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December 31, 1981

Mr. Harold R. Denton, Director
 Office of Nuclear Reactor Regulation
 U.S. Nuclear Regulatory Commission
 Washington, DC 20555

Subject: Byron Station Units 1 and 2
 Braidwood Station Units 1 and 2
 Advance FSAR Information
NRC Docket Nos. 50-454/455/456/457

Dear Mr. Denton:

This is to provide advance copies of information which will be included in the Byron/Braidwood FSAR in the next amendment. Attachment A to this letter lists the information enclosed.

One (1) signed original and fifty-nine (59) copies of this letter are provided. Fifteen (15) copies of the enclosures are included for your review and approval.

Please address further questions to this office.

Very truly yours,

T. R. Tramm

T. R. Tramm
 Nuclear Licensing Administrator
 Pressurized Water Reactors

Attachment

3129N

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 1/15*

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 PDR ADOCK 05000454
 A PDR

Attachment A

List of Information Enclosed

I. FSAR Question Responses

10.44
40.162 thru 40.183
331.33
130.43

II. FSAR Text Changes

1. Text changes in Sections 3.6, 3.7 and 3.9 to document resolution of MEB items listed below:

N1	N12
N2	N16
N3	N18
N5	N21
N6	N22
N7	N23
N9	N24

2. Text change in Section 7.3 to resolve ICSB agenda item 97.
3. Text changes in Section 9.2 (and the response to 10.44) to document the resolution of ASB open item no. 3
4. A description of the Condensate Cleanup System for Section 10.4.6 of the FSAR (Chemical Engineering Branch).
5. New FSAR Figures for Section 3.8 corresponding to the response to Question 130.43.

III. Miscellaneous (Not for FSAR)

1. Unresolved Safety Issue A-12
2. Additional information to close open items for the Structural Design Audit:

4A
19
23
27
29
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QUESTION 010.44

"Your response to Q010.9 is not complete. You have indicated that tests of the reactor coolant pumps performed by Westinghouse indicate that the pumps can function satisfactorily for 10 minutes without component cooling water supply. Low component cooling water flow alarms and high component cooling water temperature alarms from the reactor coolant pump oil coolers are provided in the control room to indicate a loss of component cooling water supply. Operator action can be taken within the 10 minutes available to secure the reactor coolant pumps. It is our position that the alarm indication of loss of component cooling water flow to the reactor coolant pumps is safety grade and meet the requirements for Class 1E instrumentation. Verify your response accordingly."

RESPONSE

Redundant safety-related indication of component cooling water flow to the reactor coolant pump thermal barrier is provided and alarmed on the main control board. In addition, the following five temperatures are input to the plant computer:

1. RCP motor stator winding temperature,
2. RCP motor upper radial bearing temperature,
3. RCP motor upper thrust bearing temperature,
4. RCP motor lower radial bearing temperature, and,
5. RCP motor lower thrust bearing temperature.

These temperatures would be alarmed and printed by the plant computer should any of them go out of range. These would provide sufficient indication that the pump was too warm and the operator would be able to trip the pump.

QUESTION 040.162

"Your response to Q040.12 on the motor operated valves requiring power lockout is not complete. Subsection 8.1.10 of the FSAR does not describe how the power to the MOVs is disconnected. Provide this information. The acceptable method to ensure that electrically operated valves will not by themselves, spuriously move for any operating or postulated plant condition is to have the motive power locked out (circuit breaker racked out). However the electrical schematics provided show that the power to the MOVs is disconnected on receipt of a safety injection signal. This is not in accordance with BTP EICSB 18 PSB and, therefore, not acceptable. Clarify this discrepancy."

RESPONSE

The design utilizes a power lockout "circuit breaker-starter" in series with the "circuit breaker-starter" for each of the MOV valves. This power lockout starter can be controlled from the main control room.

Example: Motor-operated valve LSI8808A

1. The power lockout feature is the starter at MCC 131X2, compartment B2 (which is the upstream starter for MOV LSI8808A), see key diagram 6/20E-1-4008E.
2. This starter contact can be opened or closed by the selector switch at the main control room panel 1PM06J (see schematic drawing 6/20E-1-4030 SI33). This feature meets the branch technical position EICSB 18, paragraph B.3.
3. The starter for MOV LSI8808A is at MCC 131X2A, compartment A3 which is downstream of the power lockout starter as mentioned in Item 1 (see drawing 6/20E-1-4008E).
4. When the power lockout starter contact opens (as mentioned in Item 2) there is no electrical power supply furnished to valve motor LSI8808A (see drawing 6/20E-1-4008E) or there is no power supply furnished to the control transformer which supplies the power supply to the control circuit.
5. During power lockout, if the safety injection signal (or any signal) is present, it will not change the valve position since there is no power to the motor or control circuit. Example for valve LSI8802A, see drawing 6/20E-1-4030 S111. When the safety injection signal is

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present, contact 7-8 of K603 will close but the starter "open" coil (O) will not energize since there is no power to the control transformer (480-120V).

Also if the 3-phase starter contact (O) gets stuck in the close position, the valve will not change position since there is no electric power to the valve motor.

Because of the design which is explained in Items 1 through 5, there is no need to rack the circuit breaker out or separate the control circuit.

However, for each passive valve, the individual breaker (located at its respective MCC compartment) will be manually tripped (opened) and tagged "Out of Service." Verification will be made that the control switch (located at the main control board) of each passive valve is in the desired position and tagged "Out of Service." This additional precaution will assure that the passive valves always remain in the correct position.

QUESTION 040.163

"The response to Q040.8 is incomplete. Describe the procedures to be followed, when temporary jumpers are installed, that will ensure removal of jumpers after the testing procedure is completed."

RESPONSE

Temporary jumpers and lifted leads are covered by Byron Operating Procedure; BAP 300-5. Jumper installation and removals are logged on the jumper log; BAP 300-T1. Lifted leads are logged on the lifted lead log; BAP 300-T2. Jumpers and lifted leads are documented in the appropriate pre-operational test during the testing program.

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QUESTION 040.164

"You state in Section 8.3.1.1 that automatic transfer of each non-Class 1E 6900 V or 4160 V switchgear from the UAT to SAT is provided upon a loss of power from the UAT or vice versa. Provide a detail description of the transfer schemes including test interval."

RESPONSE

The above referenced statement in Subsection 8.3.1.1 was eliminated in Amendment 21 of the FSAR dated July 1979, to avoid confusion.

Each non-Class 1E 6900-V and 4160-V switchgear can be supplied from either the UAT or the SAT. If a switchgear is supplied from the UAT and the control switch for the SAT feeder breaker is in the trip position and the UAT feeder breaker trips for some reason other than a fault on the bus, then the SAT feeder breaker will automatically close. The converse is true if the switchgear is supplied from the SAT. Schematic diagrams which illustrate the transfer scheme in more detail were provided.

The operating time for the breakers is less than 10 cycles. Operating times are as follows:

Main contacts open:

"a" contacts	2.7 ± 0.4 cycles	4.16 kV
	2.5 ± 0.4 cycles	6.9 kV
"b" contacts	3.0 ± 0.4 cycles	

Main contacts close:	5.1 ± 0.8 cycles	4.16 kV
	6.9 ± 0.6 cycles	6.9 kV

The second source of offsite power to the ESF buses is provided by the corresponding SAT of the other unit. The tie breakers are manually operated.

QUESTION 040.165

"Your response to 040.13 is not complete. You do not describe how you will test the penetration primary and backup protective devices. Also, provide the test interval."

RESPONSE

The last paragraph of our response to Question 040.13 stated the following regarding testing of the penetration primary and backup protective devices.

"There are no provisions for periodic testing of penetrations or fuses under simulated fault conditions because such testing would be detrimental to the penetrations and fuses. The overcurrent relays and circuit breakers that provide the primary and backup protection will be periodically tested under simulated fault conditions to demonstrate that the overall coordination scheme remains within the specified limits."

The test interval will be at least once every 18 months during refueling outages.

The following will supplement our response to Question 040.13 which identified each type of typical circuit that penetrates the reactor containment and described the type of protective devices used to provide primary and backup protection:

a) Circuits Energized from 125 Vdc Distribution Panels

The 125 Vdc emergency lighting cabinet is the only such load. The primary protection consists of a 125-Vdc molded case air circuit breaker that feeds the penetration directly. The backup protection consists of another molded case air circuit breaker similar (and connected in series) to that provided for primary protection.

b) Circuits (Solenoid Operated Valves) Energized From 125 Vdc Distribution Panels

The primary protection consists of a nonrenewable cartridge-type fuse that feeds the penetration directly. The backup protection consists of another fuse similar (and connected in series) to that provided for primary protection.

c) The primary and backup protection for the rod control system lift and gripper coil circuits consists of fuses connected in series.

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- d) The primary and backup protection for the rod position indication data cabinets consists of two 120 Vac molded case air circuit breakers connected in series.

QUESTION 040.166

"We have not received the qualification test reports for the containment electrical penetrations assemblies requested in Q040.14. Provide this information."

RESPONSE

The qualification test reports for the containment electrical penetrations do not contain the information necessary to evaluate the penetration failure described in Question 040.14. We have addressed this failure in our revised response to Question 040.14.

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QUESTION 040.167

"The response to question 040.62 omitted the essential parts of the question.

1. Provide a tabulation of each Class 1E system required to bring the plant to a safe cold shutdown.
2. Provide the cable routing locations for the power, control, and instrumentation cables for each system. (This is best accomplished with a drawing or a sketch)."

RESPONSE

The complete response to Questions 040.62 and 040.167 is covered by the Applicant's "Fire Protection Safe Shutdown Analysis."

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QUESTION 040.168

"Provide a complete description of the load sequencer that is used to connect the ESF loads to the emergency buses when power is being supplied from the diesel generator after a loss of offsite power. Also provide a description of how the load sequencer adds loads to the emergency bus after a LOCA and offsite power is available. The description should include details of how loads are shed and buses are isolated. Discuss the conditions required to connect the diesel generator to the emergency bus and how the ESF loads are connected to the emergency bus by the load sequencer."

RESPONSE

The automatic loading of the emergency bus is accomplished by the sequencing panels. There is one panel per diesel generator. The sequencing panels are electrically and physically separated per Electrical Division.

The sequence circuit (Ref. Schematic 6/20E-1-4030EF02 typical) uses relays and timers to accomplish the function of sequencing. Upon loss of offsite power, the emergency bus undervoltage relay (via auxiliary relays): a) trips all loads on the bus except the 4160-480V transformers, b) trips the feed breakers to the emergency bus, and c) interrupts power to the sequencing control circuit to ensure that the circuit resets, if in testing mode. Sequencing will begin provided: 1) the diesel generator breaker has closed and power is restored to the sequencing circuit, and b) inter-unit (cross-tie) feed breaker and SAT breaker are open. Sequencing is accomplished via timers whose contacts energize relays which in turn interlock the start circuits of the required loads. The time intervals for sequencing loads are as indicated in Table 8.3-1 of the FSAR.

Because the sequencing circuit will not have power for 10 seconds while the diesel is coming up to speed, there is a 10 second difference between the time settings in the circuit, and those in Table 8.3-1.

During a safety injection signal with sustained offsite power, the required loads are started immediately. With sustained offsite power, the timers are bypassed and the relays are energized directly by the safety injection signal (K608) via auxiliary relay SARB. The bypass circuit is blocked on loss of offsite power by a) the Unit 1 SAT breaker (52/a contact) and b) the inter-unit (cross-tie) breakers 52/a contacts in series with the Unit 2 SAT feed breaker (52/a contact).

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The conditions required to connect the diesel generator to the emergency bus are as follows (Ref. Typical Schematic 6/20E-1-4030DG02):

Automatic

- a) SAT breaker open and not locked out (486-1422)
- b) Inter-unit feed breaker open (52/b 1424)
- c) Tie breaker open (52/b 1421)
- d) Diesel generator breaker not locked out (486-1423)
- e) Diesel generator reaches rated voltage and speed (DGLBX).

Manual

- a) Control switch at Main Control Board closed (CS/C)
- b) Synchronizing switch closed (SS/ON)
- c) Synchro-check relay contact (HACR-1) closed.
- d) Diesel generator reaches rated speed and voltage (DGLBX)
- e) Feed from SAT breaker not locked out (486-1422)
- f) Diesel generator breaker not locked out (486-1423).

B/B-FSAR

QUESTION 040.169

"Concerning the emergency load sequencers which are associated with the offsite and onsite power sources we require that you either provide a separate sequencer for offsite and onsite power (per Electrical Division) or a detailed analysis to demonstrate that there are no credible sneak circuits or common failure modes in the sequencer design that could render both onsite and offsite power sources unavailable. In addition provide information concerning the reliability of your sequencer and reference design detailed drawings."

RESPONSE

The Byron/Braidwood design utilizes a load sequencer only when the onsite (diesel-generator) power source is being used. The necessary loads are connected simultaneously (block load) when the offsite power source is in use.

QUESTION 040.170

"The response to 040.70 attempts to justify sharing of the dc systems of Unit 1 and 2, which is not permitted by Regulatory Guide 1.81, with the statement, 'The provision of the interconnection of the power supplies increases the overall reliability of the dc system since power could be obtained from the non-redundant dc bus of the other unit if necessary.' As noted in the FSAR, administrative controls are the only restrictions provided to prevent the interconnection of dc bus 111 (112) of Unit 1 to dc bus 211 (212) of Unit 2.

"The staff believes that any design having interconnections between Class 1E batteries in a multi-unit plant will affect the reliability and availability of dc power to both units. Therefore, the applicant is required to meet the staff position, i.e., 'dc systems in multi-unit nuclear power plants should not be shared' as per P.G. 1.81, position C.1."

RESPONSE

The Applicant also believes that the provisions of administratively controlled, manually actuated, interconnections between the non-redundant Class 1E dc buses will affect (i.e., increase) the overall reliability and availability of the dc systems for each unit in that it will provide a means for manually providing power to a dc bus at a time when it would otherwise have to be out-of-service (e.g., to perform a battery discharge test during a refueling outage, to replace a damaged cell, etc.). We believe that the intent of Regulatory Guide 1.81 (Position C.1) was to disallow "normal" sharing of dc systems, not to disallow the temporary connection of one dc bus to a source in the other unit during periods of testing and/or maintenance. That this was the intent is evident from the "Discussion," in Regulatory Guide 1.81, Part B, second paragraph, first sentence, which reads as follows:

"Sharing of onsite power systems at multi-unit power plant sites generally results in a reduction in the number and capacity of the onsite power sources to levels below those required for the same number of units located at separate sites."

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The "interconnection" provided in the Byron/Braidwood design does not result in a reduction in either the number or the capacity of the dc power sources . . . , i.e., the number and capacity of the dc power sources for each of the two units are exactly the same as they would be if the units were located at separate sites.

