) and the Logic Diagrams (LD). The documentation for this activity is thus contained on the P&ID's, CID's, and LD's as well as the instrument index and the instrument data sheets. (See attached

ple of an instrument data sheet, refer to encircled Item 2 in boldface type).

(4) The operational limits for each variable are documented on the Instrument Data Sheet. (See attached instrument data sheet, refer to encircled Item 4).

(5) The margin, with appropriate interpretive information, between each operational limit and the level considered to mark the onset of unsafe conditions are determined from existing calculations, design drawings and/or vendor supplied component design data. The margin on the data sheet is the relationship between the instrument's range and the setpoint. (See attached instrument data sheet, refer to encircled Item 5).

(6) The levels that, when reached, will require protective actions are determined using the design input discussed in Item (2) above and documented on the data sheet. (See attached data sheet, refer to encircled Item 6).

(7) The range of transient and steady state conditions of the power supply and the environment during normal, abnormal, and accident circumstances throughout which the system must perform are contained in the procurement specification which is referenced on the data sheet. In addition, the data sheet calls for seismic and/or Class 1E qualifications, all of which are documented in the EQ binders. (See attached instrument data sheet, refer to encircled Item 7).

(8) Malfunctions, accidents, or other unusual events (for example, fire, explosion, missiles, lightning, flood, earthquake, wind, etc.) which could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protective action, are reviewed at the time a particular condition is identified (see also Item (7) above and encircled Item 7 on instrument data sheet, for environmental conditions). For example, cubicle flooding was reviewed and the review of these flood levels (as related to instrumentation) was incorporated into the flooding calculation.

(9) The required instrument accuracy of an instrument is determined from an engineering assessment of information contained in the system calculations, design drawings and/or vendor supplied component design data. The setpoint accuracy required is then used in the review of vendor catalog information to establish

the instrument selection. The selected instruments are documented on the instrument data sheet by manufacturer and model number using vendor standard designs which envelop the system operating requirements. (See attached instrument data sheet, refer to encircled Item 9). Past experience has shown that response times and ranges of the magnitudes and rates of change of sensed variables have had little effect on the instrument selection. Therefore, these parameters are not reviewed unless a specific application is needed.

II. Complex setpoints for safety-related Sargent & Lundy instruments have been identified via a documented memo and calculations exist for these instruments.

<u>ATTACHMENT TO RESPONSE</u>
<u>TO FINDING 6-3, 6-7, 6-8</u>

Example of
Instrument Data Sheet

FORM ME-3.3.21 (7-76) APPROVED BY [illegible] DEPT. MGR.

INSTRUMENT NO.	(2)1PSL-AF051		(2)1PSL-AF055						REMARKS
SERVICE	Aux FW Pump (2)1A Suct		Aux FW Pump (2)1B Suct		2				Note 1 0.55 A Inductive @ 125 VDC (10 Amp DC switch, Switch MT4R)
FLUID	WATER		→						
PRESS. MAX. / TEMP. MAX.	100 PSIG	100 °F	→						
PRESS. NORM.	20-25 PSIG		→		4				
MEAS. ELEM. TYPE/MAT'L.	Diaphragm	Teflon	→					2	Note 2 One SPDT Switch For Each Function With Switches To Be Individually Adjustable. Switch Functions: ① Open Suct. Valves ② Trip Pump
RATING	225 PSIG		→						
PRESS. RANGE	30" Hg - 20 PSI		→		4&5				
SWITCH TYPE / RATING	SNAP	Note 1	→						
CONTACTS QTY./TYPE	Note 2	SPDT	→						
DIFF. ADJUST	FIXED		→		EXAMPLE				
OP. OR CL. INC.-DEC.@PRESS.	CL - Inc Vacuum	① 1.22" Hg vac ② 4.48" Hg vac	Note 4						
RESET@PRESS.	MFR STD		→						Note 3 Furnish With Bracket Suitable For Seismic Mounting.
HOUSING TYPE (ELEC.)	NEMA 4		→						
MOUNTING TYPE	SURFACE		→						
ACCESSORIES	Note 3		→		9				Note 4 ① CL-Inc Vacuum @ 1.22" Hg Vac ② OP-Inc Vacuum @ 4.48" Hg Vac
MANUFACTURER	United Electric		→				5&6		
MODEL NO.	J302-552		→						
NUCLEAR PLANT DATA — SAF. CL. / SEIS. CAT.	—	I	—	I	7				
QA	—		—						
ELECT. CLASS	1E		1E						
FURN. SPEC./INSTALL.SPEC.	F/L-2906	F/L-2906	→			7			

RELEASE RECORD

REV.	DATE REL'D.	PREPARED	REVIEWED	APPROVED	REV.	DATE REL'D.	PREPARED	REVIEWED	APPROVED
E	8-24-83	E. Hayward	M.S. Berg	[illegible]	Released For Records				
C	4-12-82	E. Hayward	[illegible]	[illegible]	Released For Records F/L-2906				
D	6-25-82	E. Hayward	M.S. [illegible]	[illegible]	Rev. [illegible] Released For Records				

PRESSURE SWITCHES

NUCLEAR SAFETY-RELATED

SARGENT & LUNDY CHICAGO

BYRON/BRAIDWOOD STATIONS

UNIT 1&2

COMMONWEALTH EDISON COMPANY

DATA SHEET NO. PS180

BPD

Unresolved Item 2-1: Diesel Engine Exhaust Pipe

"You are requested to describe the basis for determining that tornado missiles will not crimp the auxiliary feedwater pump diesel engine exhaust stack completely closed. Include a discussion of the potential for damaging the hinged cap in such a way as to incapacitate the pump."

RESPONSE

Sargent & Lundy performed an analysis to determine the effect of tornado missiles impacting the auxiliary feedwater pump diesel exhaust stack. Tornado missiles defined as Spectrum II missiles in Section 3.5.1.4 of NUREG-0800 were postulated. The calculations demonstrate that crimping the exhaust pipe due to a tornado missile impact results in a maximum 60% reduction in flow area at the roof interface. Missile impact on the 1/8-inch thick aluminum weather cap will not affect the flow area since the cap will either be destroyed or blown out of the stack by the exhaust pressure. The 60% reduction in flow area of the exhaust stack at the roof interface will not incapacitate the auxiliary feedwater pump diesel.

Unresolved Item 4-2: Top Reinforcing for Slabs

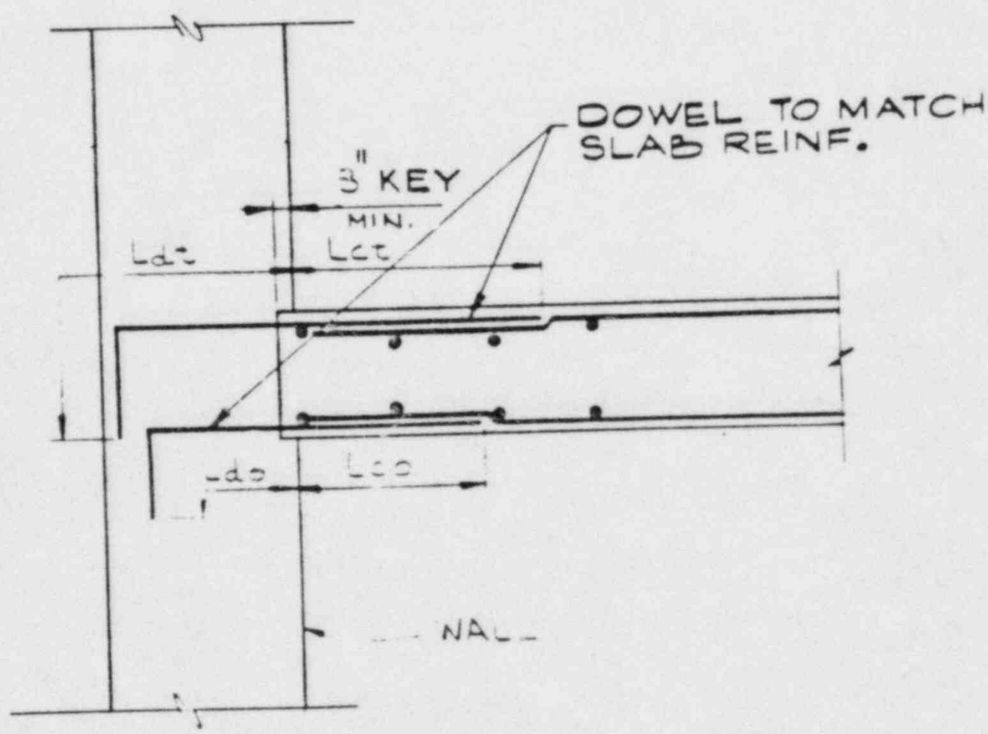
"The design procedure outlined in the response, if applied for all slab designs on the Byron project, would yield conservative steel areas (bottom steel) for midspan positive moments. You are requested to indicate whether this concept was used throughout the plant. Indicate if the procedure described in the original answer for supplying negative steel (top steel) at each slab boundary was used throughout the project and if so what portion of the maximum moment for a simply supported case was provided in negative moment capacity at the boundaries. Indicate, by providing detailed references to written documents, how these project-wide concepts (if used) were provided to individual designers in the way of instructions or procedures. If no project-wide concept was applied, indicate what technique was used in providing slab reinforcing based on varying boundary conditions."

RESPONSE

As noted in our original response, negative moment steel equal to that at the continuous support was provided at the junction to the wall of slab 4AB53. This is a conservative design approach which was used for all slabs supported by walls. This typical detailing is shown as Slab Support Types 5 and 6 (see Figures U4.2-1 and U4.2-2) on Structural Drawing S-473 which was included in project Specification F/L 2722 and initially released to the appropriate contractors on August 7, 1974.

These standard details are specified for construction at all slab-to-wall junctions and, because they are standard details, no other reinforcing arrangements could have been used. Thus, their use is a project-wide concept and did not depend on the judgment of the individual designer involved.

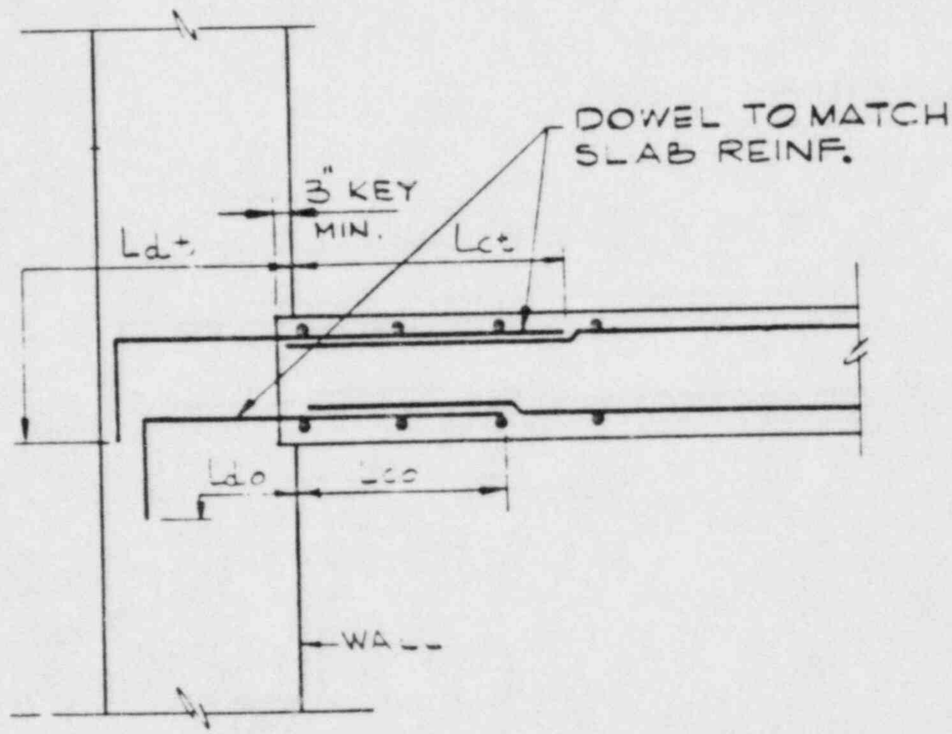
FIGURE U4.2-1*



SLAB SUPPORT TYPE-5

*Taken from S&L Drawing S-473.

FIGURE U4.2-2*



SLAB SUPPORT TYPE-6

*Taken from S&L Drawing S-473.

Unresolved Item 6-2: Pressure Switch Qualification

"When review of the pressure switch qualification data is complete, you are requested to provide a description of the basis for acceptance. If qualification by similarity with the tested switch is used, describe the rationale for using similarity."

RESPONSE

The qualification program for pressure switches 1PSL-AF051 and 1PSL-AF055 is described in the following discussion.

The original pressure switch specified for this application was United Electric Model J-302-S156, which is a metal bellows type sensor. Later, due to operating requirements, this switch was changed to Model J-302-552, which is a teflon diaphragm type sensor. Since the test program for Model J-302-S156 was in progress, it was decided to continue the test and evaluate the acceptability of the report upon receipt.

Since the time of the IDI, the report has been received, reviewed, and found to be unacceptable for qualification of Model J-302-552. Due to internal mechanism differences between the two switch models, seismic testing of Model J-302-552 is required and in progress.

Since the switches are located in a mild environment, the environmental qualification will be by a combination of similarity between the tested and supplied switch models for parts that are identical, and a material analysis for parts that are different.

<u>Section II</u>

Response to NRC Letter Dated May 14, 1984

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."

RESPONSE

The turbine building crane was not included in the subject walkdown since only the non-safety-related equipment in the immediate vicinity of the auxiliary feedwater diesel air intake line was reviewed. The diesel air intake line is located at grade elevation 401 feet while the turbine building crane is located above the main operating floor (at approximately elevation 500 feet). The turbine building crane rail girders are designed to withstand SSE loads. The bridge is normally parked at the south end of the turbine building during power operation, corresponding to a horizontal distance of nearly 300 feet from the diesel air intake line. In the unlikely event that the bridge fails during an SSE, the auxiliary feedwater diesel air intake line will not be affected.

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."

RESPONSE

(1) Project Instruction PI-BB-58, "Incorporation of Electrical Schematic Control Diagram Changes Into Control Logic Diagrams," has been written to formalize the engineering practice described in the previous response to this finding. This instruction requires that changes to schematics are reviewed against the logic diagrams and, if applicable, requires that logic diagrams are revised. In addition, Sargent & Lundy is conducting a review of the logic diagrams against the schematics. This review is scheduled to be completed in mid-July 1984.

(2) The following drawings are referenced in FCR-21265 with the respective systems requiring revisions:

1-4030 OG01	OG (Off-Gas)
1-4062B	WO (Chilled Water)
1-4062C	WO (Chilled Water)
1-4062E	Bill of Material
1-4062G	WO (Chilled Water)
1-4062H	WO (Chilled Water)
1-4600E	FW (Main Feedwater)
1-4030 SX01	SX (Essential Service Water)
1-4611B	AP (Auxiliary Power 480 V and above)
2-4045B	EH and TG (Turbine EHC and Turbine Generator Auxiliaries)

Finding 2-8: Missing Calculation For Containment Spary

We believe that the FSAR statements are design bases and are licensing commitments. Our letter dated March 23, 1984 (page 1 of enclosure) requested you to describe the provisions in your review porgram (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.

RESPONSE

We concur with the IDI Team that the FSAR statements represent licensing commitments. The Independent Design Review will address the issue concerning calculations, as described in our response to the General Item of NRC letter dated March 23, 1984. (The response to the General Item is contained in Section I of this attachment.)

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."

RESPONSE

The "Documentation Package for Equipment Status Display System; Byron and Braidwood Nuclear Power Stations - Units 1 and 2," is now complete and available for review. This document contains final design information for the ESD system.

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

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

1. 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."

RESPONSE

Piping analysis is a design tool for providing a basis for selection of support hardware and evaluation of piping stresses. The EG&G analysis method considers pipe support stiffness values whereas the Sargent & Lundy analysis method considers pipe support to be infinitely rigid.

Both EG&G and Sargent & Lundy methodologies are acceptable means for analyzing a piping system. Large margins and considerable conservatisms do exist in both approaches. These are demonstrated and discussed in detail in technical literature such as NUREG/CR-3526, "Impact of Changes in Damping and Spectrum Peak Broadening on the Seismic Response of Piping Systems." In addition, it is not reasonable to take extreme differences resulting from the two methodologies and review the results of one analysis method (EG&G) against the other (S&L) and to suggest that the uplift limits should be increased for rod hangers and rest type supports.

Sargent & Lundy's support modeling practice was discussed in detail with the staff of the NRC Mechanical Engineering Branch on August 19, 1983 and it was found to be acceptable.

Attached for your reference are two memos by the NRC Mechanical Engineering Branch and the Division of Licensing which state that the approach used by Sargent & Lundy is acceptable.

Reference 1, "Memorandum from D. G. Eisenhut" states:

> "Based on our review of the Sargent & Lundy design practices, the staff concludes that the method used by Sargent & Lundy for the modelling of the pipe supports in the piping design analyses together with the engineering rationale presented in some detail in the attachment provides an adequate basis for the calculation of piping stresses and support loads."

Reference 2, "Memorandum from D. Terao" states:

> "It is the staff's belief that S&L's design practice of modelling supports as infinitely rigid is acceptable when used with sound engineering judgement."

We believe that we have addressed your concerns in regard to this item.

ATTACHMENT IN RESPONSE TO
UNRESOLVED ITEM 3-1

1. NRC Memo from D. G. Eisenhut to R. L. Bangart, dated October 31, 1983.

2. NRC Memo from D. Terao to R. J. Bosnak, dated September 19, 1983.

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

OCT 3 1 1983

MEMORANDUM FOR: R. L. Bangart, Director
Division of Vendor & Technical Programs
Region IV

FROM: Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: EVALUATION OF SNUBBER PROBLEMS AT SARGENT & LUNDY

This is in response to your memorandum of June 17, 1983, to R. H. Vollmer on the above subject. NRR was requested to review the adequacy of the piping design criteria being used at Sargent & Lundy relative to the six concerns identified in your memorandum. The concerns are relevant to the La Salle, Zimmer, Fermi-2, Clinton, Byron, Braidwood, and Marble Hill facilities.

On August 19, 1983, the staff visited the Sargent & Lundy offices and reviewed the design practices used by Sargent & Lundy in their piping system analyses and support design. Attachment 1 is the trip report which summarizes the details of the six concerns as discussed during the meeting.

Based on our review of the Sargent & Lundy design practices, the staff concludes that the method used by Sargent & Lundy for the modelling of the pipe supports in the piping design analyses together with the engineering rationale presented in some detail in the attachment provides an adequate basis for the calculation of piping stresses and support loads. However, practical experience and sound engineering judgement should always be exercised by any design organization to assure that assumptions used in the piping system analyses are not invalidated in subsequent support design and actual installation. It is incumbent upon anyone reviewing design analysis procedures to assure that the mechanical engineer performing the piping system analysis has adequately communicated with the civil-structural engineer providing the piping system supports. Our experience substantiates the fact that breakdowns in this interface frequently result in conditions not meeting those stipulated in the piping system design specification. With respect to item 4 and the fact that auxiliary steel for small bore supports has no criteria for stiffness, there is an implied stiffness even if not specified based on past experience with this type of support and pipe which have been correlated with several analyses. Therefore, marked deviations in support characteristics should be questioned but the fact that stiffness is not explicitly specified is acceptable.