The Applicant therefore believes (a) that the Byron/Braidwood design fully complies with the intent of Position C.1 of Regulatory Guide 1.81, and (b) that the added reliability, availability, and flexibility of operation afforded by the interconnection justifies its retention.

The Technical Specification will preclude closing the crosstie during any mode of operation except Mode 5 (Cold Shutdown) and Mode 6 (Refueling). Closing the crosstie (when not allowed) will be administratively controlled by locking the breaker in the open position.

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QUESTION 040.171

"The response to question 040.65 refers to Mr. H. K. Stolt's letter of December 7, 1976 which argues against identification of cables by marking every five feet which was a proposed modification for IEEE Standard 384-1974. Stolt argues that (1) there is no technical or safety-related basis for requiring the cable to be marked in the fashion described, (2) marking the cable does not justify the costs, and (3) coloring and cable jacket could degrade the quality of the cable.

"Stolt mentions the negative aspects of cable identification without offering any alternative method of cable marking to facilitate visual verification that the cable installation is in conformance with separation criteria.

"The FSAR proposes to identify cables with a unique number of color-coded tags at each terminating point inside electrical equipment and in all intervening manholes, cable rooms, pull boxes, etc. This system of identification will leave many thousand feet of unmarked black-jacketed safety-related cable installed in cable trays throughout the plants, and it will be extremely difficult to verify that adequate separation is maintained between redundant discussed cables.

"The proposed method of cable identification is not an acceptable alternative to the requirements of Regulatory Guide 1.75. The applicant should review his proposed means of cable identification and submit a method of cable marking that will aid ready visual verification that the cable installation is in conformance with separation criteria."

RESPONSE

As stated in Subsection 8.3.1.3.4 "All power, control, and instrumentation cables are identified by a unique number of permanent color-coded tags at each terminating point in switchboards, switchgear, motor-control centers, motor conduit boxes, control cabinets, equipment cubicles, etc., and in all intervening manholes, cable rooms, pull boxes, etc. The tags shall be color-coded as in Table 8.3-4, allowing positive identification of safety-related cables."

The concern expressed in Question 040.171 is that this system of identification does not facilitate "visual verification that the cable installation is in conformance with the separation criteria." Additional verification can be obtained as described below.

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The cables installed at Byron/Braidwood can be divided into two categories: (a) those purchased under separate cable specifications (representing approximately 95% of the cables), and (b) those supplied by equipment vendors with the equipment (representing approximately 5% of the cables).

- a. The cables purchased by the separate cable specifications are supplied with the following information marked on the jacket:

Manufacturer, number of conductors, conductor size, voltage rating, specification number, purchase order number, cable type code, station reel number and sequential footage.

These markings are imprinted on the jacket every 30 inches for power and control cable and every 24 inches for instrumentation cable.

After the Electrical Installation Contractor installs a cable, the Contractor records (in addition to other information) the cable reel number, the beginning footage marker and the end footage marker on the cable pull card. A Cross Reference Index will be available at each site which will list (in sequence) the cable type code, cable reel number, cable footage markings, cable number, and the cable segregation code. This index will allow an inspector to verify the installation (including separation) of any cable by using the following procedure:

- 1) For the selected cable, note the cable type code, reel number, and sequential footage markings on the jacket.
 - 2) Enter the Cross Reference Index at the cable type code and cable reel number. Locate the corresponding sequential footage, opposite which will be both the cable number and segregation code.
 - 3) Compare the segregation code in the index with the segregation code marked on the cable tray in which the cable was found.
 - 4) If the segregation code of the cable (in the cross-index) agrees with the mark on the tray, the cable separation is correct. If the segregation code of the cable (in the cross-index) does not agree with the mark on the tray, the cable segregation is not correct.
- b. The cables supplied by equipment vendors are identified as described in Subsection 8.3.1.3.4 (first paragraph of response above).

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QUESTION 040.172

"In the response to Q040.93, paragraph 2 states that surveillance testing meets requirements of applicable NRC guides. However, the testing procedures specified for the diesel generator in the Technical Specification subparagraph 4.8.1.1.2.c are not in agreement with the requirements of Regulatory Guide 1.108 positions C.2.2.1, C.2.a.2, C.2.a.3, C.2.a.4, C.2.a.5, and C.2.a.6. Correct the tech specs to conform to the above."

RESPONSE

Surveillance testing will conform to the requirements of Regulatory Guide 1.108 except for the requirements in Regulatory Positions C.2.a.5, C.2.b, C8, C9c, and C9d.

Appendix A (Regulatory Guide 1.108) will be revised to identify the specific exceptions and to provide a detailed description of Applicant's position and justification for same. The appropriate sections of the Technical Specifications will be revised accordingly.

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QUESTION 040.173

"The response to Q040.69 is incomplete. Provide the ampere-hour rating of each 125V dc battery and the KVA output rating of the battery charger."

RESPONSE

a. Batteries

1. Each of the (two/unit) 125-V Class 1E batteries consists of 58 Gould Type NCX-1200 lead-calcium cells. The (nominally 1,200 AH) discharge characteristics, and other specifications for these cells are shown on the attached Gould Catalog Sheets (2).
2. Battery capacity is also addressed in Question 040.182.

b. Chargers

Each of the (two/unit) 125-V Class 1E battery chargers is a Power Conversion Products, Model No. 3S-130-400. The output rating of each charger is 400 amperes at 130 volts (nominal) dc. This is equivalent to a rating of 52 kW (nominal).

TYPE: NCX

**CAPACITIES—600 A.H. TO 2550 A.H.
@ 8 HOUR RATE TO 1.75 V.P.C. AVERAGE**

SPECIFICATIONS

- Container**—Styrene-Acrylonitrile Plastic.
- Cover**—Acryl.-Buta.-Styr. Terpolym. Plastic.
- Separators**—Microporous Material.
- Retainers**—Fiberglass Mats.
- Posts**—See Below.®
- Post Seals**—Floating O-Ring—Seal Nut.
- Vents**—Gould "Pre-Vent".**
- Level Lines**—High and Low—All Jar Faces.
- Electrolyte**—Height Above Plates—2-3/4".
- Electrolyte Withdrawal Tubes**—Each Cell.
- Sediment Space**—1-1/16".
- Specific Gravity**—1.215 @ 77°F. (25 C.).
- Inter-Cell Connectors**—Lead Plated Copper.

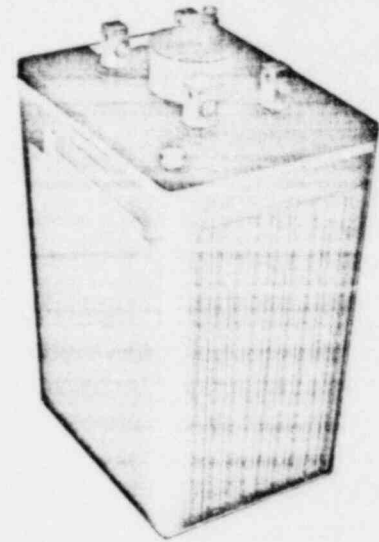
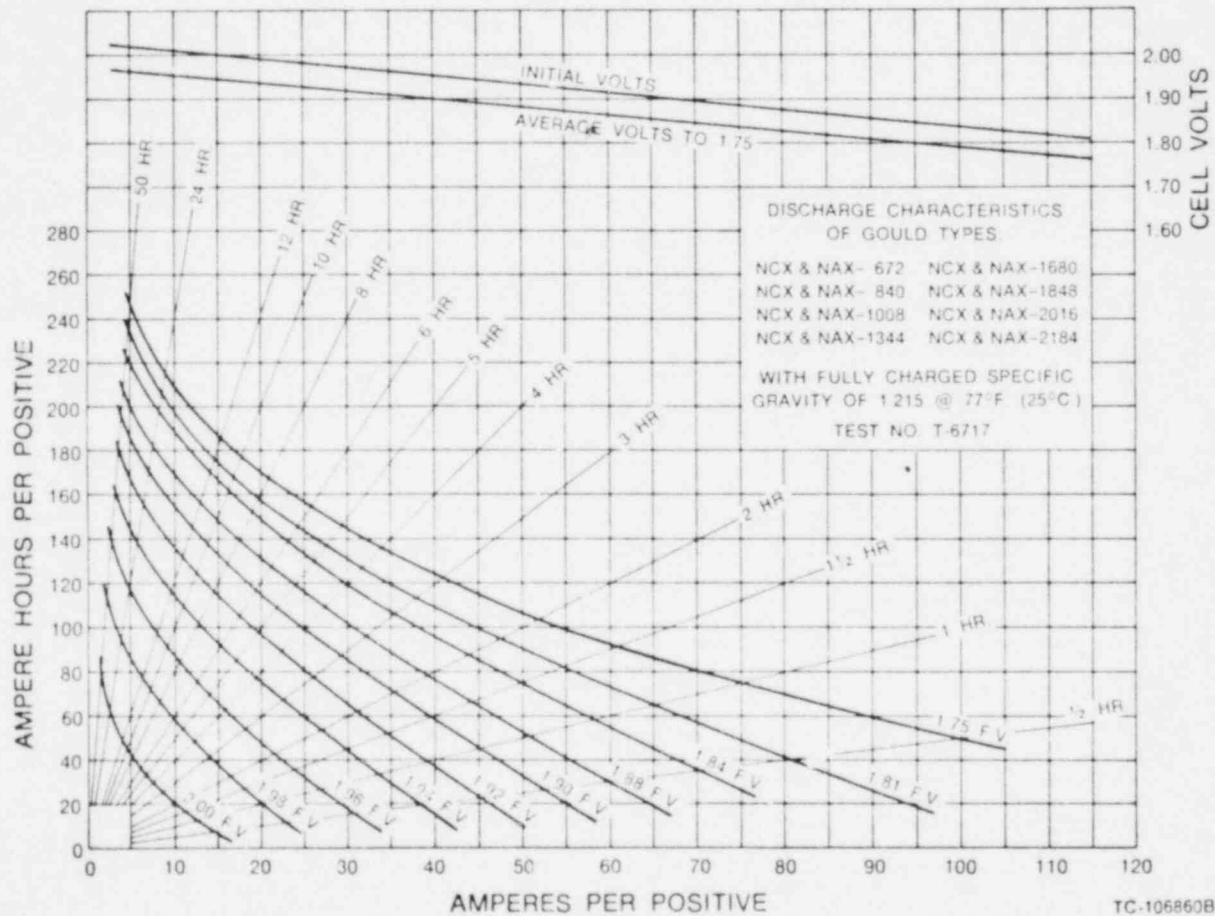
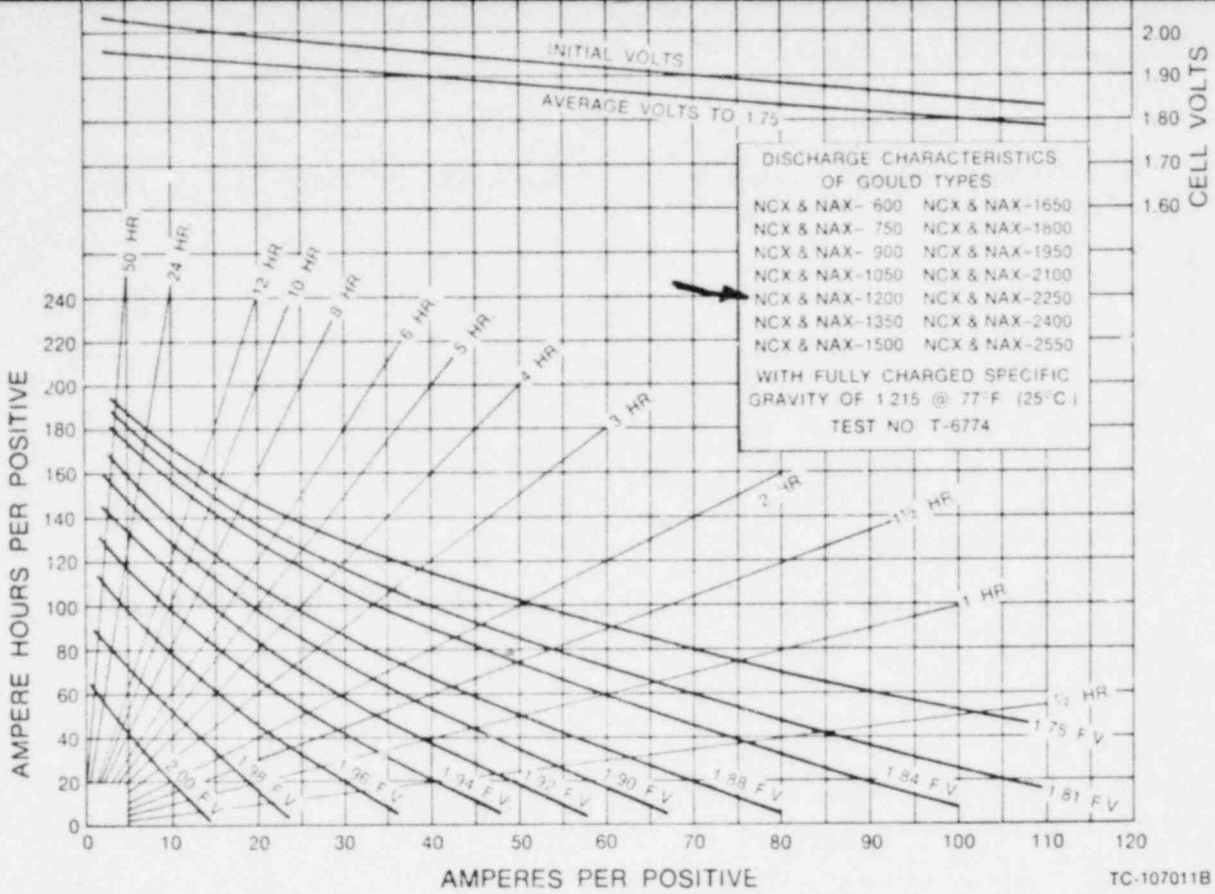


Plate Dimensions	Height	Width	Thick-ness
Positive Plate	15"	12 1/2"	.320
Negative Plate	15"	12 1/2"	.215

® Posts—600 A.H. to 1200 A.H. Two—1 1/2" square. 1344 A.H. to 1950 A.H. Four—1" square. (Except 1848 A.H.) 1848 A.H. to 2550 A.H. Four—1 1/2" square.