We trust that the information provided is responsive to your concern, and the NRR responsibilities under TIA No. 83-65 have been completed.

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

cc: R. Starostecki, R-I
J. Olshinski, R-II
R. Spessard, R-III
T. Bishop, R-V
A. Bournia
L. Kintner
D. Lynch
G. Harrison
J. Stevens
L. Olshan
P. O'Conner
R. Wessman
R. Vollmer
R. Bosnak
D. Terao

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

SEP 1 9 1983

MEMORANDUM FOR: R. J. Bosnak, Chief
Mechanical Engineering Branch, DE

FROM: D. Terao
Mechanical Engineering Branch, DE

THRU: HLB H. L. Brammer, Section Leader
Mechanical Engineering Branch, DE

SUBJECT: TRIP REPORT SUMMARY ON SARGENT & LUNDY SNUBBER MODELLING DESIGN PRACTICES

On August 19, 1983, the staff met with representatives of Sargent & Lundy (S&L) to discuss the design practices used by S&L in their modelling of snubbers in piping design analyses. The discussion focused on the six concerns identified in a memorandum from R. L. Bangart (RIV) to R. H. Vollmer dated June 17, 1983. The six concerns as discussed in the meeting will be addressed in detail in the following paragraphs. The formal S&L responses to the six concerns are included in this memorandum as Attachment 1. The meeting attendees are included in Attachment 2.

Evaluation of S&L Design Practices

1) "With one exception, LaSalle, no project has a minimum snubber stiffness specified and no project has a maximum gap (travel before lock-up) specified."

The concern appears to be that if no minimum snubber stiffness is specified by the purchaser (Sargent & Lundy), then the piping design analysis might incorrectly assume that the piping is adequately restrained while the snubber could be very flexible. Similarly, the second part of the concern appears to be that if no maximum gap is provided in the design specifications, then the snubbers might not lock-up adequately to perform its intended function to restrain the piping system.

Sargent & Lundy's response indicates that they do not specify snubber stiffnesses and gaps in their design specifications (except for the LaSalle project) because this information is always reviewed by S&L in the manufacturer's literature and test reports prior to accepting a manufacturer as the snubber supplier. The review performed by S&L is to assure that the snubber stiffnesses for the selected vendor is adequate for the snubber to perform its intended function of restraining the piping system. For LaSalle, S&L specified snubber stiffnesses (obtained from Pacific Scientific literature) because when the design specifications were written, they knew that Pacific Scientific would be the sole source for mechanical snubbers. The response is acceptable.

2) "No actual stiffness values for supports or snubbers were used in the pipe stress runs for any project in S&L's scope. Such supports/snubbers are assumed to be pinned joints and infinitely stiff in the direction of the force."

The concern is apparently related to Item 1 in that S&L models snubbers in their piping analysis as an infinitely rigid member which acts as a force restraining the pipe from movement in the direction of the snubber. Because snubbers do have a finite stiffness value and some lost motion (or travel before lock-up), the concern appears to be that piping analyses could be incorrectly calculating piping stresses and support loads by assuming infinitely rigid snubber stiffnesses.

The modelling of infinitely rigid versus flexible supports in piping analyses is a controversial subject throughout the industry. Many technical papers and parametric studies have been written which specifically address this issue. However, it is not obvious that any paper or study conclusively states that supports should always be modelled as either rigid or flexible. Indeed, there are many uncertainties in the design analysis of piping systems. In the (draft) report by E. C. Rodabaugh entitled, "Sources of Uncertainties in the Calculation of Loads on Supports of Piping Systems." (May 1983) prepared for NRC/RES, the significance of flexibilities in piping support characteristics is deemed to be small. However, the significance of gaps and non-linearities for dynamic loads is shown to be uncertain. Although some of the other architect engineering firms do consider support stiffnesses in their piping system model, the effect of support non-linearites, are usually not explicitly factored into their model.

From a dynamic analysis standpoint, the use of flexible supports would tend to decrease the modal frequencies of the piping system response. Conversely, the use of infinitely rigid supports would tend to increase the modal frequencies of the piping system response. Depending on the amplified acceleration response spectra used for the specific piping system, the piping response will either increase or decrease. Broadening the peaks of the response spectra can conservatively "stabilize" the effect of minor modal frequency shifts. For Sargent & Lundy plants, the peak broadening values are shown in Attachment 3 and appear to be conservative.

In further discussions with S&L, it was found that although most piping systems are modelled with the piping supports as infinitely rigid (because the S&L computer program PIPSYS was initially developed in that manner), the computer program does have the capability to model the piping supports with finite stiffness characteristics. S&L stated that they do model more detailed piping supports in their piping model when it is deemed necessary. For example, the LaSalle suppression pool piping systems were modelled with the more detailed piping support characteristics in order to calculate the effect of pool impact and drag loads on the supports.

S&L's belief is that developing the capability in their computer program to allow inputting the stiffness values would not be difficult, however, very little improvement in accuracy would be gained. S&L's position is that "the piping analysis is a design tool for providing a reasonable and prudent basis for selection of support hardware and evaluation of piping stress rather than a precise prediction of actual stresses, and that there is no significant advantage or compelling reason for increasing the complexity of (their) models by including support stiffnesses."

It is the staff's belief that S&L's design practice of modelling supports as infinitely rigid is acceptable when used with sound engineering judgement. The adequacy of the design procedure to model piping supports as rigid rather than flexible appears to be reasonable and acceptable provided the supports perform their intended function (which is to restrain the piping). However, if the supports are allowed to deflect excessively under dynamic loading conditions, then the assumption of modelling rigid supports in the piping analyses could result in underestimated loads and stresses in the piping system. For S&L piping systems, the staff believes that the large bore piping supports are designed with adequate stiffnesses to perform their intended function. However, for small bore piping systems, the staff has developed a concern with respect to the support functionality. (See Item 4 below).

3) "LaSalle, Zimmer, Fermi 2, and Clinton have no FSAR commitments to use actual stiffness values for supports or snubbers. Byron/Braidwood and Marble Hill have commitments in their FSAR's to include actual calculated stiffness values in the model of the piping systems for Class 1 piping. All Class 1 piping for these plants, however, is in the scope of the nuclear steam system supplier.

   Our inspection focused on Class 1 piping but it is believed that Class 2 and Class 3 piping is handled in a similar manner."

   The design practice to use infinite stiffness values for piping supports is used for all S&L piping analyses (Class 1, 2, 3 and non-ASME piping). The design adequacy is addressed in Item 2 above.

4) "The only requirement relating to stiffness of auxiliary steel pipe supports is 1/4 inch maximum deflection under load for large bore pipe supports and no requirement at all for small bore pipe

supports. This deflection requirement excludes deflections of primary steel to which the auxiliary steel is attached."

The staff identified two concerns in this item. The first concern is that the auxiliary steel pipe supports for large bore pipe supports can potentially deflect up to ±1/4". Secondly, the supports for small bore piping has no maximum allowable deflection criteria at all. The concern is that excessive pipe deflections might result in an underestimation of loads and stresses in the piping system. (See Item #2).
S&L provided the two referenced papers in their response to this item. The first paper, "Impact of Structural Steel and Snubber Gaps on the Dynamic Behavior of Piping," by T. G. Youtsos dated 6/29/81 was used as the S&L justification for allowing ±1/4" deflection in the auxiliary steel supports. S&L further clarified that the ±1/4" allowable deflection included the total auxiliary steel pipe support deflection due to both pipe loading and self-weight excitation loading. S&L stated that the common industry practice of ±1/16" or ±1/8" deflection allowable considers only the pipe loading (and neglects the self-weight excitation). S&L believes that in this respect their ±1/4" deflection criteria is equivalent to or better than the industry practice. S&L further stated that for large bore piping systems where the dynamic load is the governing design load for the supports, the auxiliary steel design is controlled by stress and not by deflection. Thus, S&L stated that the auxiliary steel deflections (when stress-controlled) aré usually much less than ±1/4".

The second referenced paper in their response, "Small Piping Restraint Stiffness Study," by J. E. Hart dated 3/15/82 was used as the S&L justification for having no maximum allowable deflection limits for small bore piping supports. The paper shows that for a sample small bore piping system in the LaSalle facility, the piping stresses for the three cases analyzed (including no deflection criteria) is acceptable.

The staff reviewed the S&L responses and justification for their auxiliary steel pipe support deflection criteria. For the large bore piping systems, the staff believes that the S&L deflection criteria of ±1/4" is acceptable. There are no deflection limits in the codes used for auxiliary steel support design. Because standard industry practice does limit auxiliary steel deflections to ±1/16 or ±1/8" for pipe loadings only, the S&L limit of ±1/4" for both pipe loadings and self-weight excitation appears to be as reasonable a criteria as standard industry practice with a further limitation on the self-weight excitation which is commonly ignored. The acceptability could also be verified by reviewing the actual deflections calculated for the auxiliary steel supports for large bore piping. S&L stated that RIV has reviewed the deflections and found that the deflections are generally small (±1/16").
Therefore, the staff concludes that the adequacy of the piping

design is acceptable based on the S&L large-bore pipe support design practice described above and on the assumption that auxiliary steel large bore pipe supports are stress-controlled and result in small deflections. However, for small-bore piping, the staff does not believe that the referenced parametric study provides an adequate basis for allowing no deflection limits for neither pipe loadings nor self-weight excitation for small-bore piping supports. Education, experience, and sound engineering judgement when used with the applicable codes and standards is usually the standard industry practice for small bore pipe support design. The staff's concern is that the above referenced parametric study was apparently developed by S&L piping designers for S&L support designers and can be misinterpreted as a substitute for good design practices.

5) "A study was completed in June 1981 (Sargent & Lundy File Number EMD-031200), in which it was concluded, among other things, that piping response is very sensitive to dynamic restraint stiffness and gap size."

S&L's response to this item indicates that the conclusion that "piping response is very sensitive to dynamic restraint stiffness and gap size" was a statement that focused on one particular parameter in the study. However, when the overall effect of this one parameter was combined with the effects of other parameters, the standard S&L design procedures were concluded to be adequate in predicting the maximum piping stress and support loads.The S&L response is acceptable.

6) "Mechanical snubbers being procured on various S&L projects are allowed a minimum of 0.2 inches of movement for each second of loading (for any load from zero to rated load). S&L was unable to explain the consequence of this characteristic relative to the accuracy of their pipe stress model and/or computer output."

The S&L response to this item indicates that the above quoted allowable snubber movement of "a minimum of 0.2 inches of movement for each second of loading", is incorrect. Instead, the acceptance criteria is that the time for the snubber to extend 1 inch shall be 0.59 seconds minimum for a unidirectional (compression or tension) loading. This test acceptance criteria is part of Pacific Scientific Acceptance Test Report No. IT-533 (paragraph 6.3).

Pacific Scientific mechanical snubbers are designed to limit piping acceleration to a fixed threshold level. Thus, the piping is restrained for dynamic cyclic loadings (e.g. seismic) and sudden shock loads. However, the Pacific Scientific mechanical snubbers do not lock-up in a fixed position by a sustained, uninterrupted force. It is this sustained, uninterrupted force for which the snubbers are tested as described above to assure a constant acceleration value.

Sargent & Lundy stated that Pacific Scientific mechanical snubbers are not used by S&L to restrain piping subjected to sustained dynamic loadings and, thus, believes that there is no concern identified. The response is acceptable.

D. Terao

D. Terao
Mechanical Engineering Branch
Division of Engineering

Attachments:
As stated

cc: R. Vollmer, DE
J. Knight, DE
E. Sullivan, DE
G. Lear, DE
T. Bournia, DL
C. Hale, RIV
D. Chamberlain, RIV
P. Sears, RIV
H. Brammer, DE
H. Shaw, DE

Attachment 1

RESPONSES TO ITEMS IN LETTER TO NRR

ITEM 1

Issue:

a) No minimum snubber stiffness is provided in the specifications

b) No maximum gap is provided in the specifications

Response:

a) Although stiffnesses are not generally contained in the specification, the values are commonly available in manufacturer's literature and test reports, which are reviewed by S&L prior to acceptance.

b) Although maximum gaps are not provided in the specifications, the allowable gaps are published by PSA and all snubbers are tested prior to shipment from the factory. The maximum gaps are .030 or .040 inches depending on the snubber model, and the actual gaps found from testing are significantly lower than these values. Furthermore, the test reports are reviewed by S&L prior to acceptance.

ITEM 2

Issue: Stiffness values for supports are not included in the piping analysis models.

Response: Modeling snubbers as rigid supports has been standard industry practice. Several papers have been written attempting to evaluate the impact of including support stiffnesses in the model, two of which are referenced below.

The first concluded that for spectra with high frequencies, such as spectra for hydrodynamic loads, modeling the support stiffenesses would not be conservative and the stiffnesses would have to be artifically increased. It also concluded that modeling rigid supports provides reliable results and conservative support loads.

The second compared results of stresses and loads from analyses with and without stiffnesses and found in general no significnat changes in stress or support loads.

In light of current technical literature as well as past industry and S&L practice, it is S&L's position that the piping analysis is a design tool for providing a reasonable and prudent basis for selection of support hardware and evaluation of piping stress rather than a precise prediction of actual stresses, and that there is no significant advantage or compelling reason for increasing the complexity of our models by including support stiffnesses.

References:

1) "Distortion of Spectral Characteristics Due to Modeling of Piping Systems, " Shulemcorich, Studnicka, and Michejda.

2) "Response Evaluation of a Piping System Supported by Linear and Hardware Dependent Models of Snubbers" M. A. Pickett

ITEM 3

Issue: In most cases there are no commitments in the FSAR to use actual stiffness values in the piping analysis, although the NSSS has made such commitments for Class 1 piping on two plants.

Response: At the time the FSAR's were written, there was no requirement to specifically state whether snubber stiffnesses would be incorporated in the analysis. However, NRR has been aware of S&L's practice over the years and has granted licenses for plants where piping was analyzed in this manner.

ITEM 4

Issue: The stiffness criteria used for auxiliary steel does not account for primary steel movements and is based on deflection only; auxiliary steel for small bore supports has no criteria for stiffness.

Response: S&L has compared results of analyses in which support gaps and stiffnesses, auxiliary steel, and primary steel have been included in the model with results from conventional models. Many gap sizes, including 1/4", were input. The results indicated that the standard procedure of analysis was adequate in predicting piping response.

For small bore piping, a separate study was done which included auxiliary steel designed to 1/4" deflection criteria and steel designed without it. The results showed the stresses to be well below allowables.

References:

1) "Impact of Structural Steel and Snubber Gaps on the Dynamic Behavior of Piping," T. G. Youtsos, 6/29/81.

2) "Small Piping Restraint Stiffness Study," J. E. Hart, 3/15/82.

ITEM 5

Issue: A study was conducted that concluded that piping response is sensitive to dynamic restraint stiffness and gap size.

Response: The conclusion quoted is taken from Reference 1 of Item 4 above. The conclusion is a statement that focuses on one particular parameter in a study that considered and included in the modeling many other parameters as well. The same report evaluates the effect of <u>all</u> of these factors and goes on to conclude that current modeling practices are adequate.

ITEM 6

Issue: The effects of a .2 inches/second rate of snubber deflection are not considered in the analysis.

Response: The source of the .2 inches/second rate referenced in Item 6 is not clear, but from discussions with Region IV personnel during the audit, this is apparently referring to paragraph 6.3 of the PSA Acceptance Test Report No. IT-533. This paragraph defines the procedures for the acceleration test of the snubbers and states, "With the required pressure applied to the unit, the time as recorded by the timer for the snubber to extend 1" shall be .59 sec. minimum."

This test verifies that the snubber will brake at the specified g-level. The one inch deflection occurs because the test load is applied in only one direction and the snubber mechanism does not "lock", but "brakes" the snubber. The load the snubber is subjected to in service is an oscillatory load in both directions, consequently, a one inch deflection of the snubber will not occur when a dynamic load is applied.

Since this condition is only to check the proper braking level of the snubber, and does not occur in service, this has no impact on piping analysis.

SUMMARY

Five of the six items above deal with the issue of including snubber stiffnesses in piping models. The referenced papers and S&L studies indicate that snubber stiffness is only one of many parameters that are not included in piping models, and that when all parameters are modeled and results compared, rigid modeling is shown to be adequate.

Furthermore, piping analysis is not intended to be a precise prediction of actual stresses, but a design tool to provide a reasonable basis for selecting restraint hardware and evaluating the pipe. Focusing on only one parameter, e.g., support stiffnesses, does not consider the overall relationship between the pipe modeling and its actual construction.

Attachment 2

8/19/83 Support Stiffness Modelling on Piping Analyses

Name	Affiliation
D. Terao	NRC/NRR
G. Kitz	S&L
S. D. Killian	S&L
R. Johnson	S&L
C. N. Krishnaswami	S&L
P. Sears*	NRC/RIV
T. Longlais	S&L

*Part time

Attachment 3

SARGENT & LUNDY ENGINEERS CHICAGO	Calc. For	Calc. No.
	Safety-Related / Non-Safety-Related	Rev. / Date / Page / of

Client	Prepared by CNKrishnaswamy Date 8-19-83
Project	Reviewed by / Date
Proj. No. Equip. No.	Approved by / Date

% SPECTRAL PEAK WIDENING

	Seismic		Pool Dynamic
	V	H	
Fermi	±20%	±10%	By Others
LaSalle	±20%	±10%	±20%
Zimmer	±20%	±10%	±20%
Byron + Braidwood	±10% + ±10%	±10% + ±10%	Envelope of Byron & Br. results in effective widening of ±20% in a majority of spectra
Marble Hill	±15%	±15%	N.A.
Clinton	±15%	±15%	±15%

Ref. 1. SRP 3.7.1. II

2. NRC Auditor's Qn through E.R. Johnson, G. Kitz, D/8-19-83

3. NRC Qn./Ans. on LaSalle Project

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

August 30, 1984

Mr. 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
NRC Docket Nos. 50-454 and 50-455

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

(b): June 19, 1984 letter from Cordell Reed to R. C. DeYoung.

Dear Mr. DeYoung:

This letter supplies revised responses to two findings and one unresolved item which were identified during the Byron integrated design inspection. These responses have been revised to address issues raised by NRC inspectors at a meeting on July 30, 1984 at Sargent and Lundy.

Enclosed are revised responses to findings 4-1 and 4-2 and unresolved item 4-2. These versions supersede the responses previously supplied in references (a) and (b).

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,

D. L. Farrar
Director of Nuclear Licensing

lm

cc: J. G. Keppler

Enclosures

FINDING 4-1: TRANSVERSE WALL LOAD CRITERIA

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

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

REVISED RESPONSE

This revised response has been prepared subsequent to a July 30, 1984, meeting between the NRC (IE) and Sargent & Lundy. As a result of the information exchanged at this meeting, this revised response supersedes the response to Finding 4-1 sent to the NRC on December 30, 1983, and the response to additional questions on Finding 4-1, which was sent to the NRC on June 19, 1984, by letters from Commonwealth Edison Company to Mr. R. C. DeYoung.

The Structural Design Criteria are organized such that all loadings are reviewed in one chapter (Chapter 10) and specific items such as shear wall design are outlined in the following chapters. Chapter 11 of the Structural Design Criteria on shear walls discusses the application of the controlling loads for shear wall design since these are of primary concern. Although design for wind and tornado pressure are not listed in Chapter 11, they are listed in Chapter 10 as being applicable to shear walls. Chapter 11 of the Structural Design Criteria will be updated to state that all loadings in Chapter 10 must be considered in shear wall design for in-plane and out-of-plane loading and that the loads listed in Chapter 11 are those loads which generally control shear wall design.