Type	Plates Per Cell	Ampere Hour Capacities to 1.75 V.P.C. Average*				1 Minute Rate in Amperes*		Overall Dimensions in inches			Approximate Wgt. in Lbs.		Elect. Gals. Per Cell
		8 Hr.	5 Hr.	3 Hr.	1 Hr.	To 1.75 V.P.C. Avg.	To 1.50 V.P.C. Avg.	L	W	H	Net Wgt.	Packed Wgt.	
NCX-600	9	600	540	468	300	712	1355	7-3/8	14-1/2	22-1/8	177	189	6.0
NCX-672	9	672	588	492	300	636	1210	7-3/8	14 1/2	22-1/8	178	190	6.0
NCX-750	11	750	675	585	375	880	1675	7-3/8	14-1/2	22-1/8	195	207	5.6
NCX-840	11	840	735	615	375	790	1500	7-3/8	14-1/2	22-1/8	196	208	5.6
NCX-900	13	900	810	702	450	1044	1985	7-3/8	14-1/2	22-1/8	213	225	5.1
NCX-1008	13	1008	882	738	450	942	1790	7-3/8	14-1/2	22-1/8	214	226	5.1
NCX-1050	15	1050	945	819	525	1204	2290	7-3/8	14-1/2	22-1/8	231	243	4.9
NCX-1200	17	1200	1080	936	600	1360	2515	7-3/8	14-1/2	22-1/8	249	261	5.0
NCX-1344	17	1344	1176	984	600	1240	2360	9-1/4	14-1/2	22-1/2	268	280	6.8
NCX-1350	19	1350	1215	1053	675	1494	2840	9-1/4	14-1/2	22-1/2	282	294	6.3
NCX-1500	21	1500	1350	1170	750	1620	3080	9-1/4	14-1/2	22-1/2	301	313	6.0
NCX-1650	23	1650	1485	1287	825	1782	3290	11-3/8	14-1/2	22-1/2	348	366	8.0
NCX-1680	21	1680	1470	1230	750	1530	2910	11-3/8	14-1/2	22-1/2	332	350	8.3
NCX-1800	25	1800	1620	1404	900	1932	3675	11-3/8	14-1/2	22-1/2	364	382	7.6
NCX-1848	23	1848	1617	1353	825	1661	3160	14-9/16	14-1/2	22-1/2	397	415	12.6
NCX-1950	27	1950	1755	1521	975	2080	3955	11-3/8	14-1/2	22-1/2	380	398	7.3
NCX-2016	25	2016	1764	1476	900	1788	3400	14-9/16	14-1/2	22-1/2	415	433	12.1
NCX-2100	29	2100	1890	1638	1050	2240	4260	14-9/16	14-1/2	22-1/2	446	464	11.5
NCX-2184	27	2184	1911	1599	975	1924	3660	14-9/16	14-1/2	22-1/2	433	451	11.5
NCX-2250	31	2250	2025	1755	1125	2400	4565	14-9/16	14-1/2	22-1/2	462	480	10.9
NCX-2400	33	2400	2160	1872	1200	2560	4865	14-9/16	14-1/2	22-1/2	479	497	10.3
NCX-2550	35	2550	2295	1989	1275	2720	5170	14-9/16	14-1/2	22-1/2	496	514	9.7

* Includes voltage drop across intercell connections used in standard layouts. ** TM Gould Inc.



Gould Inc., Industrial Battery Division
2050 Cabot Boulevard West, Langhorne, Pa. 19047
Telephone (215) 752-0555

CABLE: GOULNATBAT, LANGHORNE, PA - TWX GOULD LAHN. 510-667-2056



B/B-FSAR

QUESTION 040.174

"The FSAR does not provide the time interval for the diesel generator to attain rated voltage and frequency, and ready to accept load. Provide this information. Also describe the conditions which will initiate automatic starting of the onsite diesel generators."

RESPONSE

The diesel generator is designed to attain rated voltage and frequency and be ready to accept load 10 seconds after the receipt of an automatic start signal. Automatic starting is initiated by a safety-injection signal and/or a loss of offsite power signal. The safety-injection signal comes from the reactor protection system logic and the loss of offsite power signal originates from under-voltage relays on the 4160-V ESF bus which the diesel-generator feeds. The conditions which will initiate automatic starting of the diesel-generator are described in Subsection 8.3.1.1.1, pages 8.3-5 and 8.3-6 of the FSAR.

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QUESTION 040.175

"Your response to Q040.6 reflects a misunderstanding of the question. The purpose of the Q040.6 was to require that one alarm be provided to explicitly indicate conditions such that a diesel generator is incapable of responding to an automatic emergency start signal. As outlined, you should separate the alarms so that there is a dedicated alarm for the conditions that render a diesel incapable of responding to an automatic start signal. Some of the examples are as follows:

- a) Diesel generator breaker feeding to the ESF bus is racked out.
- b) Diesel generator is started locally for test or maintenance and the operator forgets to return the control to the control room after the activity is completed.
- c) Failure of DC control power supply to the diesel generator breaker.
- d) Failure of the control power supply to the start air train circuits.
- e) After testing the diesel generator, the operator forgets to reset the exciter pushbutton.

"You may have more than the conditions as mentioned above which would render a diesel generator incapable of responding to an automatic emergency start signal.

"We require that you provide a list of all these conditions and that your design comply with Regulatory Guide 1.47 'Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.'"

RESPONSE

In order to meet the requirements of Regulatory Guide 1.47 and inform the operator at the system level of conditions that would render a diesel generator unable to respond to an automatic emergency start signal, the following inputs are provided to the Byron/Braidwood Station's Equipment Status Display (ESD) systems (the listing is for Unit 1; the Unit 2 listing is identical):

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- a. Diesel Generator A Output Breaker Not Available.
- b. Diesel Generator B Output Breaker Not Available.
- c. Diesel Generator A Engine/Generator DC Control Power Off.
- d. Diesel Generator B Engine/Generator DC Control Power Off.
- e. Diesel Generator A Control Switch Maintenance Lockout.
- f. Diesel Generator B Control Switch Maintenance Lockout.
- g. Diesel Generator A Cooling Water Flow Low.
- h. Diesel Generator B Cooling Water Flow Low.
- i. Diesel Oil Storage Tank A Level Low.
- j. Diesel Oil Storage Tank B Level Low.
- k. Diesel Oil Storage Tank C Level Low.
- l. Diesel Oil Storage Tank D Level Low.
- m. Diesel Oil Day Tank A Level Low.
- n. Diesel Oil Day Tank B Level Low.

The ESD system level indication for the diesel generators is dedicated to these inputs only.

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QUESTION 040.176

"Your response to Q040.5 is not sufficient to allow a fully independent review of this aspect of your design. We require that the adequacy of station electric distribution system voltages should meet the following criteria.

A. Background

Events at the Millstone station have shown that adverse effects on the Class 1E loads can be caused by sustained low grid voltage conditions when the Class 1E buses are connected to offsite power. These low voltage conditions will not be detected by the loss of voltage relays (loss of offsite power) whose low voltage pickup setting is generally in the range of .7 per unit voltage or less.

The above events also demonstrated that improper voltage protection logic can itself cause adverse effects on the Class 1E systems and equipment such as spurious load shedding of Class 1E loads from the standby diesel generators and spurious separation of Class 1E systems from offsite power due to normal motor starting transients.

A more recent event at Arkansas Nuclear One (ANO) station and the subsequent analysis performed disclosed the possibility of degraded voltage conditions existing on the Class 1E buses even with normal grid voltages, due to deficiencies in equipment between the grid and the Class 1E buses or by the starting transients experienced during certain accident events not originally considered in the sizing of these circuits.

B. Branch Technical Position

1. In addition to the undervoltage scheme provided to detect loss of offsite power at the Class 1E buses, a second level of undervoltage protection with time delay should also be provided to protect the Class 1E equipment; this second level of undervoltage protection shall satisfy the following criteria:
 - a) The selection of undervoltage and time delay setpoints shall be determined from an analysis of the voltage requirements of the Class 1E loads at all onsite system distribution levels;

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- b) Two separate time delays shall be selected for the second level of undervoltage protection based on the following conditions:
- 1) The first time delay should be of a duration that establishes the existence of a sustained degraded voltage condition (i.e., something longer than a motor starting transient). Following this delay, an alarm in the control room should alert the operator to the degraded condition. The subsequent occurrence of a safety injection actuation signal (SIAS) should immediately separate the Class 1E distribution system from the offsite power system.
 - 2) The second time delay should be of a limited duration such that the permanently connected Class 1E loads will not be damaged. Following this delay, if the operator has failed to restore adequate voltages, the Class 1E distribution system should be automatically separated from the offsite power system. Bases and justification must be provided in support of the actual delay chosen.
- c) The voltage sensors shall be designed to satisfy the following applicable requirements derived from IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations":
- 1) Class 1E equipment shall be utilized and shall be physically located at and electrically connected to the Class 1E switchgear.
 - 2) An independent scheme shall be provided for each division of the Class 1E power system.
 - 3) The undervoltage protection shall include coincidence logic on a per bus basis to preclude spurious trips of the offsite power source;
 - 4) The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage set point and time delay limits (cited in item 1.b.2 above) have been exceeded;

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- 54 Capability for test and calibration during power operation shall be provided.
 - 64 Annunciation must be provided in the control room for any bypasses incorporated in the design.
- d4 The Technical Specifications shall include limiting conditions for operations, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the second-level voltage protection sensors and associated time delay devices.
2. The Class 1E bus load shedding scheme should automatically prevent shedding during sequencing of the emergency loads to the bus. The load shedding feature should, however, be reinstated upon completion of the load sequencing action. The technical specifications must include a test requirement to demonstrate the operability of the automatic bypass and reinstatement features at least once per 18 months during shutdown.

In the event an adequate basis can be provided for retaining the load shed feature during the above transient conditions, the setpoint value in the Technical Specifications for the first level of undervoltage protection (loss of offsite power) must specify a value having maximum and minimum limits. The basis for the setpoints and limits selected must be documented.

3. The voltage levels at the safety-related buses should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power sources by appropriate adjustment of the voltage tap settings of the intervening transformers. The tap settings selected should be based on an analysis of the voltage at the terminals of the Class 1E loads. The analyses performed to determine minimum operating voltages should typically consider maximum unit steady state and transient loads for events such as a unit trip, loss of coolant accident, startup or shutdown; with the offsite power supply (grid) at minimum anticipated voltage and only the offsite source being considered available. Maximum voltages should be analyzed with the offsite power supply (grid) at maximum expected voltage concurrent with minimum unit loads (e.g.,

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cold shutdown, refueling). A separate set of the above analyses should be performed for each available connection to the offsite power supply.

4. The analytical techniques and assumptions used in the voltage analyses cited in item 3 above must be verified by actual measurement. The verification and test should be performed prior to initial full power reactor operation on all sources of offsite power by:

- a) loading the station distribution buses, including all Class 1E buses down to the 120/208 v level, to at least 30%;
- b) recording the existing grid and Class 1E bus voltages and bus loading down to the 120/208 volt level at steady state conditions and during the starting of both a large Class 1E and non-Class 1E motor (not concurrently);

Note: To minimize the number of instrumented locations, (recorders) during the motor starting transient tests, the bus voltages and loading need only be recorded on that string of buses which previously showed the lowest analyzed voltages from item 3 above.

- c) using the analytical techniques and assumptions of the previous voltage analyses cited in item 3 above, and the measured existing grid voltage and bus loading conditions recorded during conduct of the test, calculate a new set of voltages for all the Class 1E buses down to the 120/208.
- d) compare the analytically derived voltage values against the test results.

"With good correlation between the analytical results and the test results, the test verification requirement will be met. That is, the validity of the mathematical model used in performance of the analyses of item 3 will have been established; therefore, the validity of the results of the analyses is also established. In general the test results should not be more than 3% lower than the analytical results; however, the difference between the two when subtracted from the voltage levels determined in the original analyses should never be less than the Class 1E equipment rated voltages."

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RESPONSE

There are two redundant and independent 4-kV emergency buses and each has two levels of undervoltage protection: 1) loss of power, and 2) degraded grid voltage. The relays will be connected to the existing potential transformers on the bus. The first level of undervoltage protection is provided by induction disk type undervoltage relays. The second level of undervoltage protection is provided by instantaneous undervoltage relays with delayed drop-out.

The voltage and time set points will be determined from an analysis of the voltage requirements of the safety-related loads and actual field measurements of bus voltages under various motor-starting conditions. The approximate pick-up voltage for the first level of protection is 70% of rated voltage. The preliminary setting for the second level of undervoltage protection is 92% of rated voltage. There is a 10 second time delay to avoid alarms on transients, and if the degraded voltage is not corrected within 5 minutes, the bus will automatically disconnect from the offsite power source and connect to its onsite diesel generator.

During a sustained degraded grid voltage condition, the subsequent occurrence of a SI (accident) signal will (immediately) trip the offsite power supply to the 4 kV ESF buses.

Testing will be conducted during refueling outages so spurious trips during testing will not affect plant operation.

The circuit will be designed to prevent automatic load shedding of the emergency power buses once the onsite sources are supplying power to all sequenced loads on the buses. The load shed interlock feature will use the "b" contact of the respective diesel generator breaker. This interlock will defeat the load shedding feature while the loads are being fed from the onsite power source. The load shed feature will be reinstated when the diesel generator breaker is open and the loads are fed from the offsite source.

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QUESTION 040.177

"You have not responded to parts (b) and (c) of question Q040.85. Provide this information. In part (b) of Q040.85, you were asked to show the differential relay in the D/G breaker trip circuit and show how this trip is retained on receipt of a safety injection signal. Provide a revised logic diagram 8.3.2."

RESPONSE

As stated in our response to Q040.85, Figure 8.3-2 has been revised to correct the breaker control logic. However, the differential relay is not shown in the breaker trip circuit since it is not directly in the circuit. The differential relay actuates the lock-out relay (shown as device 486-1413) which trips the breaker. The logic diagram correctly shows the lockout relay directly tripping the breaker while the remaining breaker trips (overcurrent, reverse power, the diesel generator shutdown relay) are "ANDED" with a Safety Injection "NOT" signal. In this way, the differential trip is retained during normal and accident (safety injection) conditions and the remaining trips are bypassed by the safety injection signal.

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QUESTION 040.178

"Your response to Q040.73 is not adequate. You state in Appendix A, Page A1.75-1, that you comply with Regulatory Guide 1.75. Regulatory Position C.1 of Reg. Guide 1.75 states that breakers that trip on receipt of a signal other than one derived from the fault current or its effects (e.g., an accident signal) are acceptable isolation devices. Your approach of coordinating the circuit breaker (which feeds the Non-Class 1E load from Class 1E bus) with the upstream circuit breaker is not in accordance with the above position and, therefore, not acceptable. Acceptable alternate approaches to the above positions would be two breakers or breaker and a fuse in series.

"Describe in detail the isolation devices being used on Byron/Braidwood Stations at all voltage levels (4.16 kV, 480V, 120V ac, 125V dc)."

RESPONSE

The current issue of Appendix A (Page A1.75-1) does not say that Applicant complies with Regulatory Guide 1.75. We agree that Appendix A (Regulatory Guide 1.75 position description) requires clarification. The Applicant's position relative to Regulatory Guide 1.75 (Rev. 2, September 1978) is described in revised Appendix A.