Transverse loads due to the combined effect of tornado wind pressure and tornado-generated missiles were considered in the original plant design (Calc. No. 7.12.7) using energy balance design techniques. It was demonstrated that the critical exterior shear wall panel was adequate for tornado-generated missiles and withstanding the tornado wind pressure.

Subsequent calculations have verified that out-of-plane seismic and tornado pressures are not controlling loads and that the structure is adequate for these loads. The transverse shear stresses in the walls due to these loads are well below the allowables in Section 11.4 and 11.6 of the ACI Code 318-71.

FINDING 4-2: SHEAR FRICTION METHOD

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

REVISED RESPONSE

This revised response has been prepared subsequent to a July 30, 1984, meeting between the NRC (IE) and Sargent & Lundy. As a result of the information exchanged at this meeting, this revised response supersedes the response to Finding 4-2 sent to the NRC on December 30, 1983, and the response to additional questions on Finding 4-2, which was sent to the NRC on June 19, 1984, by letters from Commonwealth Edison to Mr. R. C. DeYoung.

The Sargent & Lundy Structural Project Design Criteria has been clarified to more explicitly show that the shear wall design was performed in accordance with Sections 11.4 and 11.6 of ACI 318-71. The relevant sections of the Criteria are as follows:

A. Concrete Design Check

1. Nominal shear stress due to out-of-plane shear shall not exceed $2\sqrt{f'_c}$ (except as noted in the ACI Code for net compression or tension on the wall). When this value is exceeded, shear stirrups shall be provided in accordance with ACI 318.

2. Nominal shear stress due to in-plane shear shall not exceed $10\sqrt{f'_c}$.

3. Combined shear stress shall not exceed $12\sqrt{f'_c}$, where combined shear stress =

$$\left[(\text{In-plane shear stress})^2 + (\text{Transverse shear stress})^2\right]^{1/2}$$

A review of shear wall calculations conducted during the July 30, 1984, meeting between the NRC and Sargent & Lundy showed that the Project Criteria, as outlined above, had been followed. (The calculations examined were drawn from Calculation Books 7.12.4, 7.12.6, and 7.12.7.) In addition, a review of the original Design Control Summary showed that it was in agreement with the Project Criteria provisions noted above.

UNRESOLVED ITEM 4-2: TOP REINFORCING FOR SLABS

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

REVISED RESPONSE

This revised response has been prepared subsequent to a July 30, 1984, meeting between the NRC (IE) and Sargent & Lundy. As a result of the information exchanged at this meeting, this revised response supersedes the response to unresolved item 4-2 sent to the NRC on December 30, 1983, and the response to additional questions on unresolved item 4-2, which was sent to the NRC dated June 19, 1984 by letters from Commonwealth Edison Company to Mr. R. C. DeYoung.

As shown on Sargent and Lundy Drawings S-690 and S-790 (IDI References 4.64 and 4.77), negative moment steel, equal to that at the continuous support was provided at the junction to the wall of slab 4AS53. This negative steel is not required by design, according to Calculation 7.43 but is typically provided at slab/wall junctions to increase the factor of safety for slabs. (This typical slab detailing is shown on Sargent & Lundy Drawing S-472.)

The design conservatively assumed that the junction of the slab and wall is hinged. By assuming a hinge with no moment capacity, the maximum possible positive moment in the center of the slab is considered. Steel reinforcement is provided for this positive moment, which is all that is required for a safe design. Any negative moment steel added at a wall or other noncontinuous support provides an additional factor of safety.

In addition, in order to demonstrate that the typical detailing referred to above was in fact followed in the design, a survey of the reinforcement provided in two-way slabs has been performed. The survey sample was made up of all of the two-way slabs supported at the shear wall along Column Line L (14 slabs in all, representing approximately 20% of the total number of two-way slabs supported by exterior walls). The survey showed that in each case, the negative reinforcement provided at the slab-wall junction was at least equal to that provided at the continuous support. (In one case, the reinforcement provided at the slab-wall junction was slightly greater.)

RELATED CORRESPONDENCE

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

'84 OCT -1 P1:41

August 16, 1984

Mr. James G. Keppler
Regional Administrator
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Byron Generating Station Units 1 and 2
Integrated Design Inspection
Inspection Report No. 50-454/84-32

Reference (a): April 9, 1984 letter from J. Nelson Grace to Cordell Reed.

(b): May 2, 1984 letter from J. Nelson Grace to Cordell Reed.

Dear Mr. Keppler:

This letter supplies additional information regarding Commonwealth Edison's responses to the findings, unresolved items, observations and general concerns which were identified during the Byron integrated design inspection.

Attachment A to this letter contains responses to the NRC concerns identified in references (a) and (b) regarding the analyses of the consequences of pipe breaks. Several of these responses refer to work done recently to confirm the adequacy of the Byron 1 design with regard to jet impingement efforts. The report of that review is also enclosed. Similar documentation will be produced for Byron 2 and the Braidwood units.

Please address further questions regarding this matter to this office.

One signed original and fifteen copies of this letter and the attachment are provided for NRC review. Three copies of the report are also enclosed. Eight copies are being provided directly to Ron Parkhill and three copies are being sent to John Streeter.

Very truly yours,

D. L. Farrar
Director of Nuclear Licensing

lm

Enclosure : "Byron 1, Confirmation of Design Adequacy of Jet Impingement Effects," August 1984

AUGUST 1984

ATTACHMENT A

RESPONSES TO NRC LETTERS DATED APRIL 9, 1984 AND
MAY 2, 1984 REGARDING REINSPECTION OF HIGH AND
MODERATE ENERGY PIPE BREAKS AND CRACKS

INTRODUCTION

The NRC letters dated April 9, 1984 and May 2, 1984 transmitted concerns regarding the reinspection of high and moderate energy pipe breaks and cracks conducted during the week of March 26, 1984. The following responses clarify the Byron design approach and design features and should resolve these concerns.

Central to many of the IDI Team's concerns are comments relating to two Sargent & Lundy documents, Report BB-JI-01 "Jet Impingement Summary Documentation Report" and Calculation 3C8-1083-001 "Verification of High Energy Line Break Design Approach for Jet Impingement on Safe Shutdown Equipment." These concerns identify areas of potential jet impingement effects which the IDI team felt were not addressed or inadequately addressed by these two documents.

The nature of these concerns indicate an incomplete communication regarding the design approach used to address high and moderate energy line breaks and cracks and the purpose and scope of these two documents. The effects of postulated high energy line breaks and cracks were an important factor in the basic layout and design of the plant and in the separation criteria used for plant design. Report BB-JI-01 and Calculation 3C8-1083-001 document specific but limited aspects of this design.

Report BB-JI-01 was written to document and explain an informal review performed at Sargent & Lundy during the SER review to confirm that the separation concept had been adequately maintained to insure a high level of protection from effluents of pipe failure. This review specifically examined separation of electrical cables and electrical and mechanical equipment required for safe shutdown on the basis that these components were more likely to be subject to jet impingement damage and/or to be relocated than safe shutdown piping and structure.

Calculation 3C8-1083-001 is a more rigorous review of the potential jet impingement effects on safe shutdown mechanical and electrical equipment. This calculation was completed after the IDI Team report of September 30, 1983, as a demonstration of the effectiveness of the Byron design approach. Again, certain types of components were not addressed because the purpose was to demonstrate that a representative group of components would not be adversely affected by jet impingement.

To address these concerns in a more global sense, in addition to the responses to individual concerns, an additional report has been completed and is included with these responses. This report, "Confirmation of Design Adequacy for Jet Impingement Effects," addresses all types of safe shutdown components and, again demonstrates the adequacy of the Byron Unit 1 design for postulated jet impingement effects. Similar documentation will be produced for Byron Unit 2 and Braidwood Units 1 and 2.

AUGUST 1984

Response to NRC Letter
Dated April 9, 1984

CONCERN NO. 1

"An evaluation needs to be made of jet impingement effects on piping (including check valves), conduit, cables and cable trays, electrical penetrations, snubbers, and structures (including tanks and heat exchangers). Calculation 3C8-1083-001 states that these items were not addressed in the jet impingement analysis covered by this calculation. Sargent & Lundy stated that cables are addressed in the Fire Protection Report. However, the analysis of cables for fire protection is not adequate for purposes of evaluating jet impingement effects."

RESPONSE

Protection from jet impingement effects results from the design approach of:

1. Isolating high energy lines from safe shutdown systems;

2. Separating redundant safe shutdown systems; and

3. Providing diverse methods of shutdown.

The potential hazard associated with High Energy Line Breaks (HELB) and jet impingement can be evaluated by reviewing:

1. Location of high energy lines;

2. Location of safe shutdown systems; and

3. The redundant and diverse equipment used for safe shutdown.

This, in fact, was the method used by the Auxiliary Systems Branch of the NRC to review the plant design for protection against the effects of high energy line breaks. This review is documented in Section 3.6 of the Byron Safety Evaluation Report (NUREG-0876, Supplement No. 2, January 1983).

After the original IDI inspection, the IDI Team found that because individual jet properties were not calculated, the required jet impingement work could not readily be determined to be complete. In response to the IDI concerns, Sargent & Lundy prepared Calculation 3C8-1083-001 to demonstrate that the original design provides adequate protection against jet impingement effects. This calculation was a damage study (or functional failure analysis) which evaluates the loss of active safe shutdown electrical and mechanical equipment due to jet impingement. This study examined this subset of safe shutdown components because of the critical nature and potential vulnerability to jets of this equipment. Failure of equipment physically near the equipment in question, plus a limiting single active failure, was addressed in the study. The study demonstrated that the original design approach is effective.

The scope of Calculation 3C8-1083-001 was limited to safe shutdown equipment and, as indicated in the calculation, specifically did not address the items listed in Concern No. 1. The scope was intentionally limited because the examination of the critical components was considered adequate to establish that the approach taken in the design provides the required protection against the effects of jet impingement.

The items listed in Concern No. 1 will have a lesser potential for unacceptable damage as a result of jet impingement when compared to equipment and cables. This conclusion is reached as a result of the design of the plant system and the physical nature of jets and fluid discharge. To address this concern, an additional confirmatory report (including a revision of Calculation 3C8-1083-001) has been performed to confirm that the piping, tanks, heat exchangers, structures, cables, conduit, snubbers, and electrical penetrations are designed such that safe shutdown capability is not adversely affected by high energy line breaks and jet impingement. This report, "Confirmation of Design Adequacy for Jet Impingement Effects," has been included with these responses.

In the particular concern expressed about cables, the basic information cited in the Fire Protection Report pertaining to cable separation is applicable to jet impingement. The Fire Protection Report was used as a source of information which locates safe shutdown equipment and systems in the plant. The effects of single active failure and the potential for jet impingement damage to walls are unique aspects which can be evaluated by examining the specific system failures and by extending the jet area of influence.

In the Auxiliary Building, the majority of the fire barriers also serve as impingement barriers. The report, "Confirmation of Design Adequacy for Jet Impingement Effects," has been prepared utilizing, as boundaries, only those walls which can be demonstrated to withstand jet loads. The study demonstrates that separation plus the diversity of shutdown paths provide safe shutdown capability considering jet impingement and single active failure.

In the containment, a judgment is made that the 20-foot horizontal separation of the redundant electrical divisions provides protection against jet impingement. This judgment is made also when evaluating the separation of mechanical and electrical equipment. This separation in conjunction with the redundancy and diversity of the design provides protection against jet impingement effects. The separation of redundant electrical cables is documented in the Safe Shutdown Analysis of the Fire Protection Report. The only locations in the containment where a large number of cables of a division could be damaged by one jet are cable trays located high in the containment. The high energy lines are located low in the containment. With very few exceptions, the high energy lines (or postulated jets) in the containment are oriented vertically or horizontally and not skewed. Given that the jets must have a vertical component to reach the trays, the 20-foot horizontal separation is judged to be adequate. In addition, most high energy lines in the containment are high temperature lines. It is judged that these postulated two-phase jets are of limited influence because of jet dissipation as the fluid flashes. This judgment is verified by the jet impingement load calculation methodology (based on test results) in NUREG/CR-2913 (January 1983) currently under review by the Mechanical Engineering Branch of the NRC.

Most cold high energy lines (either inside or outside of containment) are limited in potential jet energy because the breaks are generally fed by closed lines or pumps (with limited flow) rather than high energy vessels.

As an additional review of jet impingement effects, Bechtel Power Corporation is reviewing the Byron design for high and moderate energy line breaks and cracks as it affects specific systems in the Independent Design Review process. This review will provide an independent confirmation of the adequacy of Byron jet impingement design approach.

CONCERN NO. 2

"An evaluation needs to be made of jet impingement effects on instrumentation lines. Sargent & Lundy initially stated to the team that instrumentation lines, both inside and outside of containment, are within Westinghouse scope of work for analyzing the effects of piping failures. When the team requested formal confirmation of Westinghouse work in this area, Sargent & Lundy stated that, based on its discussions with Westinghouse during the week of March 26, 1984, it was now unclear who had the responsibility for this work and whether it had been accomplished."

RESPONSE

Sargent & Lundy has the responsibility for design against the effects of jet impingement. This was accomplished in the original routing of the instrument lines. Sargent & Lundy meeting notes of a special project meeting on High Energy Rupture Studies (March 23, 1976) states that wherever 20 foot separation cannot be maintained between redundant essential instrumentation lines, the need for additional protec-tion will be investigated.

Westinghouse performed a review of the separation of instrument lines in 1983. The Sargent & Lundy responsible project engineer thought, based on discussion with Westinghouse, that this review had included an evaluation of potential jet impingement effects. Subsequently, the responsible engineer contacted Westinghouse and was informed that the Westinghouse review had not included jet impingement effects. This was reported by the IDI Team during the week of March 26, 1984, when the issue was first raised.

As-built drawings of safe shutdown instrument line routings have been reviewed against the original routing criteria to confirm that adequate separation is provided. These lines are included in the August, 1984 report, "Confirmation of Design Adequacy for Jet Impingement Effects," described in the response to Concern No. 1.

CONCERN NO. 3

"An evaluation needs to be made of jet impingement effects on block walls and other walls surrounding equipment cubicles to determine whether piping failures in one cubicle can affect equipment in other cubicles. The jet impingement analyses performed by Sargent & Lundy address line failures and equipment confined to areas which are defined by block walls or other walls. Analyses have not been made of the effects of jet impingement upon these walls. The inspection team was informed that the Structural Department has some preliminary data on loading of walls due to jet impingement forces, but that it is necessary to perform final load checks based on final postulated impingement forces. Sargent & Lundy stated that there are about ten cubicle areas involving block walls, and these would fail under jet impingement forces."

RESPONSE

Early in the design, certain concrete and block walls were designed considering jet impingement loads. A partial list of documentation discussing these considerations is listed at the conclusion of this response.

Block walls in these 10 cubicle areas had not been specifically determined to fail due to jet impingement forces, but rather that, if exposed to high jet impingement forces, the integrity of the block walls had not been established.

Using final HELB locations, wall loads have been postulated for use in the final load check of the structure. The results of this load check confirm that HELB will potentially cause failure only in walls where failure does not affect safe shut-down capability.

The previously mentioned report, "Confirmation of Design Adequacy for Jet Impingement Effects," has included the potential effects of block wall failure.

Documentation of Concrete and Block Wall Design

- Project Communication, "Schedule for Pipe Whip Restraint Information for Auxiliary Building, Auxiliary Feedwater Tunnel and Main Steam Tunnel," March 15, 1976. This memo states that impingement loads will be provided to structural engineers for design and lists location of high energy line breaks.

- Project Meeting Notes, "High Energy Line Rupture Studies - March 23, 1976 Interdepartmental Meeting." At this meeting:

(1) the schedule for transmitting impingement loads was reviewed, and;

(2) the structural engineers were informed that for much of the auxiliary building, the loads were very low and would be transmitted to verify that they could be neglected in the structural design.

- Project Communication, "Pipe Rupture Analysis at El. 346'-0" in Auxiliary Building," July 7, 1976. This memo confirmed a discussion held in a June 14, 1976 Inter-departmental Meeting where it was determined that block walls were acceptable around the blowdown condenser because potential failure due to high energy line break would not affect safety-related equipment.

- Project Communication, "Preliminary Pipe Rupture Analysis for A.S. lines inside Auxiliary Building El. 426.00-401.0 and Pipe Tunnel El. 394.0," November 17, 1976. This memo transmitted jet impingement loads due to auxiliary steam line ruptures.

- Project Communication, "Pipe Rupture Analysis Progress Inside the Main Steam Tunnel and Auxiliary Building," February 22, 1977. This memo documented that potential jet impingement loads in the main steam tunnel and auxiliary building were being addressed.

CONCERN NO. 5

"Sargent & Lundy needs to evaluate Westinghouse design criteria SS 1.19, "Protection Criteria Against Dynamic Effects Resulting From Pipe Rupture" for applicability to Byron. Sargent & Lundy informed the team that this is a baseline design document and their normal practice would be to review it and identify any areas where compliance would be considered impractical. SS 1.19, Rev. 0 was transmitted by Westinghouse to Sargent & Lundy in 1978, but was never sent to the Project Management Division, which is responsible for reviews for protection against the effects of pipe failure. The team is concerned that this oversight may make it difficult for Sargent & Lundy to comply with all design provisions of SS 1.19 at the current stage of construction."

RESPONSE

This concern should be clarified in light of further investigation at Sargent & Lundy since the IDI visit the week of March 23, 1984. Westinghouse Design Criteria SS 1.19 (proprietary) contains design information applicable to Byron and was utilized in the design of the Byron Station.

When the IDI Team asked questions about information in a preliminary 1970 version of this document (SS 1.19) which the IDI Team had brought to the inspection, they were told that the information appeared familiar and consistent with the plant design basis but the responsible Mechanical Project Engineer (K. J. Green) could not recall having reviewed the specific document. Revision 0 of SS 1.19 was located in the Structural Project Engineer's file during the inspection.

The following has been determined since the IDI inspection:

- A copy of SS 1.19 (preliminary) was in the office of J. Lazowski, Mechanical Design and Drafting Project Leader on the Byron Project and a piping designer in the initial layout of the safety-related piping systems.

- A copy of SS 1.19 (preliminary) was in the Byron Project Files. This copy bears the name of R. B. Johnson, the responsible project engineer in the Project Management Division for high energy line break work in 1974 through 1980.

- A copy of SS 1.19 (preliminary) was in the files of K. J. Green, the current responsible engineer in the Project Management Division for High Energy Line Break. Retrieval of this document from the files was complicated at the time of the inspection because it is part of the

Westinghouse Reactor Fluid Systems Standard Design Package Four Loop Plant Nuclear Steam Supply Systems. The document is identified in this package as STD-DES-4L-RFS-4L7 rather than SS 1.19. The designation SS 1.19 appears only as a handwritten note on page 2 of this document.

- Revision 0 of SS 1.19 was transmitted to the Sargent & Lundy Project Team by Westinghouse Letter CAW-2725 (4-6-79). While it has not been clearly determined where this particular copy was subsequently filed, the copy was received by the Project Manager.

A documented review of Revision 0 of SS 1.19 has been completed and no inconsistency between this document and the Byron design has been found.