There are no non-safety-related loads supplied from the safety-related 4.16-kV buses other than non-safety-related buses 143 and 144 which may be manually connected to safety-related buses 141 and 142 (respectively) as described in the last paragraph of Subsection 8.3.1.1 (page 8.3-2).

Non-safety-related loads supplied from safety-related buses at 480-V, 120 Vac, and 125 Vdc utilize circuit breakers (actuated by fault current) for isolation, all as described in revised Appendix A.

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QUESTION 040.179

"The response to Q040.49 included single line diagrams for the Byron and Braidwood Stations. Figures Q040.49-3 and Q040.49-4 do not match with the figures 8.2-1, 8.2-2, 8.2-6 and 8.2-7 provided with the FSAR. Correct this discrepancy and provide corrected drawings.

RESPONSE

Figures Q040.49-3 and Q040.49-4 have been corrected and transmitted.

QUESTION 040.180

"We are still waiting for your response to 040.83.
Provide your response."

RESPONSE

Applicant will formally answer Question 040.83.

QUESTION 040.181

"IEEE-387-77 Section 5.6.2.2(1) and Regulatory Guide 1.108 position C.1.b.3 recommend that the periodic testing of diesel generator units should not impair the capability of the unit to supply emergency power within the required time. The following discussion provides the necessary guidelines (requirements) to meet the objective of the above referred Industry Standard and Regulatory Guide.

"The diesel generator unit design should include an emergency override of the test mode to permit response to bona fide emergency signals and return control of the diesel-generator unit to the automatic control system. A design which does not have such a feature would necessitate operator action of varying levels of complexity depending on the specific design and plant conditions, in order to enable a diesel generator in the test mode to respond to a bona fide emergency signal. The concern here is the possible consequent disabling of a D/G due to operator's inaction or wrong action thereby reducing the available safety margin with regard to onsite a-c power at a time when it is most needed.

"Each diesel generator must be periodically tested at a frequency as specified in R.G. 1.108. This test frequency is normally once per month but could be as high as once every three days. The duration of each test is one hour. During a normal successful test the diesel generator is loaded on its bus with the governor operating in a droop mode, and the load carried by the diesel engine is a function of governor speed and speed droop setting. In order to enable a diesel generator in the test mode to respond to a bona fide emergency signal, the design must incorporate the following features for the stated plant conditions:

1. Accident Conditions

During the periodic testing of a diesel generator, if a safety injection signal is generated, the diesel generator breaker should be tripped automatically. This will permit the unit to be cleared from parallel operation with the system and enable the diesel generator to attain the emergency standby mode.

In this mode of operation, the diesel governor control should change automatically to the isochronous mode which will maintain the engine running at a synchronous speed corresponding to 60 Hz at the generator terminals. All noncritical protective trips except engine overspeed and generator differential should be bypassed for

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this accident condition. Additionally, the voltage regulator should change to the automatic mode thereby maintaining the generator at a preset constant voltage. With the above actions complete, the diesel generator unit will be ready to accept the required load in the event of a loss of offsite power.

2. Loss of Offsite Power (LOOP) Conditions

Normally, during periodic testing of diesel generator, the diesel generator is paralleled with the offsite power system. During such a test, should a LOOP occur, a LOOP signal would probably not be generated because the D/G would attempt to provide power to the bus and to the offsite system through the closed offsite power feedbreaker. In this case, the D/G breaker will trip on overcurrent or underfrequency and in some designs the D/G breaker also locks out for this condition. To assure the continued availability of the D/G unit it is essential that the diesel generator breaker should not be locked out for such overload conditions. At the same time, the governor is shifted automatically from droop to isochronous mode and the voltage regulator to automatic mode. With the above actions complete, the diesel generator unit will be ready to accept its required load for LOOP conditions.

3. Accident Conditions/LOOP

For simultaneous accident/LOOP condition or sequential accident and LOOP condition, the requirements stated in items 1 and/or 2 would be adequate to assure the restoration of the diesel generator from the test mode to the emergency mode.

Provide a discussion how you meet the above stated guidelines."

RESPONSE

We agree with the guidelines set forth in this question to meet the requirements of IEEE-387 Section 5.6.2.2(1) and Regulatory Guide 1.108 Position C.1.b.3. The Byron/Braidwood diesel-generator system design includes emergency override of the test mode for both accident conditions (Safety Injection) and loss of offsite power (LOOP) to permit response to bona fide emergency signals and return control of the diesel-generator to the automatic control system. Upon

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receipt of either a safety injection signal or a loss of offsite power signal, the governor is automatically shifted from droop to the isochronous mode and the voltage regulator is shifted to the automatic mode. Therefore, the diesel-generator breaker controls will be revised to trip the breaker upon receipt of a safety injection signal concurrent with the diesel generator operating in the test mode. Also, the Byron/Braidwood diesel-generator breaker control scheme does not lock out the diesel-generator breaker on overcurrent or underfrequency which assures the availability of the diesel generator to accept its required load for LOOP conditions.

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QUESTION 040.182

"The description of the dc Power System in the FSAR is incomplete and should be expanded. Except for a general statement to the effect that the batteries are sized to carry their connected loads for the time periods shown in Table 8.3-5, there is no information provided about the capacity of the batteries. Another area with insufficient information is the subsection on Non-Safety-Related 125V dc loads. Information about how the non-safety loads are separated from the Class 1E bus is not included in the FSAR text or drawings.

"Expand the section on the dc Power System to provide an adequate description of the battery capacity and how the non-safety loads are connected to and separated from the Class 1E dc systems."

RESPONSE

- a. It is unusual for the NRC to request that specific/detailed electrical equipment rating/capacity data be included in the text of the FSAR (i.e., such detailed system descriptions/rating information is not required by Regulatory Guide 1.70, Standard for SAR). The following will supplement the detailed description of the dc systems contained on pages 8.3-23 through 8.3-26 of the FSAR; however, Applicant does not propose to add same to the FSAR:

DC Power Systems

The station dc systems supply power to the plant instrumentation and control under all modes of plant operation. In addition, upon loss of ac power, the dc systems provide power for emergency lighting and certain turbine-generator auxiliary motors.

The dc systems for each unit consist of one 250-V and one 125-V non-Class 1E battery system and two independent and redundant Class 1E 125-V battery systems. One (common for two units) 48-V non-Class 1E battery system is provided for the control of the non-Class 1E switchgear at the river screen house.

The Class 1E 125-V battery systems supply power to Class 1E loads without interruption during normal operations or DBA conditions. Each Class 1E 125-Vdc system consists of one battery, one main distribution bus with molded-case

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circuit breakers, one static battery charger, and local distribution panels. Redundancy and independence of components precludes the loss of both systems as a result of a single failure. For Unit 1, Battery 111 supplies ESF Division 11 load requirements; Battery 112 supplies ESF Division 12 load requirements. There are no bus ties, or sharing of power supplies, between redundant trains.

Each Class 1E 125-V battery, battery charger, and distribution panel associated with one ESF division is located in a seismic Category I room, physically separated from the redundant equipment. Electrical separation is also maintained to ensure that a single failure in one train does not cause failure in the redundant train. There is no sharing between redundant Class 1E trains of equipment such as batteries, battery chargers, or distribution panels.

Each Class 1E 125-Vdc system has the capacity to continuously supply all the connected normal running load while maintaining its respective battery in a fully charged condition. Each battery has a nominal rating of 1200 ampere-hours and is capable of carrying the various loads continuously, for the time periods indicated in Table 8.3-5 of the FSAR, in the event of a total loss of onsite and offsite ac power.

One 400 ampere capacity static battery charger supplied by a Class 1E MCC, is provided for each Class 1E 125-V battery system. Protection is incorporated in the battery chargers to preclude the ac supply source from becoming a load on the battery as a result of power feedback upon loss of ac input power. Backup protection is incorporated by an overvoltage relay mounted on the charger, which trips the charger supply and annunciates the tripped condition in the control room.

Each battery charger is capable of floating the battery on the bus or recharging a completely discharged battery within a 24-hour period while supplying the largest combined demands of the various steady-state loads under all plant operating conditions.

All battery areas are ventilated to prevent the accumulation of gases produced during charging operations. Each Class 1E 125-V battery area is provided with an independent safety-related ventilation system. A separate safety-related exhaust fan and duct is provided for each Class 1E battery area.

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In addition, the battery cells are provided with explosion-resistant vent caps that prevent the ignition of gases within the cell from an ignition source outside the cell.

The Class 1E dc systems are testable, independent, and conform to the requirements of Regulatory Guide 1.6 and 1.32. These systems meet the requirements of General Design Criteria 5, 17, and 18.

- b. The capacity of the Class 1E batteries and chargers is covered in the response to Question 040.173.
- c. Non-Safety-Related Loads Connected to Class 1E Bus
 - (1) The single line diagram of this distribution system is shown on Drawing 6E-0-4001 (referenced in FSAR, Subsection 8.3.2.1) and on the Station One Line Diagrams, 6E-1-4001A and 6E-2-4001A. This connection is described in Subsection 8.3.2.1.2.
 - (2) The isolation of the non-safety-related loads from the safety-related (Class 1E) bus, as the result of a fault on non-safety-related circuits, is performed by circuit breakers (2 in series), qualified to Class 1E requirements, operated by overcurrent.

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QUESTION 040.183

"The FSAR states in the seventh paragraph on page 8.3-6 that ESF buses 141 and 241 cannot be fed from the same SAT except when one of the two SAT's is unavailable and the removable links are manually relocated from the transformer secondary to the bus duct crosstie.

"If the above statement is true then there is a discrepancy with the bus ties shown on drawing Q040.49-3. The bus ties between bus 141 and 241 are made by circuit breakers 1414 and 2414. There is a similar bus tie between buses 142 and 242 by means of circuit breakers 1424 and 2424.

"These bus ties are in conflict with statements in the FSAR. Review the FSAR and provide a description of the bus connection that agrees with the drawings."

RESPONSE

The above referenced paragraph of the FSAR will be revised as follows:

The 4160-volt ESF buses 141 (241) and 142 (242) will not be fed from the same SAT (parallel operation) except when one of the unit's SAT's is unavailable, and the removable links are manually relocated from the transformer secondary to the bus duct crosstie.

QUESTION 331.33

"Provide additional information regarding the sensitivity of airborne radioactivity monitors in accordance with Section 1.3 of Regulatory Guide 1.70. Verify that the airborne radioactivity monitors described in Section 12.3.4 of the FSAR are capable of detecting 10 MPC-hours of particulate and iodine radioactivity in compartments which may be occupied and may contain airborne radioactivity (the acceptance criteria in Standard Review Plan Section 12.3)."

RESPONSE

Standard Review Plan Section 12.3 implies an assessment of potential contamination is required in "...any compartment which has a possibility of containing airborne radioactivity and which may be occupied by personnel."

A system of fixed Continuous Airborne Monitors (CAMs) is provided to monitor for airborne radioactivity in compartments which may be occupied and may contain airborne radioactivity. Since there are too many rooms or cubicles to monitor independently, a limited number of CAMs are provided to continuously monitor the air from selected branch exhaust ducts of the HVAC system. Thus, the exhaust from a single room may be diluted by the exhaust from other rooms before the air gets to the monitoring point. Therefore, the monitor must be sensitive enough to respond to the diluted activity. The maximum possible dilution factor for any cubicle is:

$$DF = \frac{F \text{ cubicle (cmf)}}{F \text{ duct (cfm)}}$$

and using the Detectability Factor for MPC_a (DMPC_a) given in Table Q331.33-1. We can write an expression for the time, T, it takes to detect the presence of MPC_a levels in the exhaust ducts:

$$T = 1 / (DMPC_a * DF), \text{ (hrs)} \quad (1)$$

where,

- T = time to detect particulate and iodine MPC_a,
(hr)
- DMPC_a = detectability factor for MPC_a (see Table Q331.33-1)
- DF = dilution factor for 10 MPC-HR detectability

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F_{cubicle} = flow in exhaust from cubicle, (cfm)

F_{duct} = flow in branch duct where monitor is located,
(cfm).

Table Q331.33-1 shows the sensitivity of the particulate, iodine, and noble gas channels for the isotopes of greatest interest. These sensitivities were compared to maximum permissible concentrations in air (MPC_a) of the most restrictive particulate and iodine radionuclides in the areas and cubicles of lowest ventilation flow rate. The criterion used was that airborne radioactivity from the areas described above and having an activity concentration of MPC_a , would be detected within 10 hours. Exhaust flow rates from cubicles and in branch ducts were examined to determine dilution factors for this assessment. The exhaust flow rates for the monitored branch ducts and the individual room exhaust flow rates are given on the HVAC drawings in Section 9.4. The location of the radiation monitors are also shown on these drawings. An investigation using the above data indicates that the system is capable of detecting 10 MPC_a -hrs of airborne particulate and iodine radioactivity in the rooms, cubicles and areas discussed above which may be occupied and may contain airborne radioactivity.

TABLE Q331.33-1

SENSITIVITY OF CONTINUOUS AIRBORNE MONITORING SYSTEM

<u>ISOTOPE</u>	<u>AVERAGE ENERGY (MEV)</u>	<u>GROSS SENSITIVITY (cpm/μCi)</u>	<u>SENSITIVITY (cpm/hr per μCi/cc)</u>	<u>MPC (μCi/cc)</u> ^a	<u>DETECTABILITY* FACTOR FOR MPC</u> ^a
I. Particulate Channel (Beta Scintillator)					
CO-60	0.096	4.69×10^5	2.01×10^{12}	9×10^{-9}	200
SR-90	0.209	1.20×10^6	5.10×10^{12}	1×10^{-9}	60
TC-99	0.085	4.47×10^5	1.92×10^{12}	6×10^{-8}	1350
CS-137	0.171	1.15×10^6	4.91×10^{12}	1×10^{-8}	575
II. Iodine Channel (NaI Spectrometry windowed on I-131 peak)					
I-131	0.364 (γ)	1.01×10^5	4.29×10^{11}	9×10^{-9}	850
III. Noble Gases Channel (Beta Scintillator)					
KR-85	0.100	-----	1.84×10^7 **	1.0×10^{-5}	40
XE-133	0.250	-----	3.6×10^7 **	1.0×10^{-5}	80

*The minimum detectable (activity) concentration is based on a signal count rate at a 95% confidence level as given by the formula in ANSI 13.10-1974 and modified for the GA system as follows:

$$\text{MDC} = 2 (\text{BCKG}/20)^{\frac{1}{2}} \div \text{Sensitivity, (for BCKG} \leq 100 \text{ cpm)}$$

$$2 (\text{BCKG}^2/2,000)^{\frac{1}{2}} \div \text{Sensitivity, (for } 100 \text{ cpm} < \text{BCKG} < 1 \times 10^5 \text{ cpm)}$$

Where BCKG is the total background counting rate (cpm). For the particulates 1905 cpm was used and for iodine and noble gases 100 cpm was used;

this criterion will yield an answer that has a 95% statistical confidence level.