CONCERN NO. 6

"Criteria should be established for reviewing design changes for impact upon completed analyses of effects of postulated piping failures. Sargent & Lundy stated that its procedures require the responsible engineer for the design change to evaluate all aspects of the change, including impact upon the piping failure analyses. The team considers this is inadequate because the piping failure analyses are highly specialized and were performed by groups other than those responsible for the design, i.e., Project Management Division and Nuclear Safeguards and Licensing Division. Criteria need to be established defining circumstances under which design changes will have an impact upon completed analyses of piping failures, and in these cases the design changes should be reviewed by the groups responsible for the piping failure analyses."

RESPONSE

Design changes which would require a review for effects of piping failures are those which involve extensive relocation of high energy lines or safe shutdown equipment. The design approach used and the verification studies ensure that minor changes do not compromise the design. This approach conservatively assumes loss of equipment in a general area after a postulated piping failure rather than an evaluation of the exact geometrical relationship of the high energy break location and safe shutdown equipment.

Design changes are reviewed by the responsible engineer and referred to other members of the project team as required. Major changes are reviewed by all affected design disciplines. The design change procedures require the responsible engineer to identify the scope of review needed. The project experience in the confirmation work performed to date demonstrates that this approach has been successful since design problems have not been found, and plant changes are not required.

CONCERN NO. 7

"Sargent & Lundy needs to confirm Westinghouse agreement with the list of equipment required for safe shutdown. Calculation 3C8-1083-001 includes the "Safe Shutdown Equipment List (SSEL)" which is based upon active valve lists in the FSAR, the Byron/Braidwood Mechan[illegible]al Equipment Qualification List and the Byron/Brai[illegible]od Fire Protection Report. There is no record of this [illegible] having been concurred with by Westinghouse. Calculation 3C8-1083-001 indicates required instrumentation needed to support safe shutdown, e.g., 2 of 4 reactor coolant cold leg temperature sensors. In four cases, the postulated piping failures result in fewer than the required instruments for safe shutdown, and in two of these four cases the report states that core exit thermocouples would provide redundancy to the failed instruments (hot or cold leg resistance temperature detectors). These conclusions are inconsistent with the "required" instrumentation indicated in Calculation 3C8-1083-001, and need to be confirmed by Westinghouse."

RESPONSE

Westinghouse has reviewed the Safe Shutdown Equipment List (SSEL) in Calculation 3C8-1083-001 as included as Appendix A of the August 1984 Confirmatory Report. Westinghouse concurs with the SSEL and the Confirmatory Report.

The Safe Shutdown Equipment List (SSEL) includes equipment in both the NSSS (Westinghouse) and Balance-of-Plant (Sargent & Lundy) scope. The term "required" when used in conjunction with this list really means "required under at least one High Energy Line Break scenario." Therefore, allowing failure of a "required" system or component is not inconsistent. This list was assembled using equipment lists and equipment classifications developed in the design of the safe shutdown systems. Westinghouse has provided, as part of the NSSS design information, descriptions of the Westinghouse designed systems, classification of NSSS equipment, and emergency operating procedures. Jet impingement analysis is in the Sargent & Lundy scope of responsibility and, therefore, Sargent & Lundy has the responsibility for defining the SSEL in Calculation 3C8-1083-001.

As was discussed with the IDI Team the week of March 26, 1984, Westinghouse provided to Sargent & Lundy a list of all safety-related electrical equipment in conjunction with the environmental qualification of Class 1E electrical equipment in 1981. P&ID's showing the safe shutdown portions of systems were also developed by Sargent & Lundy and reviewed with Westinghosue in 1982. For the mechanical equipment qualification program, Sargent & Lundy developed a list of safe shutdown mechanical components.

This list was transmitted to Westinghouse and reviewed with Westinghouse in 1981. The information developed during these efforts was the basis of the list of safe shutdown equipment included in Calculation 3C8-1083-001. The final calculation was not reviewed by Westinghouse. However, the input information was.

The formulation of a calculation such as this draws on the information provided by sources such as Westinghouse and Commonwealth Edison. The Sargent & Lundy responsible engineers utilize their experience and expertise to interpret available information and to provide clarification to additional information as they determine necessary. The concern also gives an example in which temperature sensor requirements were modified based on evaluations performed by Sargent & Lundy. This is an example where further investigation was made by Sargent & Lundy responsible engineers in consultation with Commonwealth Edison.

In the course of the design of Byron Station, Sargent & Lundy has conferred with Westinghouse on questions which involve clarification in Westinghouse design information. Because of the questions being raised, Westinghouse has reviewed the SSEL in Calculation 3C8-1083-001 to confirm the adequacy of the NSSS portion of this list.

CONCERN NO. 8

"Additional information needs to be provided with respect to specific piping failure analyses as follows.

a. Report PD-J1-01 states for Zone 11.3-1 that failure of steam generator blowdown lines (3") do not pose a jet impingement hazard to a motor control center. Analysis needs to be made of the effects of jet impingement from these breaks upon essential service water lines in the area (6", 8", and 20").

b. Calculation No. SC8-1083-001 states that Motor Control Center 1AP21E is postulated to fail due to jet impingement; failure would render all its dependent equipment inoperable. The calculation assumes a single active failure to one specific equipment item which is powered by the redundant Motor Control Center (MCC 1AP23E), but not failure of that entire motor control center. A failure analysis needs to be performed to substantiate this assumption.

c. Report BB-J1-01 indicates for Zone 11.6-0 that water spray could result in failure of 2 of 3 cooling fans in an electrical equipment cubicle in addition to single active failure resulting in loss of the redundant power division. Analysis needs to be made of the heatup of electrical equipment in this case and its effect upon ability to achieve safe shutdown.

d. Report BB-J1-01 indicates for Zone 11.3-0 there are two each unit 1 and unit 2 component cooling water pumps, a pump common to both units, and valves used to align the common pump to either unit. Based on review of the drawings and the fact that fire protection piping resulting from a recent design change was not on the drawings, the team is concerned that the right combination of water spray damage and assumed single active failure could result in loss of component cooling water to one unit. A detailed pipe break/crack review should be performed, including the new fire protection piping, to determine whether the design is adequate.

e. Report BB-J1-01 states that for Zone 11.2A-1, fire protection and containment spray lines are about 20' from the residual heat removal pump and are therefore, unlikely to damage the pump. The team determined, based on review of the drawings, that this separation is only about 15'. An analysis needs to be made as to whether this separation is adequate and, if not, whether necessary repairs can be made and cold shutdown can be achieved within the technical specification allotted time for shutting down when the containment spray system is unavailable.

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

RESPONSE

(a) Report BB-JI-01 describes the conclusions reached in informal reviews of jet impingement and water spray effects. These conclusions were reached prior to finalization of break locations in high energy systems. As a result, a conservative approach was taken to potential line break locations. Final break locations are now available. Break information for the steam generator blowdown (SD) lines in question are shown in the attachment to Westinghouse letter CAW-7145 (3-22-84) which has been provided to the IDI Team. This letter indicates that no blowdown system breaks are postulat 1 in the same room as the essential service water nes in question.

If a break were postulated in the SD piping at the fitting closest to the Essential Service Water Lines, the resulting loads, calculated using NUREG/CR-2913, would be negligible.

(b) The only active component in a motor control center (MCC) is the contactor portion of each combination starter. The main power feed cable for Motor Control Center 1AP23E is connected directly to the bus. There is no active motor control center component whose failure can directly affect the power supply to the motor control center. The operation or failure of any active motor control center component will affect only the individual circuit connected to that component. Complete failure of the MCC would result only from loss of power to the MCC circuits.

Further review of the high energy line break location reveals that MCC 1AP21E will not be affected by jet impingement. Calculation 3C8-1083-001 is being augmented to reflect this change and to include an assumption of loss of an entire electrical division as the postulated single failure. Although this exceeds the apparent intent of SRP Section 3.6.1, this simplifying assumption can be made because of the conservatism of the Byron design.

(c) Failure of two of three fans in the miscellaneous electrical equipment and battery room was judged not to affect safe shutdown because of the conservatism used in the Electrical Enviornmental Qualification Program and the results of studies on fan loss in similar areas. A calculation has subsequently been completed and it has been determined that, with the two fans out of service, a maximum transient temperature of about 108° F will be experienced in the room. The steady state qualification temperature in this zone (Environmental Zone A3) is specified as 108° F (FSAR Table 3.11-2, Byron Environmental Qualification Report Table 3.1-1). Therefore, as noted in Report BB-JI-01, no adverse effect of the jet impingement will impair safe shutdown.

(d) This concern, as written, does not fully explain the potential water spray hazard review in the component cooling pump area. In the original design layout of this area, the only piping in the area was component cooling piping and an essential service water line to the component cooling heat exchangers. This piping is designed to meet ASME, Section III requirements. A review of the stress levels shows that only one moderate energy crack location need be postulated in this area. This location is in a 12-inch component cooling supply header. Based on the single failure exclusion for dual purpose moderate energy systems (Ref: SRP Section 3.6.1), this crack would be of concern if it disabled three component cooling pump motors. Spray from the postulated crack would not disable these three motors because of the separation and orientation of the three motors and the location of the postulated break with respect to the motors.

The additional fire protection piping is the result of comments by the NRC Fire Protection Reviewers and was being finalized during the IDI reinspection. The potential for water spray damage had already been reviewed in detail at the time of the inspection and a decision had been made to install spray shields on the component cooling pump motors and to install partial walls between the pumps. This information was communicated to the IDI Team during the inspection.

A detailed review of potential pipe breaks/cracks has been performed in this area and the design is confirmed to be adequate.

(e) The fire protection and containment spray lines in the residual heat removal pump cubicle are designed to meet ASME, Section III requirements. The stress level for these lines has been reviewed and it has been determined that no cracks must be postulated in these lines in accordance with the stress criteria given for moderate energy line failure exclusion in Standard Review Plan Subsection 3.6.2 (Branch Technical Position MEB 3-1). Therefore, no crack is required to be postulated in these lines and a spray hazard does not exist for the residual heat removal pumps from these lines.