**cpm per μ Ci/cc

3.6.2.3.3.4 Pipe Restraints and Locations

Pipe restraints and locations are discussed in Subsection 3.6.2.3.1.1 and 5.4.14.

3.6.2.3.3.5 Design Loading Combinations

As described in Section 3.9, the forces associated with rupture of reactor piping systems are considered in combination with normal operating loads and earthquake loads for the design of supports and restraints in order to assure continued integrity of vital components and engineered safety features.

The stress limits for reactor coolant piping and supports are discussed in Section 3.9.

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipes assemblies were not utilized in the design of either the Byron or Braidwood power stations.

3.6.2.5 Dynamic Analysis Applicable to Postulated High Energy Pipe Break

3.6.2.5.1 Reactor Coolant Loops

- a. Table 3.6-6 and Figure 3.6-24 identify the design-basis break locations and orientations for the main Reactor Coolant Loop.

The primary-plus-secondary stress intensity ranges and the fatigue cumulative usage factors at the design break locations specified in WCAP 8082 (8172) (Reference 1) are given in Table 3.6-7 for a reference fatigue analysis. The reference analysis has been prepared to be applicable for many plants. It uses seismic umbrella moments which are higher than those used in WCAP 8082 (8172) such that the primary stress is equal to the limits of Equation 9 in NB-3650 (Section III of the ASME Boiler and Pressure Vessel Code) at many locations in the system where in WCAP 8082 (8172) one location was at the limit. Therefore, the results of the reference analysis may differ slightly from WCAP 8082 (8172) but the philosophy and conclusions of the WCAP are valid. There are no other locations in the model used in the reference fatigue analysis, consistent with WCAP 8082 (8172), where the stress intensity ranges and/or usage factors exceed the criteria of $2.4 S_m$ and 0.2, respectively.

Additionally, the thermal transients used in the reference analysis, although different from those in WCAP-8082, are the same as the thermal transients which are specified for Byron/Braidwood in Subsection 3.9.1.1.

Actual plant moments for the Byron/Braidwood Units are also given in Table 3.6-7 at the design basis break locations so that the reference fatigue analysis can be shown to be applicable for this plant. By showing actual plant moments to be no greater than those used in the reference analysis, it follows that the stress intensity ranges and usage factors for the Byron/Braidwood Units will be less than those for comparable locations in the reference model. By this means it is shown that there are no locations other than those identified in WCAP 8082 (8172) where the stress intensity ranges and/or usage factors for the Byron/Braidwood Units might exceed the criteria of $2.4 S_m$ and 0.2, respectively. Thus, the applicability of WCAP 8082 (8172) to the Byron/Braidwood Units has been verified.

- b. Pipe whip restraints associated with the main Reactor Coolant Loop are described in Subsections 3.6.2.3.1.1 and 5.4.14.
- c. Jet deflectors associated with the main Reactor Coolant Loop are described in Subsection 3.6.2.3.1.2.
- d. Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Subsection 3.6.2.3.3.5. Pipe rupture loads include not only the jet thrust forces acting on the piping but also jet impingement loads on the primary equipment and supports.
- e. The interface between Sargent & Lundy and Westinghouse concerning the design of the primary equipment supports and the interaction with the primary coolant loop is described in Subsection 3.9.3.4.4.1.

3.6.2.5.2 Postulated Breaks in Piping Other than Reactor Coolant Loop

The following material pertains to dynamic analyses completed for piping systems other than the reactor main coolant piping which connects the reactor vessel, the main coolant pumps, and the steam generators.

3.6.2.5.2.1 Implementation of Criteria for Defining Pipe Break Locations and Configurations

The locations and number of design basis breaks, including postulated rupture orientations, for the high energy piping systems are shown in Figures 3.6-25 to 3.6-44.

The above information was derived from the implementation of the criteria delineated in Subsection 3.6.2.1.

Stress levels and usage factors (usage factors for Class 1 piping only) for the postulated break locations are shown in Tables 3.6-11 and 3.6-12.

3.6.2.5.2.2 Implementation of Criteria Dealing with Special Features

Special protective devices in the form of pipe whip restraints and impingement shields are designed in accordance with Subsection 3.6.2.3.

Pipe whip restraint locations, configuration, and orientations in relation to break locations for each applicable piping systems are shown in Figures 3.6-25 to 3.6-44.

Inservice inspection is discussed in Subsection 3.6.1.2.2.

3.6.2.5.2.3 Acceptability of Analyses Results

The postulation of break and crack locations for high and moderate energy piping systems and the analyses of the resulting jet thrust, impingement and pipe whip effects has conservatively identified areas where restraints, impingement shields, or other protective measures are needed and has yielded the conservative design of the required protective devices.

Results of jet thrust and pipe whip dynamic effects are given in Tables 3.6-13 and 3.6-14.

3.6.2.5.2.4 Design Adequacy of Systems, Components, and Component Supports

For each of the postulated breaks the equipment and systems necessary to mitigate the consequences of the break and to safely shut down the plant (i.e., all essential systems and components) have been identified (Subsection 3.6.1). The equipment and systems are protected against the consequences of each of the postulated breaks to ensure that their design-intended functions will not be impaired to unacceptable levels as a result of a pipe rupture or crack.

When it became necessary to restrict the motion of a pipe which would result from a postulated break, pipe whip restraints were added to the applicable piping systems, or structural barriers or walls were designed to prevent the whipping of the pipe.

Design adequacy of the pipe whip restraints is demonstrated in Tables 3.6-13 and 3.6-14. Data in the tables was obtained through use of the criteria delineated in Subsections 3.6.2.1 through 3.6.2.3 inclusive.

The design adequacy of structural barriers, walls, and components is discussed in Section 3.8.

3.7.3.1.1 Seismic Analysis Methods (Westinghouse)

This subsection describes the seismic analysis methods performed for safety-related components and systems supplied by Westinghouse.

Those components and systems that must remain functional in the event of the SSE (Seismic Category I) are identified by applying the criteria of Subsection 3.2.1. This equipment is classified into three types according to its dynamic characteristics. The analysis methods used for this equipment also depended on these classifications.

The first type is flexible equipment. This equipment is characterized by several modes in the frequency range that could produce amplification of the base input motion. The components which are classified as flexible equipment, i.e., with more than one mode below 33 Hz, are the steam generators, reactor coolant pumps, pressurizers, control drive mechanisms, reactor internals, and fuel. Dynamic analyses were performed for these components using modal analysis techniques with either the response spectrum method, integration of the uncoupled modal equations of motion, or by direct integration of the coupled differential equation of motion. Details of the methods used for these analyses are described in Subsections 5.3.7.3.1.1.1 through 3.7.3.1.1.5.

The second classification is rigid equipment. This equipment has a fundamental natural frequency that is sufficiently high (greater than 33 Hz) so that base input motions are not amplified. Such equipment is particularly suitable for static analysis as described in Subsection 3.7.3.1.1.6.

Finally, the third type of equipment is classified as limited flexible, with only one predominate mode in the frequency range subject to possible amplification of the input motion. The fundamental mode of this type of equipment is basically a translational bending mode at a frequency less than 33 Hz. The second mode is usually a rocking mode with a frequency greater than 33 Hz. Because of the simple response characteristics of the equipment, dynamic analysis techniques that account for multiple mode effects and closely spaced modes are not required. Therefore, this equipment was evaluated using static analysis methods as described in Subsection 3.7.3.1.1.6.

3.7.3.1.1.1 Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dashpots suitable for mathematical analysis. The essence of this step is to select a model so that the displacements obtained will be a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure or component. Some typical modeling techniques are presented in Reference 3.

Equations of Motion

Consider the multi-degree-of-freedom system shown in Figure 3.7-54. Making a force balance on each mass point r , the equations of motion can be written in the form:

$$m_r \ddot{y}_r + \sum_i c_{ri} \dot{u}_i + \sum_i k_{ri} u_i = 0 \quad (3.7-3)$$

where:

- m_r = the value of the mass or mass moment of rotational inertia at mass point r
- \ddot{y}_r = absolute translational or angular acceleration of mass point r
- c_{ri} = damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i , maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity

$$\frac{2}{(\Delta t + \Delta t_1)} M \left\{ \frac{x_{n+2} - x_{n+1}}{\Delta t} - \frac{x_{n+1} - x_n}{\Delta t_1} \right\} + \frac{1}{(\Delta t + \Delta t_1)} C \left\{ x_{n+2} - x_n \right\} + \frac{1}{3} [K] \left\{ x_{n+2} + x_{n+1} + x_n \right\} = \left\{ F_{n+2} \right\} \quad (3.7-27)$$

By factoring x_{n+2} , x_{n+1} , and x_n , and rearranging terms, Equation (3.7-28) is obtained as follows:

$$\left\{ c_5 [M] + c_3 [C] + (1/3) [K] \right\} \left\{ x_{n+2} \right\} = \left\{ F_{n+2} \right\} + \left\{ c_7 [M] - (1/3) [K] \right\} \left\{ x_{n+1} \right\} + \left\{ -c_2 [M] + c_3 [C] - (1/3) [K] \right\} \left\{ x_n \right\} \quad (3.7-28)$$

where:

$$c_2 = \frac{2}{\Delta t_1 (\Delta t + \Delta t_1)}$$

$$c_3 = \frac{1}{\Delta t + \Delta t_1}$$

$$c_5 = \frac{2}{\Delta t (\Delta t + \Delta t_1)}$$

$$c_7 = c_2 + c_5$$

The above set of simultaneous linear equations is solved to obtain the present values of nodal displacements $\{x_t\}$ in terms of the previous (known) values of the nodal displacements. Since $[M]$, $[C]$, and $[K]$ are included in the equation, they can also be time or displacement dependent.

3.7.3.1.1.6 Static Analysis - Rigid and Limited Flexible Equipment

Rigid equipment and limited flexible equipment as defined in Subsection 3.7.3.1.1 are generally analyzed using the static analysis method. This technique involves the multiplication of the total weight of the equipment or component member by a specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient was established on the basis of the excitation level that the component was expected to experience in the plant.

For rigid equipment, the seismic acceleration coefficients were compared with the high frequency (greater than 33 Hz) acceleration levels for the applicable response spectra developed for the plant to confirm the design analysis. The seismic acceleration coefficients for limited flexible equip-

ment are compared with the acceleration levels from the applicable response spectra at the calculated fundamental natural frequency of the component. If the design seismic acceleration coefficients for either rigid or limited flexible equipment are exceeded by the actual plant acceleration levels, the design analysis is performed again at the actual level to confirm the equipment adequacy.

3.7.3.1.2 Differential Seismic Movements of Interconnected Supports

Systems that are supported at points which undergo certain displacements due to a seismic event are designed to remain capable of performing their Seismic Category I functions. The displacements, obtained from a time-history analysis of the supporting structure, cause moments and forces to be induced into the piping system. Since the resulting stresses are self-limiting, it is justified to place them in the secondary stress category. Therefore these stresses exhibit properties much like a thermal expansion stress and a static analysis is used to obtain them.

flexible building, the number of cycles exceeding 90% of the maximum stress was not greater than three cycles.

This study was conservative since it was performed with single degree-of-freedom models which tend to produce a more uniform and unattenuated response than a complex interacted system. The conclusions indicate that 10 maximum stress cycles for flexible equipment (natural frequencies less than 33 Hz) and 5 maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz) for each of 20 OBE occurrences should be used for fatigue evaluation of Westinghouse systems and components.

3.7.3.3 Procedure Used for Modeling

Procedures used for modeling safety-related components and systems within Westinghouse's scope are discussed in Subsection 3.7.3.1.1.1.

Rigid valves (i.e., with natural frequencies greater than 33 Hz) are included in the piping system model as lumped masses on rigid extended structures. If it is shown, by test or analysis, that a valve is not rigid (one or more natural frequencies below 33 Hz), then a multimass, dynamic model of the valve, including the appropriate stiffnesses, is developed for use in the piping system model. The valve model used in the piping analysis is constructed such that its calculated frequencies correspond to those obtained by test.

For Class 1 piping systems, the actual calculated stiffness of pipe supports are included in the model of the piping system. In the modeling of Class 2 and 3 piping systems, pipe supports are represented as minimum rigid elements for which the stiffness is predetermined based upon the particular support type.

The mathematical model used for the dynamic analyses of the reactor coolant system is shown in Figure 3.9-1.

3.7.3.3.1 Modeling of the Piping System

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at nodes which are connected by weightless elastic members, representing the physical properties of each segment. The pipe lengths between mass points are not greater than the length which would have a natural frequency of 33 Hz when calculated as a simply supported beam. All concentrated weights on the piping system such as main valves, relief valves, pumps, and motors are modeled as lumped masses. The torsional effects of the valve operators and other equipment with offset center of gravity with respect to centerline of the pipe is included in the analytical model.

3.7.3.3.2 Field Location of Supports and Restraints

The field location of seismic supports and restraints for Seismic Category I piping and piping systems components is selected to satisfy the following two conditions:

- a. The location selected must furnish the required response to control strain within allowable limits.
- b. Adequate building strength for attachment of the components must be available.

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraints devices is made to ensure that the location and characteristics of these supports

rod drive mechanisms (CRDM's) and the fuel assemblies of the nuclear steam supply system, when used in seismic system analysis, are in conformance with the values for welded and/or bolted steel structures (as appropriate). For reactor internals analysis, Westinghouse uses 2% damping for OBE and 4% damping for SSE as given by Regulatory Guide 1.61.

Tests on fuel assembly bundles justified conservative component damping values of 7% for OBE and 10% for SSE to be used in the fuel assembly component qualification. Documentation of the fuel assembly tests is found in Reference 2.

The damping values used in component analysis of CRDM's and their seismic supports were developed by testing programs performed by Westinghouse. These tests were performed during the design of the CRDM support; the support was designed so that the damping in Table 3.7-2 could be conservatively used in the seismic analysis. The CRDM support system is designed with plates at the top of the mechanism and gaps between mechanisms. These are encircled by a box section frame which is attached by tie rods to the refueling cavity wall. The test conducted was on a full size CRDM complete with rod position indicator coils, attachment to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support. The internal pressure of the CRDM was 2250 psi and the temperature on the outside of the pressure housing was 400° F.

The program consisted of transient vibration tests in which the CRDM was deflected as specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variations in the drive shaft axial position, upper seismic support clearance, and initial deflection amplitude was investigated.

The upper support clearance had the largest effect on the CRDM damping with the damping increasing with increasing clearance. With an upper clearance of 0.06 inches, the measured damping was approximately 8%. The clearances in a typical upper seismic CRDM support is a minimum of 0.10 inch. The increasing damping with increasing clearances trend from the test results indicated that the damping would be greater than 8% for both the OBE and the SSE based on a comparison between typical deflections during these seismic events to the initial deflections of the mechanisms in the test. Component damping values of 5% are, therefore, conservative for both the OBE and the SSE.