(f) The auxiliary feedwater system consists of a motor-driven auxiliary feedwater pump, a diesel-driven auxiliary feedwater pump and associated piping and valves. There are no high energy line breaks which will adversely affect the Auxiliary Feedwater System, including power and control functions. Spray from moderate energy lines could, at most, damage one train. The moderate energy lines in question are lines such as service water and fire protection. None of these potential cracks would, in itself, cause loss of offsite power or reactor trip. In accordance with Standard Review Plan Subsection 3.6.1 (Branch Technical Position APCSB 3-1), loss of feedwater is not assumed and the auxiliary feedwater system is required only if the single active failure causes loss of offsite power. In that event, one train of auxiliary feedwater is available.

Therefore, there are no postulated high or moderate energy line breaks which would result in both loss of an auxiliary feedwater train and demand for auxiliary feedwater (loss of offsite power or main feedwater). The NRC has previously required that feed and bleed cooling be incorporated into the Byron Emergency Operating Procedures. The NRC has required that the pressurizer PORV's at Byron be upgraded to afford greater assurance of success for feed and bleed cooling operations. The NRC has accepted feed and bleed cooling for design basis events as specifically noted in the Byron SER, Section 5.4.3; Supplement 2 to the Byron SER, Section 5.4.3; and Supplement 4 to the Byron SER (Draft), Section 5.4.6. Feed and bleed cooling is clearly acceptable to the NRC staff for a variety of design basis events and has been technially accepted as viable by the NRC for a wide variety of postulated events which go beyond the plant design bases. Feed and bleed cooling constitutes a technically acceptable alternative cooling mode for high and moderate energy line break events and that feed and bleed events should also be an acceptable licensing alternative for such events.

As demonstrated by the August 1984 Confirmatory Report, jet impingement will not result in a need for feed and bleed cooling. The references (12-30-83 letter) to feed and bleed in the event of total loss of auxiliary feedwater are options required only in the event of failure beyond the design basis and are included to shown that the diversity and redundancy of the plant design exceeds regulatory requirements.

AUGUST 1984

Response to NRC Letter
Dated May 2, 1984

SITE INSPECTION

CONCERN NO. 1

"Report BB-JI-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 to the potential for pipe cracks and, if any, the path of water spray."

RESPONSE

The two motor control centers (MCC) 131 x 5 and 132 x 5, are located on plant Elevation 426 feet 0 inch, in a corridor. These two MCC's were placed on opposite walls to afford the maximum possible separation within the area. The fire protection line is the only liquid line in the area. Fire protection headers, which are located in many areas of the plant where fluid systems would not ordinarily be routed, are designed to the requirements of ASME Section III. The fire protection line in this area was not originally reviewed for postulated crack locations because it was believed that the routing of the line and the cables, cable trays, drain lines, and air lines in the area made it very unlikely that water spray from a single crack could damage both MCC's.

Existing piping stress analyses have now been reviewed to assess the potential for pipe cracks. The fire protection line in question is designed to the requirements of ASME Section III, with stress levels adequately low such that no postulated cracks are required by the guidelines in Standard Review Plan Section 3.6.2. There are no other liquid piping lines in this area.

SITE INSPECTION

CONCERN NO. 2

"Report BB-JI-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 & Lundy analysis, whether they are postulated to crack, and, if so, the impact on ability to reach safe shutdown."

RESPONSE

Report BB-JI-01 contained an error concerning Zone 11.4-0 apparently caused by a misinterpretation of the design drawings. In reality there are fire protection, essential service water, and nonessential service water lines in this area. All of these lines have been reviewed to determine required postulated crack locations based on stress level. One postulated crack location on nonessential service water line 1WS57A-18 was identified. This location is over 30 feet west of the MCC. There are two structural columns between the crack location and the MCC. As a result, there is no impact on safe shutdown capability.

SITE INSPECTION

CONCERN NO. 3

> "Report BB-JI-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 & Lundy's analysis. Therefore, there is uncertainty 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)."

RESPONSE

Line 1CV42E-2" is the Charging Pump Miniflow Line. This line had been recently rerouted as part of a design change in response to I. E. Bulletin 80-18. This change had not yet been made to the composite drawings at the time of the IDI inspection. This line is downstream of the charging pump miniflow orifice. Therefore, a break in this line would not produce significant jet impingement forces. The current routing of the line is farther from MCC 131 x 1 than the original routing. An additional review demonstrated that jets do not impact MCC 131 x 1.

SITE INSPECTION

CONCERN NO. 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 & 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."

RESPONSE

It appears that this concern results from a misinterpretation of SS 1.19. Small line LOCA's are limited to the affected leg to ensure natural circulation in the unfaulted loops. As was explained to the IDI Team at the time of the inspection, the 3/4-inch sample line in question is isolated by a 3/8-inch orifice at the connection to the main loop. The limited flow area (approximately 0.1 square inches) clearly eliminates concern about this failure. Section 3-3-2 of SS1.19 discusses the limitation on break propagation in light of the 3/8-inch orifice.

SITE INSPECTION

CONCERN NO. 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."

RESPONSE

For the purpose of Calculation 3C8-1083-001, Sargent & Lundy had not distinguished between longitudinal and circumferential breaks. This calculation evaluated safe shutdown equipment as a representative sample of safe shutdown components and assumed that all equipment in a conservative area of influence was damaged by jets from breaks, including breaks B10A and B10B. Sargent & Lundy and Westinghouse had established in the design of the piping that no longitudinal breaks need be postulated on the Byron high energy piping with the exception of one elbow on each main reactor coolant loop which is mitigated by a jet impingement shield. This was determined by comparing piping stresses with the guidelines of Branch Technical Position MEB 3-1.

Therefore, the postulated longitudinal breaks identified in this concern are not potential breaks. It should be noted, however, that most of the targets listed are normally isolated by manual valves or 3/8-inch orifices. The design of the plant provides adequate protection against the postulated circumferential break.

SITE INSPECTION

CONCERN NO. 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."

RESPONSE

The intermediate break locations are in the pressurizer enclosure and at the auxiliary spray line connection from the charging system. The requirements and guidelines of Westinghouse Design Criteria SS 1.19 are met. The breaks have been plotted on composite drawings and will be included in the next amendment to the FSAR.

SITE INSPECTION

CONCERN NO. 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."

RESPONSE

Safe shutdown instrument cables inside containment have been routed to maintain adequate separation between redundant cables. This was demonstrated in the Fire Protection Report. Similar separation was maintained when instrument lines were routed.

A confirmatory review of the separation of and potential jet impingement effects on safe shutdown instrument lines and cables has been completed as part of a full verification study of jet impingement effects. The results establish that adequate separation exists.

As for the two instruments mentioned in the IDI concern, the neutron detectors have been deleted from the high energy line break safe shutdown list as a result of meetings between Sargent & Lundy, Commonwealth Edison, and Westinghouse; and the pressurizer pressure transmitters have been demonstrated to be adequately separated to perform required safe shutdown function.

INSPECTION AT SARGENT & LUNDY OFFICES

CONCERN NO. 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)."

RESPONSE

The "single train" zones are Auxiliary Building subcompartments. In these zones there are no piping, cables, or instrument lines associated with the redundant train. However, piping damage is not a concern because the postulated Auxiliary Building jets in these areas do not contain sufficient energy to damage piping. This is discussed in more detail in the confirmatory report, "Confirmation of Design Adequacy for Jet Impingement Effects."

INSPECTION AT SARGENT & LUNDY OFFICES

CONCERN NO. 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."

RESPONSE

The potential for jet impingement damage to piping has been addressed in the confirmatory report, "Confirmation of Design Adequacy for Jet Impingement Effects."

INSPECTION AT SARGENT & LUNDY OFFICES

CONCERN NO. 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."

RESPONSE

The Byron plant is designed to safely remain in a hot standby condition for an extended period of time and the licensing basis is hot shutdown. For postulated accidents within the design basis, the Byron design includes an established capability to reach cold shutdown. In accordance with the licensing basis, this capability may include use of alternate procedures or non-qualified equipment.

The procedure of using steam generators to reach a cold shutdown condition is within the accepted Byron shutdown procedures. It requires use only of equipment normally used for shutdown. However, this procedure will not be required after high energy line breaks for the following reasons:

- The only active component inside containment (RHR suction valves) is not in proximity to any non-LOCA breaks. Therefore, manual operation will be possible prior to initiating RHR after an in-containment, non-LOCA HELB.

- No active components inside containment are required after a LOCA because suction is taken from the RWST and later from the containment sump.

- The only active component outside containment is the RHR pump. There are no breaks or cracks in the same cubicle as the RHR pump.

Therefore, no modifications are necessary to protect RHR equipment.

INSPECTION AT SARGENT & LUNDY OFFICES

CONCERN NO. 4

"The Sargent & 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 nonseismic Category I piping with Category I supports would remain intact in an SSE. Alternatively, you should re-evaluate the consequences of breaks and cracks in nonseismic Category I piping, using the assumption that an SSE could result in piping failure concurrent with loss of offsite power."

RESPONSE

Piping in the safety-related areas of the Byron plant falls into two categories:

1. Piping designed to the requirements of ASME Section III and supported to withstand seismic loads; and

2. Piping designed to the requirements of B31.1 and supported to withstand seismic loads.

Piping in either category which was designed using a specific stress analysis and demonstrated to be below allowable stresses at all points, was not considered to crack or break as a result of seismic events since this would be a nonmechanistic load combination. Cracks were postulated as initiating events in locations where the stress exceeded 40% of allowable in accordance with the Standard Review Plan Section 3.6.2. A limited amount of non-safety-related piping in safety-related areas has been designed by simplified methods and no specific stress analysis is available. Cracks were postulated at all fittings for this piping.