These damping values are used and applied to CRDM component analysis by response spectra techniques.

3.7.3.15 Analysis Procedure for Damping

In instances of the equipment supplied by Westinghouse, either the lowest damping value associated with the elements of the system is used for all modes, or an equivalent modal damping

transient problem (the dynamic response of the system for the static equilibrium position). The time-history displacement solution of all dynamic degrees of freedom is obtained using subprogram FIXFM and employing 4% critical damping.

The loss-of-coolant accident displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the reactor coolant loop. The loss-of-coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions, cavity pressure loads, and loop mechanical loads. The reactor vessel analysis is described in Subsection 3.9.1.4.6.

The resultant asymmetric external pressure loads on the RCP and steam generator resulting from postulated pipe ruptures and pressure buildup in the loop compartments are applied to the same integrated RCL/supports system model used to compute loadings on the components, component supports, and RCL piping as previously discussed. The response of the entire system is obtained for the various external pressure loading cases considered. For each pipe break case considered, the equipment support loads and piping stresses resulting from the external pressure loading are added to the support loads and piping stresses calculated using the loop LOCA hydraulic forces and RPV motion.

The break locations considered for subcompartment pressurization are those postulated for the RCL LOCA analysis, as discussed in Section 3.6 and SCAP-5172 (Reference 1 of Section 3.6). The asymmetric subcompartment pressure loads are provided to Westinghouse by Sargent & Lundy. The analysis to determine these loads is discussed in Section 6.2.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

The support loads $[F]$ are computed by multiplying the support stiffness matrix $[K]$ and the displacement vector $[\delta]$ at the support point. The support loads are used in the evaluation of the supports.

The time-history displacements of the FIXFM subprogram are used as input to a WESTDYN2 to determine the internal forces, deflections, and stresses at each of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses. The results of this solution are used in the piping stress evaluation.

Transients

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into three parts, a uniform, a linear, and nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the nonlinear portion causes a skin stress.

The transients as defined in Subsection 3.9.1.1 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat conduction program is used to solve the thermal transient problem. The pipe is represented by at least 50 elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time-varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic, while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. An arbitrary temperature distribution across the wall is shown in Figure 3.9-2.

3.9.1.4.4 Primary Component Supports Models and Methods

Primary component supports are discussed in Subsection 3.9.3.4.

3.9.1.4.5 Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop include the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is Seismic Category I and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3.9-2. The equipment is analyzed for (1) the normal loads of deadweight, pressure, and thermal; (2) mechanical transients of OBE, SSE, and pipe ruptures; including the effects of asymmetric subcompartment pressurization and (3) pressure and temperature transients outlined in Subsection 3.9.1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. This is, on the basis of previous plant analyses, a set of loads is determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformation is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the umbrella load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectrum corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator and pressurizer are performed using 2% damping for the OBE and 4% damping for the SSE. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump internals is performed using the damping for bolted steel structures, that is 4% for the OBE and 7% for the SSE (2% for OBE and 4% for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump, since the main flange, motor stand, and motor are all bolted assemblies (see Section 5.4). The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

Reactor coolant pressure boundary components are further qualified to ensure against unstable crack growth under faulted conditions by performing detailed fracture analyses of the critical areas of this boundary. Actuation of the emergency core

total stresses in excess of the yield strength does not affect the conservatism of the results, provided that these thermal stresses are included in the evaluation of the stress intensity factors. Therefore, for faulted condition analyses, LEFM is considered applicable for the evaluation of the vessel inlet nozzle and beltline region.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw at the vessel inlet nozzle and beltline region.

For the safety injection and charging line nozzles, which are fabricated from 304 stainless steel, LEFM is not applicable because of extreme ductility of the material. For these nozzles, the thermal effects are evaluated using the principles of Miner's hypothesis of linear cumulative damage in conjunction with fatigue data from constant stress or strain fatigue tests. The cumulative usage fatigue defined as the sum of the ratios of the number of cycles of each transient (n) to the allowable number of cycles for the stress range associated with the transient (N) must not exceed 1.0.

An example of a faulted condition evaluation carried out according to the procedure discussed previously is given in Reference 3. This report discusses the evaluation procedure in detail as applied to a severe faulted condition (a postulated loss-of-coolant accident).

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III. These valves are identified in Subsection 3.9.3.2.

Valves in sample lines connected to the RCS are not considered to be Seismic Category I nor ASME Class 1. This is because the nozzles where the line connects to the primary system piping are orificed to a 3/8-inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

3.9.1.4.6 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss of Coolant Accident

The structural analysis of the reactor vessel and internals considers simultaneous application of the time-history loads resulting from the reactor coolant loop mechanical loads, internal hydraulic pressure transients, and reactor cavity pressurization. The vessel is restrained by four reactor vessel supports under every other reactor vessel nozzle and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

Pipe displacement restraints installed in the primary shield wall limit the break opening area of the vessel nozzle pipe breaks. An upper pump break area is determined from break areas calculated using reactor vessel and pipe relative motions for similar plant analyses. Detailed studies have shown that pipe breaks at the hot or cold leg reactor vessel nozzles, even with a limited break area, would give the highest reactor vessel support loads and the highest vessel displacements, primarily due to the influence of reactor cavity pressurization. By considering these breaks, the most severe reactor vessel support loads are determined.

3.9.1.4.6.1 Loading Conditions

Following a postulated pipe rupture at the reactor vessel nozzle, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of three phenomena: (1) reactor coolant loop mechanical loads, (2) reactor cavity pressurization forces, and (3) reactor internal hydraulic forces.

The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. The reactions on the nozzles of all the unbroken piping legs are applied to the vessel in the reactor pressure vessel blow-down analysis.

Reactor cavity pressurization forces arise for the pipe breaks at the vessel nozzles from the steam and water which is released into the reactor cavity through the annulus around the broken pipe. The reactor cavity is pressurized asymmetrically with higher pressure on the side of the broken pipe resulting in horizontal forces applied to the reactor vessel. Small vertical forces arising from pressure on the bottom of the vessel and the vessel flanges are also applied to the reactor vessel. The cavity pressure analysis is described in Section 6.2.

The internal reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For a vessel inlet nozzle break, the depressurization wave path is through the broken loop inlet nozzle and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, down, and around the downcomer annulus and up through the fuel. In the case of a reactor pressure vessel outlet nozzle break the wave passes through the reactor pressure vessel outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel. Thus, for an outlet nozzle break, the downcomer annulus is depressurized with much smaller differences in pressure horizontally across the core barrel than for the inlet break. For both the inlet and outlet nozzle breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomena has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam-bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8708. (6)

3.9.1.4.6.2 Reactor Vessel and Internals Modeling

The reactor vessel model consists of two nonlinear elastic models connected at a common node. One model represents the dynamic vertical characteristics of the vessel and its internals, and the other model represents the translational and rotational characteristics of the structure. These two models are combined in the DARI-WOSTAS code (Reference 1) to represent motion of the reactor vessel and its internals in the plane of the vessel centerline and the broken pipe centerline.

The model for horizontal motion is shown in Figure 3.9-23. Each node has one translational and one rotational degree of freedom in the vertical plane containing the centerline of the nozzle attached to the broken pipe and the centerline of the vessel. A combination of beam elements and concentrated masses are used to represent the components including the vessel, core barrel, neutron panels, fuel assemblies, and upper support columns. Connections between the various components are either pin-pin rigid links, translational impact springs with damping, or rotational springs.

The model for vertical motion is shown in Figure 3.9-24. Each mass node has one translational degree of freedom. The structure is represented by concentrated masses, springs, dampers, gaps, and frictional elements. The model includes the core barrel, lower support columns, bottom nozzles, fuel rods, top nozzles, upper support structure, and reactor vessel.

The horizontal and vertical models are coupled at the elevation of the primary nozzle centerlines. Node 1 of the horizontal model is coupled with node 2 of the vertical model at the reactor vessel nozzle elevation. This coupled node has external restraints characterized by linear horizontal springs which describe the tangential resistance of the supports and by individual nonlinear vertical stiffness elements which provide downward restraint only. The supports as represented in the horizontal and vertical models are not indicative of the complexity of the support system used in the analysis. The individual supports are located at the actual support pad locations and accurately represent the independent nonlinear behavior of each support.

3.9.1.4.6.3 Analytical Methods

The time-history effects of the cavity pressurization loads, internals loads, and loop mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time-history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of the vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated. In addition, using the results of the RCL analysis, the actual break opening area is verified to be less than the estimated area used in the analysis and assures that the analysis is conservative.

3.9.1.4.7 Stress Criteria for Class 1 Components

All Class 1 components are designed and analyzed for the design, normal, upset, and emergency conditions to the rules and requirements of the ASME Code Section III. The design analysis or test methods and associated stress or load allowable limits that will be used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code with supplementary options outlined below.

The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

The reactor vessel support pads are qualified using the test option. The reactor pressure vessel support pads are designed to restrain unidirectional horizontal motion in addition to supporting the vessel. The design of the supports allows radial growth of the vessel but restrains the vessel from horizontal displacements since tangential displacement of the vessel is prevented at each vessel nozzle.

To duplicate the loads that act on the pads during faulted conditions, the tests, which utilized a one-eighth linear scale model, were performed by applying a unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted by a support shoe which was mounted to the test fixture.

The above modeling and application of load thus allows the maximum load capacity of the support pads to be accurately established. The test load, L_T , was then determined by multiplying the maximum collapse load by 64 (ratio of prototype area to model area) and including temperature effects in accordance with the rules of the ASME Code, Section III.

The loads on the reactor vessel support pads, as calculated in the system analysis for faulted conditions are limited to the value of $0.80 L_T$. The tests performed and the limits established for the test load method ensure that the experimentally obtained value for L_T is accurate and that the support pad design is adequate for its intended function.

Loading combinations and allowable stresses for ASME Class 1 components are given in Tables 3.9-2 and 3.9-3, respectively.

The methods of load combination for each operating condition are as follows:

Design

Loads are combined by algebraic sum.

Normal, Upset

These loads are used in the fatigue evaluation in accordance with the methods prescribed in the ASME code. Loadsets are defined for each transient including the OBE and are combined such that the maximum stress ranges are obtained without regard to the order in which the transients occur. (This is discussed in more detail in Subsection 3.9.1.4.3).

Emergency

Loads are combined by algebraic sum.

Faulted

LOCA and SSE loads are combined using the square-root-of-the-sum-of-the-squares (SRSS) method on a load component basis (i.e., the LOCA F_x is combined with the SSE F_x by SRSS, the LOCA F_y is combined with the SSE F_y by SRSS, and likewise for F_z , M_x , M_y , and M_z). The sustained loads, such as weight effects, are combined with the SRSS results by algebraic sum.

3.9.1.4.8 Analytical Methods for RCS Class 1 Branch Lines

The analytical methods used to obtain the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectrum method for seismic dynamic analysis, and static or dynamic structural analysis for the effect of a reactor coolant loop pipe break.

The integrated Class 1 piping/supports system model is the basic system model used to compute loadings on components, component and piping supports, and piping. The system models include the stiffness and mass characteristics of the Class 1 piping components, the reactor coolant loop, and the stiffness of supports which affect the system response. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The Class 1 piping system models are constructed for the WESTDYN computer program, which numerically describes the physical system. A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points are determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads are also computed by multiplying the stiffness matrix by the displacement vector at the support point.

Seismic

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is accomplished by locating the total mass at points in the system which will appropriately represent the response of the distributed system. Effects of the primary equipment motion, that is, reactor vessel, steam generator, reactor coolant pump, and pressurizer, on the Class 1 piping system are obtained by modeling the mass and the stiffness characteristics of the primary equipment and loop piping in the overall system model.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Shock suppressors which resist rapid motions are also included in the analysis. The solution for the seismic disturbance employs the response spectra method. This method employs the lumped mass technique, linear elastic properties, and the principle of model superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response from differential anchor motions. The stresses resulting from the anchor motions are considered to be secondary and, therefore, are included in the fatigue evaluation.

Loss of Coolant Accident

The mathematical models used in the seismic analyses of the Class 1 lines are also used for RCL pipe break effect analysis. To obtain the dynamic solution for lines six inches and larger and certain small-bore lines required for ECCS considerations, the time-history deflections from the analysis of the reactor coolant loop are applied at branch nozzle connections. For other small bore lines which must maintain structural integrity, the motion of the RCL is applied statically.

Fatigue

A thermal transient heat transfer analysis is performed for each different piping component on all the Class 1 branch lines. The normal, upset, and test condition transients identified in Subsection 3.9.1.1 are considered in the fatigue evaluation.

The thermal quantities ΔT_1 , ΔT_2 , and $(\alpha_a T_a, -\alpha_b T_b)$ are calculated on a time-history basis, using a one-dimensional finite difference heat transfer computer program. Stresses due to these quantities were calculated for each time increment using the methods of NB-3650 of ASME III.

For each thermal transient, two loadsets are defined, representing the maximum and minimum stress states for that transient.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus insuring the most conservative combinations of seismic loads are used in the stress evaluation.

The WESTDYN computer program is used to calculate the primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors and cumulative usage factors for all possible loadset combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of loadsets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.1.4.9 Evaluation of Control Rod Drive Mechanisms and Supports

The control rod drive mechanisms (CRDM's) and CRDM support structure are evaluated for the loading combinations outlined in Table 3.9-3.

A detailed finite element model of the CRDM's and CRDM supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis, nonlinearities in the structure are represented. The time-history

motion of the reactor vessel head, obtained from the RPV analysis described in 3.9.1.4.6 is input to the dynamic model. Maximum forces and moments in the CRDM's and support structure are then determined. For the seismic analysis, the structural model is linearized and the floor response spectra corresponding to the CRDM tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the CRDM's for the various loading conditions are compared with maximum allowable moments determined from a detailed finite element stress evaluation of the CRDM's. Adequacy of the CRDM support structure is verified by comparing the calculated stresses to the criteria given in ASME III, Subsection NF.

NSSS (For ASME Code Class 2 and 3 Components)

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9.3.1.1 Design Loading CombinationsBalance of Plant

The combination of design loadings is categorized with respect to plant conditions identified as normal, upset, emergency, or faulted as shown in Tables 3.9-5 through 3.9-14 for the major components and piping.

NSSS

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9-5. The design loading combinations are categorized with respect to normal, upset, emergency, and faulted conditions. Stress limits for each of the loading combinations are presented in Tables 3.9-6, 3.9-7, 3.9-8, and 3.9-9 for tanks, inactive* pumps, active pumps, and valves, respectively. Active** pumps and valves are discussed in Subsection 3.9.3.2. Design of primary equipment supports is discussed in Subsection 3.9.3.4.

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- * Inactive components are those whose operability is not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.
 - ** Active components are those whose operability is relied upon to perform a safety function (as well as to accomplish and maintain a safe reactor shutdown) during and following the transients and events considered in the respective operating condition categories.

3.9.3.1.2 Design Stress Limits3.9.3.1.2.1 Stress Level for Class 1 Piping and Components (Balance of Plant)

Stress analysis was used to determine structural adequacy of pressure components under the operating conditions of normal, upset, emergency, or faulted, as applicable.

Significant discontinuities were considered such as nozzles, flanges, etc. In addition to the design calculation required by the ASME III code, stress analysis was performed by methods outlined in the code appendices or by other methods by reference to analogous codes or other published literature.

3.9.3.1.2.2 Stress Levels for ASME Code Class 2 and 3

Balance of Plant

For safety-related ASME Code Class 2 and 3 components and piping, the design stress limits are listed in Tables 3.9-5 through 3.9-14. Inelastic methods as permitted by ASME Section III for Class I components were not used for these components.

NSSS

The design stress limits established for Class 2 and 3 components are sufficiently low to assure that violation of the pressure-retaining boundary will not occur. These limits for each of the loading combinations are component oriented and are presented in Tables 3.9-6 through 3.9-9.

The criteria for Class 2 and 3 component supports are as follows:

a. Supports for Vessels Procured After July 1, 1974

Class 2 and 3 vessel supports are designed and analyzed to the rules and requirements of ASME III, Subsection NF.

For linear supports designed by analysis, the increased design limit for stress identified in NF-3231.1(a) shall be limited to the smaller of $2.0 S_y$ or $3 S_u$, unless otherwise justified by shakedown analysis. The methods for analysis and associated allowable limits that are used in the evaluation of linear supports for faulted conditions are those defined in ASME III Appendix F.

Plate and shell supports shall satisfy the following stress criteria for faulted conditions: $\sigma_1 < 2.0 S$, $\sigma_1 + \sigma_2 < 2.4 S$. (σ_1 and σ_2 are defined in NF-3221.1 of ASME III.)

b. Supports for Vessels Procured Prior to July 1, 1974

1. Linear

- a) Normal - The allowable stresses of A.I.S.C.-69, Part 1 are employed for normal condition allowables.
- b) Upset - Stress limits for upset conditions are 33% higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of A.I.S.C.-69, Part 1 which permits

one-third increase in allowable stresses for wind or seismic loads.

- c) Emergency - Not applicable.
- d) Faulted - Stress limits for faulted condition are the same as for the upset condition.

2. Plate and Shell

- a) Normal - Normal condition limits are those specified in ASME Section VIII, Division 1 or A.I.S.C.-69, Part 1.
- b) Upset - Stress limits for upset conditions are 33% higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of A.I.S.C, Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.
- c) Emergency - Not applicable.
- d) Faulted - Stress limits for faulted condition are the same as for the upset condition.

c. Supports for Pumps

The stress limits used for Class 2 and 3 pump supports are identical to those used for the supported component, as indicated in Tables 3.9-7 and 3.9-8.

3.9.3.1.2.3 Field Run Piping (Balance of Plant)

No Seismic Category I field run piping system exists. Category II piping, 2-inch nominal pipe size and smaller, and 200° F and colder, are field run. Criteria are provided to the contractor to ensure proper routing and design interface with Seismic Category I systems and equipment, or interfaces are appropriately controlled by guides.

3.9.3.2 Pump and Valve Operability Assurance

Balance of Plant

Design methods are a combination of analysis, past testing, and operating experience.

Active mechanical equipment classified as Seismic Category I has been shown capable of performing its function during the life of the plant under postulated plant conditions.

Equipment with operating condition functional requirements includes "active" (active equipment must perform a mechanical motion during the course of accomplishing a safety function) pumps and valves in fluid systems such as the residual heat removal system, safety fluid injection systems, and the essential service water system.

Operability will be ensured by satisfying the requirements of the following programs. Continued operability is ensured by periodic testing.

NSSS

Mechanical equipment classified as safety-related must be capable of performing its function under postulated plant conditions. Equipment with faulted condition functional requirements includes active pumps and valves in fluid systems such as the residual heat removal system, safety injection system, and the containment.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the rod cluster control assembly and the stationary gripper return spring is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the stationary gripper return spring and weight acting upon the latches. After the rod cluster control assembly is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9.4.2 Applicable CRDS Design Specifications

For those comparable in the control rod drive system comprising portions of the reactor coolant pressure boundary, conformance with General Design Criteria 15, 30, 31, 32 and 10 CFR 50: Section 50.55a is discussed in Section 5.2. Conformance with Regulatory Guides pertaining to materials suitability is described in Section 4.5 and Subsection 5.2.3.

Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

Design Stresses

The control rod drive system is designed to withstand stresses originating from various operating conditions as summarized in Table 3.9-1. Loading combinations for the Class 1 components of the control rod drive system are given in Table 3.9-2.

a. Allowable Stresses

For normal operating conditions Section III of the ASME Boiler and Pressure Code is used. All pressure boundary components are analyzed as Class I components under Article NB-3000.

b. Dynamic Analysis

The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the control rod drive system.

Control Rod Drive Mechanisms

The control rod drive mechanism (CRDM) pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and

In addition, dynamic testing programs have been conducted by Westinghouse and Westinghouse licensees to demonstrate that control rod scram time is not adversely affected by postulated seismic events. Acceptable scram performance is assured by also including the effects of the allowable displacements of the driveline components in the evaluation of the test results.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7.7-9 shows the Basic Flux-Mapping System).

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. These are the only conditions which affect the incore instrumentation support structure. Reactor vessel surveillance specimen capsules are covered in Subsection 5.3.1.6.

3.9.5.2 Design Loading Conditions

The design loading conditions that provide the basis for the design of the reactor internals are:

a. Normal and Upset

The normal and upset loading conditions that provide the basis for the design of the reactor internals are:

1. fuel and reactor internals weight.
2. fuel and core component spring forces including spring preloading forces.
3. differential pressure and coolant flow forces.
4. temperature gradients.
5. vibratory loads including OBE seismic.
6. the normal and upset operational thermal transients listed in Table 5.2-1.
7. control rod trip (equivalent static load).
8. loads due to loop(s) out-of-service.
9. loss of load pump overspeed.

b. Emergency Conditions

The emergency loading conditions that provide the basis for the design of the reactor internals are:

1. small loss of coolant accident.
2. small steam break
3. complete loss of flow

c. Faulted Conditions

The faulted loading conditions that provide the basis for the design of the reactor internals are:

1. the large loss of coolant accident
2. the safe shutdown earthquake.

The main objectives of the design analysis are to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue

characteristics of the materials. Both low and high cycle fatigue stresses are considered when the allowable amplitude of oscillation is established. Dynamic analyses on the reactor internals are provided in Subsection 3.9.2.

As part of the evaluation of design loading conditions, extensive testing and inspections are performed from the initial selection of raw materials up to and including component installation and plant operation. Among these tests and inspections are those performed during component fabrication, plant construction, startup and checkout, and during plant operation.

3.9.5.3 Design Loading Categories

The combination of design loadings fits into either the normal, upset, emergency or faulted conditions as defined in the ASME Code, Section III, and as indicated by Figures NG-3221.1, NG-3224.1 and by Appendix F, Rules for Evaluating Faulted Conditions.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3.9-1.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the loss-of-coolant accident plus the safe shutdown earthquake condition, the deflection criteria of critical internal structures are the limiting values given in Table 3.9-14. The corresponding no-loss-of-function limits are included in Table 3.9-4 for comparison purposes with the allowed criteria.

The criteria for the core drop accident is based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately 1/2 inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy-absorbing devices of the secondary core support; thus the total drop distance is about 1-1/4 inches, which is insufficient to permit the tips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the

must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.9-14. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited so as not to exceed the value shown in Table 3.9-4.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9.2.

The basis for the design stress and deflection criteria is identified below:

Allowable Stresses

For normal operating conditions, the intent of Section III of the ASME Nuclear Power Plant Components Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered. It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design-basis accident used for the core support structures are based on the intent of the draft ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

The stress criteria for the reactor internals that Westinghouse applied before the existence of Subsection NG of the ASME code are composed of two parts, and depend upon the nature of the stress state membrane or bending. A direct or membrane stress has a uniform stress distribution over the cross section. The allowable (maximum) membrane or direct stress is taken to be equal to the stress corresponding to 20% of the uniform material strain or the yield strength whichever is higher. For unirradiated Type 304 stainless steel at operating temperature, the stress corresponding to 20% of the uniform strain is 39,500 psi.

For a bending state of stress, the strain is linearly distributed over a cross section. The average strain value is, therefore, one-half of the outer fiber strain where the stress is maximum. Thus, by requiring the average bending stress to satisfy the allowable criteria for the direct state of stress, the average absolute strain may be 20% of the uniform strain. Consequently, the outer fiber strain may be 40% of the uniform strain. The

maximum allowable outer fiber bending stress is then taken to be equal to the stress corresponding to 40% of the uniform strain or the yield strength, whichever is higher. For unirradiated Type 304 stainless steel at operating temperature, the stress is 50,000 psi.

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of pumps and valves will be done in accordance with a plan approved per 10 CFR 50.55g.

3.9.6.1 Inservice Testing of Pumps

All ASME Code Class 1, 2, and 3 pumps requiring inservice testing are listed in Table 3.9-11. Section III pumps not listed are those excluded by the scope of IWP-1000. The pump test plan and schedule is included in the technical specifications, Chapter 16.0.

3.9.6.2 Inservice Testing of Valves

ASME Code Class 1, 2, and 3 valves requiring inservice testing are listed in Table 3.9-12 and identified as to valve category as defined by IWV-2110 of Section XI. Section III valves not listed are those excluded by the scope of IWV-1000. The valve test plan

3.9.7 References

1. WCAP-8252, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," Revision 1, April 1977.
2. "Sample Analysis of a Class 1 Nuclear Piping System," prepared by ASME Working Group on Piping, ASME Publication, 1972.
3. W. H. Bamford and C. B. Buchalet, "Methods for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients," WCAP-8510, June 1976.
4. WCAP-8317-A, "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," July, 1975.
5. C. N. Bloyd, W. Ciarametaro, and N. R. Singleton, "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Prooperational Tests on the Trojan 1 Power Plant," WCAP-8780, May 1976.
6. K Takeuchi, et al., "Multiflex-A Fortran-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708, February 1976.
7. G. J. Bohm and J. P. La Faille, "Reactor Internals Response Under a Blowdown Accident," First Intl. Conf. on Structural Mechanics in Reactor Technology, Berlin, September 20-24, 1971.
8. F. W. Cooper, Jr., "17 x 17 Drive Line Components Tests - Phase 1B, II, III, D-Loop - Drop and Deflection," WCAP-8446 Proprietary and WCAP-8449, December 1974.
9. S. Kraus, "Neutron Shielding Pads," WCAP-7870, May 1972.
10. "Benchmark Problem Solutions Employed for Verification of WECAN Computer Program," WCAP-8929, June 1977.

TABLE 3.9-3

ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS (Note 1)

<u>OPERATING CONDITION CLASSIFICATION</u>	<u>VESSELS/ TANKS</u>	<u>PIPING</u>	<u>PUMPS</u>	<u>VALVES</u>
Design	NB-3221 (Design)	NB-3652 (Design)	NB-3221 (Design)	NB-3520 (Design)
Normal	NB-3222 (Level A)	NB-3653 (Level A)	NB-3222 (Level A)	NB-3525 (Level A)
Upset	NB-3223 (Level B)	NB-3654 (Level B)	NB-3223 (Level B)	NB-3525 (Level B)
Emergency	NB-3224 (Level C)	NB-3655 (Level C)	NB-3224 (Level C)	NB-3526 (Level C)
Faulted	NB-3225 (Level D)	NB-3656 (Level D)	NB-3225 (Level D)	Note 2

NOTES:

- Limits identified refer to subsections of the ASME Code, Section III.

TABLE 3.9-3 (Cont'd)

NOTE 2:CLASS 1 VALVE FAULTED CONDITION CRITERIA

<u>ACTIVE</u>	<u>INACTIVE</u>
<p>a) Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.25PS $P_m \leq 1.5S_m$</p> <p>b) Calculate Sn from para. NB3545.2 with Cp = 1.5 Ps = 1.25Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of 3545.2(b) (1) $S_n \leq 3S_m$</p>	<p>a) Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.50Ps $P_m \leq 2.4S_m$ or $0.7S_u$</p> <p>b) Calculate Sn from para. NB3545.2 with Cp = 1.5 Ps = 1.50Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of NB3545.2(b) (1) $S_n \leq 3S_m$</p>

TABLE 3.9-5

DESIGN LOADING COMBINATIONS* FOR ASME CODE CLASS 2 AND 3
COMPONENTS AND SUPPORTS

<u>CONDITION CLASSIFICATION</u>	<u>LOADING COMBINATION</u>
Design and Normal	Design pressure, Design temperature**, Dead weight, Nozzle loads***
Upset	Upset condition pressure, Upset condition metal temperature**, Deadweight, OBE, Nozzle loads***
Emergency	Emergency condition pressure, Emergency condition metal temperature**, Deadweight, Nozzle loads***
Faulted	Faulted condition pressure, Faulted condition metal temperature**, Deadweight, SSE, Nozzle loads***

*The responses for each loading combination are combined using the absolute sum method. On a case-by-case basis, algebraic summation may be used when signs are known for final design evaluations.

**Temperature is used to determine allowable stress only.

***Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

TABLE 3.9-6

STRESS CRITERIA FOR SAFETY-RELATED ASME CLASS 2 AND CLASS 3 VESSELS

<u>CONDITION</u>	<u>STRESS LIMITS*</u>
Design and Normal	The vessel shall conform to the requirements of ASME Section III, NC-3300 (or ND-3300)
Upset	$\sigma_m \leq 1.1 S$ $(\sigma_m \text{ or } \sigma_L) +$ $\sigma_b \leq 1.65 S$
Emergency	$\sigma_m \leq 1.5 S$ $(\sigma_m \text{ or } \sigma_L) +$ $\sigma_b \leq 1.80 S$
Faulted	$\sigma_m \leq 2.0 S$ $(\sigma_m \text{ or } \sigma_L) +$ $\sigma_m \leq 2.4 S$

*Stress limits are taken from ASME III, Subsections NC and ND, or, for vessels procured prior to the incorporation of these limits into ASME III, from Code Case 1607.

TABLE 3.9-7

STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3
INACTIVE PUMPS AND PUMP SUPPORTS

<u>CONDITION</u>	<u>STRESS LIMITS*</u>	<u>P_{max}**</u>
Design and Normal	The pump shall conform to the requirements of ASME Section III, NC-3400 (or ND-3400)	
Upset	$\sigma_m \leq 1.1 S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 S$	1.1
Emergency	$\sigma_m \leq 1.5 S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80 S$	1.2
Faulted	$\sigma_m \leq 2.0 S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4 S$	1.5

*Stress limits are taken from ASME III, Subsections NC and ND, or, for pumps procured prior to the incorporation of these limits into ASME III, from Code Case 1636.

**The maximum pressure shall not exceed the tabulated factors listed under P_{max} times the design pressure.

TABLE 3.9-8

DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS

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

*The stress limits specified for active pumps are more restrictive than the ASME III limits to provide assurance that operability will not be impaired for any operating condition.

TABLE 3.9-9

STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2
AND CLASS 3 ACTIVE AND INACTIVE VALVES

<u>CONDITION</u>	<u>STRESS LIMITS (NOTES 1-4, 6)</u>	<u>P_{max} (Note 5)</u>
Design and Normal	Valve bodies shall conform to ASME Section III.	
Upset	$\sigma_m \leq 1.1 S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65 S$	1.1
Emergency	$\sigma_m \leq 1.5 S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80 S$	1.2
Faulted	$\sigma_m \leq 2.0 S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4 S$	1.5

NOTES:

- Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied: (1) the section modulus and area of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110% of those for the piping connected (or joined) to the valve body inlet and outlet nozzles; and, (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above shall be multiplied by the ratio of $S_{\text{pipe}}/S_{\text{valve}}$. If unable to comply with this requirement, the design by analysis procedure of NB3545.2 is an acceptable alternate method.
- Casting quality factor of 1.0 shall be used.
- These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

TABLE 3.9-9 (Cont'd)

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.
5. The maximum pressure resulting from upset, emergency, or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.
6. Stress limits are taken from ASME III, Subsections NC and ND, or, for valves procured prior to the incorporation of these limits into ASME III, from Code Case 1635.

to an indicator on the main control board (nominal accuracy will be $\pm 5\%$) and on the Remote Shutdown Panel, and to a flow switch which energizes a high flow alarm on the main control board.

Transfer switches (REMOTE, LOCAL) are provided on the Remote Shutdown Panel. The auxiliary feedwater flow to the four steam generators is normally controlled from eight manual control stations mounted on the main control board if the transfer switch is in remote. Each manual control station electrically transmits a flow signal to an electric-to-pneumatic (E/P) converter. The pneumatic output flow signal is transmitted through a permissive three-way solenoid valve (which is deenergized for normal control) to the level control valves.

An equipment status display (ESD) "alarm" will be actuated if the manual control station for any of the eight level (flow) control valves is set below 160 gpm while the associated auxiliary feedwater pump drive is not operating.

Going to local control transfers the pneumatic control at the level control valves from the manual station on the main control board to a local controller mounted on the Remote Shutdown Panel, and energizes a "Valve on Local Control" alarm at the Manual Control Board.

The pneumatic control of the level control valves from the Remote Shutdown Panel is identical with the control from the main control room.

A failure in the control system for the control valve will cause the valve to fail open. The failure analysis is provided in Table 10.4-3.

Flow restricting devices are provided upstream of each flow control valve (Figure 10.4-2) in order to limit flow in the unlikely event of a pipe break.

f. Diverse Sources of Energy

Diverse sources of energy are provided for the auxiliary feedwater pumps, valve operators, instrumentation, and controls as discussed below.

Electrical equipment and cabling for both auxiliary feedwater pumps is physically separated into separate ESF trains.

One auxiliary feedwater pump per unit is motor driven. Its source of energy is 4-kV ESF bus 141(241). The other auxiliary feedwater pump for each unit is direct diesel driven. Fuel oil is supplied from its own Category I day tank; all necessary electrical auxiliaries for the diesel-driven auxiliary feedwater pump are powered from its own battery system. The auxiliary feedwater pump diesel, its auxiliaries and

9.2.2.2.2.3 Component Cooling Surge Tanks

There are two surge tanks provided. Each tank serves one loop of the systems under normal operating conditions. The tanks are connected to the pump suction lines. The tanks purpose is to: (1) accommodate system water expansion and contraction due to temperature changes; (2) accommodate inleakage to the system; (3) provide makeup for small system leaks until they can be isolated; and (4) as a point of chemical addition to the system. Design of the tanks is covered in Table 9.2-3.

9.2.2.2.2.4 Component Cooling Heat Exchangers

Three heat exchangers serve the system. Each heat exchanger is sized for 100% capacity of normal single unit heat loads. One heat exchanger serves each loop with the third available as a maintenance spare or for additional heat requirements of a particular loop. Design of the heat exchangers is covered in Table 9.2-3.

9.2.2.2.2.5 Component Cooling Pumps

There are five pumps serving the system. Under normal conditions, up to two pumps will serve each loop with the fifth pump available as a maintenance spare or for additional load requirements of a particular loop. Design of the pumps is covered in Table 9.2-3.

9.2.2.2.2.6 Component Cooling Instrumentation

The operation of the loop is monitored with the following instrumentation:

- a. a temperature detector in the component cooling pump suction line;
- b. temperature detectors in the outlet lines for the component cooling heat exchangers;
- c. pressure detectors on the lines between the component cooling pumps and the component cooling heat exchangers;
- d. a temperature and return-flow indicator in the pump suction header from the heat exchangers;
- e. redundant safety-related flow indicators on the reactor coolant pump motor and shaft seal cooling water discharge line;
- f. water-level indicators on the component cooling surge tank;

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

- a. letdown heat exchanger CCWS flow controllers,
- b. reactor coolant pump thermal barrier outlet flow controller, and
- c. component cooling surge tank radiation control valve.

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

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

Each component cooling surge tank vent has an air operated valve which will close on a high radiation signal from the radiation monitors in the discharge headers from the CCWS heat exchangers. This high radiation alarm normally indicates a primary to CCWS leak. Three radiation monitors are provided, any of which will alarm and close the vent valve on both surge tanks.

9.2.2.4.5 Electrical Power Supply

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

9.2.2.5 Tests and Inspections

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

10.4.6 Condensate Cleanup System

10.4.6.1 Design Bases

The condensate cleanup systems at Byron and Braidwood will be utilized primarily during plant startup to flush the condensate, condensate booster, and feedwater systems. This system will not be operated continuously. The equipment is designed to treat one-third of the condensate system flowrate supplied from the discharge header of the condensate pumps. The treated water returns to the condensate booster pumps suction header.

The condensate cleanup system is designed to produce an effluent at the design flowrate within the following limits:

- a. Sodium < 1 ppb
- b. Conductivity < 0.1 $\mu\text{mho/cm}$
- c. SO_4 < 1 ppb
- d. Iron < 10 ppb

All pressurized vessels in the system are designed and constructed in accordance with the ASME code for Unfired Pressure Vessels of ASME Division 1, Section VIII.

No part of the system is safety-related, thus it is designated Safety Category II.

10.4.6.2 System Description

10.4.6.2.1 General Description and System Operation

The condensate cleanup system for each station consists of four mixed bed polishers each designed for a capacity of 3750 gpm. Two vessels are normally assigned to each unit, however, the valving arrangement permits operation of the vessels with either unit. Normally the flowrate from each unit is equally divided among two vessels.

The external resin regeneration system, common to all four mixed bed polishers, consists of one resin mixing and storage tank, one resin separation and cation regeneration tank, and one anion regeneration tank. Resin is sluiced from a mixed bed polisher to the resin separation and cation regeneration tank. The anion and cation resin are separated and the anion resin is transferred to the anion regeneration tank. The cation resin is regenerated with sulfuric acid, and the anion resin is regenerated with sodium hydroxide. After regeneration is complete, the resins are transferred to the resin mixing and storage tank.

When placed in service, the operation of this system is controlled and maintained by a solid state controller. The control system will prevent the initiation of any automatic or sequence of operations that would conflict with any operation already in progress, whether such operation is under automatic or manual control. The operation status of each polisher and each regeneration vessel, including which automatic sequence is in progress, is indicated by means of lights on the polisher control panel.

Improper operation of the regeneration system and components will cause an alarm to sound and the system will be shut down. Improper regeneration solution strength will sound an alarm and the system will shut down if the situation is not corrected within five minutes.

10.4.6.2.2 Component Description

10.4.6.2.2.1 Mixed Bed Polisher

Each of the four mixed bed polishers are 114 inches in diameter with a 60-inch side seam and are sized for a flowrate of 3750 gpm. Each vessel contains approximately 5 kg/ft³ of anion resin and 5kg/ft³ of cation resin. The vessels are equipped with viewports on the side shell and an illumination port in the upper head. The mixed bed polishers are designed to Section VIII of the ASME Boiler and Pressure Vessel Code, and are rated at 300 psig. A high pressure resin trap in each polisher effluent line is designed to retain particles larger than 50 mesh.

10.4.6.2.2.2 Resin Separation and Cation Regeneration Tank

The resin separation and cation regeneration tank is 84 inches in diameter with a 174-inch side shell and equipped with four viewports in the side shell and an illumination port in the top head. The design pressure of the tank is 100 psig. The resin is backwashed to separate the anion and cation resins. The anion resin is drawn off before the cation resin is regenerated.

A 3-foot diameter by 5-foot side shell resin hopper is located above the resin separation and cation storage tank to make up for any lost resin.

10.4.6.2.2.3 Anion Regeneration Tank

Anion resin is transferred to this tank to be regenerated with caustic. The anion regeneration tank is 78 inches in diameter with a 120-inch side shell. The tank is equipped with two view ports in the side shell and one illumination port in the top head. The design pressure is 100 psig.

10.4.6.2.2.4 Resin Mix and Storage Tank

The resin mix and storage tank is 96 inches in diameter with a 102-inch side seam and the design pressure is 100 psig. Three viewports are located in the side shell and one illumination port is located in the top head. The tank is sized to contain a complete change of resin for one mixed bed polisher. The anion and cation resin is sluiced from their respective regeneration tanks to this storage tank. The resins are mixed and stored until being transferred to a mixed bed polisher.

10.4.6.2.2.5 Regeneration Equipment

The acid regeneration skid consists of a 200-gallon acid storage tank, two metering pumps, and a dilution station. The storage tank is sized for two regenerations. The caustic regeneration skid consists of a 700-gallon caustic tank, two metering pumps, and a dilution station. A hot water tank provides dilution water for regeneration of the anion resin. Both regeneration systems are equipped with the necessary instrumentation and controls to automatically provide regeneration chemicals in the required amount, temperature, and concentration to the respective regeneration tanks.

10.4.6.2.2.6 Sluice Water Pumps

Two 400-gpm, 100-psig pumps are used to supply water from the condensate storage tank for sluicing the resin between the various tanks. The pumps also supply the required dilution water to the acid and caustic regeneration systems.

10.4.6.3 Safety Evaluation

The condensate cleanup system is a non-safety-related system and is not required for safe shutdown of the plant.

10.4.6.4 Testing and Inspection

All pressurized tanks are designed in accordance with the ASME Code for Unfired Pressure Vessels of ASME Division 1, Section VIII. All equipment is factory inspected and tested in accordance with the applicable equipment specifications and codes. Preoperational tests will be performed on this system. The equipment manufacturer's recommendations and station practices are considered in determining required maintenance.

10.4.7 Condensate and Feedwater System

The purpose of the Condensate and Feedwater System is to provide feedwater from the condenser to the steam generators. This subsection discusses the Condensate and Feedwater System from the condenser to the connection with the steam generators.

10.4.7.1 Design Bases

10.4.7.1.1 Safety Design Bases

The only part of the Condensate and Feedwater System classified as safety-related (i.e., required for safe shutdown or in the event of postulated accidents) is the main feedwater piping from the preheater section of the steam generators to, and including, the outermost containment isolation and check valves; the tempering feedwater lines between the steam generator preheater bypass connections and the outermost check and isolation valves; the interconnecting piping between the tempering lines and the auxiliary feedwater system, and the chemical feed piping from the interface into the tempering piping to, and including, the shutoff valves. These parts of the system are designated as Safety Category I, Quality Group B.

10.4.7.2 System Description

The Condensate and Feedwater System consists of the piping, valves, pumps, heat exchangers, controls, instrumentation, and the associated equipment and subsystems that supply the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The system is shown in Figure 10.4-1, Sheets 1 and 2.

There are four 1/3-capacity centrifugal condensate pumps per unit with motor drives and common suction and common discharge headers, and four 1/3-capacity condensate booster pumps per unit with common suction and discharge headers. Each condensate and condensate booster pump set is driven by a single motor. Three sets of pumps are normally in operation. The fourth set of pumps will automatically start on low pressure at the feedwater pump suction to assure adequate flow to the feedwater pumps.

The Feedwater System is of the closed type, with deaerating accomplished in the condenser. The condensate pumps take suction from the condenser hotwell and pump condensate through the air ejector condensers and the gland steam condensers to the suction of the condensate booster pumps. These pump the condensate through six stages of low-pressure feedwater heating to the feedwater pumps. The water discharge from the feedwater pumps

UNRESOLVED GENERIC SAFETY ISSUES

A-12 Fracture Toughness of Steam Generator and Reactor
Coolant Pump Supports

During the course of the licensing action for North Anna Power Station Unit No. 1 and 2, a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was therefore necessary to reassess the fracture toughness of the steam generator and the reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

Fracture toughness and lamellar testing has been adequately addressed for the Byron/Braidwood NSSS supports. The materials used in the Byron/Braidwood supports were, as a minimum charpy V-notch tested in accordance with NF-2300. In general, items subject to through thickness stresses were ultrasonically tested to preclude lamellar tearing. Furthermore, the Byron/Braidwood NSSS supports employ the same materials and more stringent fracture toughness requirements than Zion Units 1 and 2. The staff, has stated in its Safety Evaluation Report "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports" transmitted January 1981, that "...the steam generator and reactor coolant pump supports at Zion Units 1 and 2 possess adequate fracture toughness to merit a Group III rating per NUREG-0577. Therefore, we consider the fracture toughness of the steam generator and reactor coolant pump supports acceptable."

Thus, CECO has concluded that Byron/Braidwood Units 1 and 2 can be operated until there is an ultimate resolution of this generic issue without undue risk to the health and safety of the public.

QUESTION 130.43

"The descriptive information of Category I structures other than containment is not in accordance with the provisions of the R. G. 1.70. Provide sufficient information, illustrated by sketches in the FSAR Section 3.8.4 to enable the staff to perform a meaningful review. Referencing the FSAR Section 1.2 which illustrates the general layout of the plant is insufficient in detail for a structural review."

RESPONSE

A plan view of auxiliary-fuel handling building shear walls, elevation views, shear wall-slab diaphragm connection (above grade and below grade), and a typical wall corner are shown in new Figures 3.8-52 through 3.8-58.

Byron river screen house foundation plans, floor and roof framing plans, and elevation views are shown in new Figures 3.8-59 through 3.8-64.

A Byron ESW cooling tower foundation plan, an air inlet plan, a fill support beam plan, a distribution support beam plan, a roof framing plan, and tower sections are shown in new Figures 3.8-65 through 3.8-73.

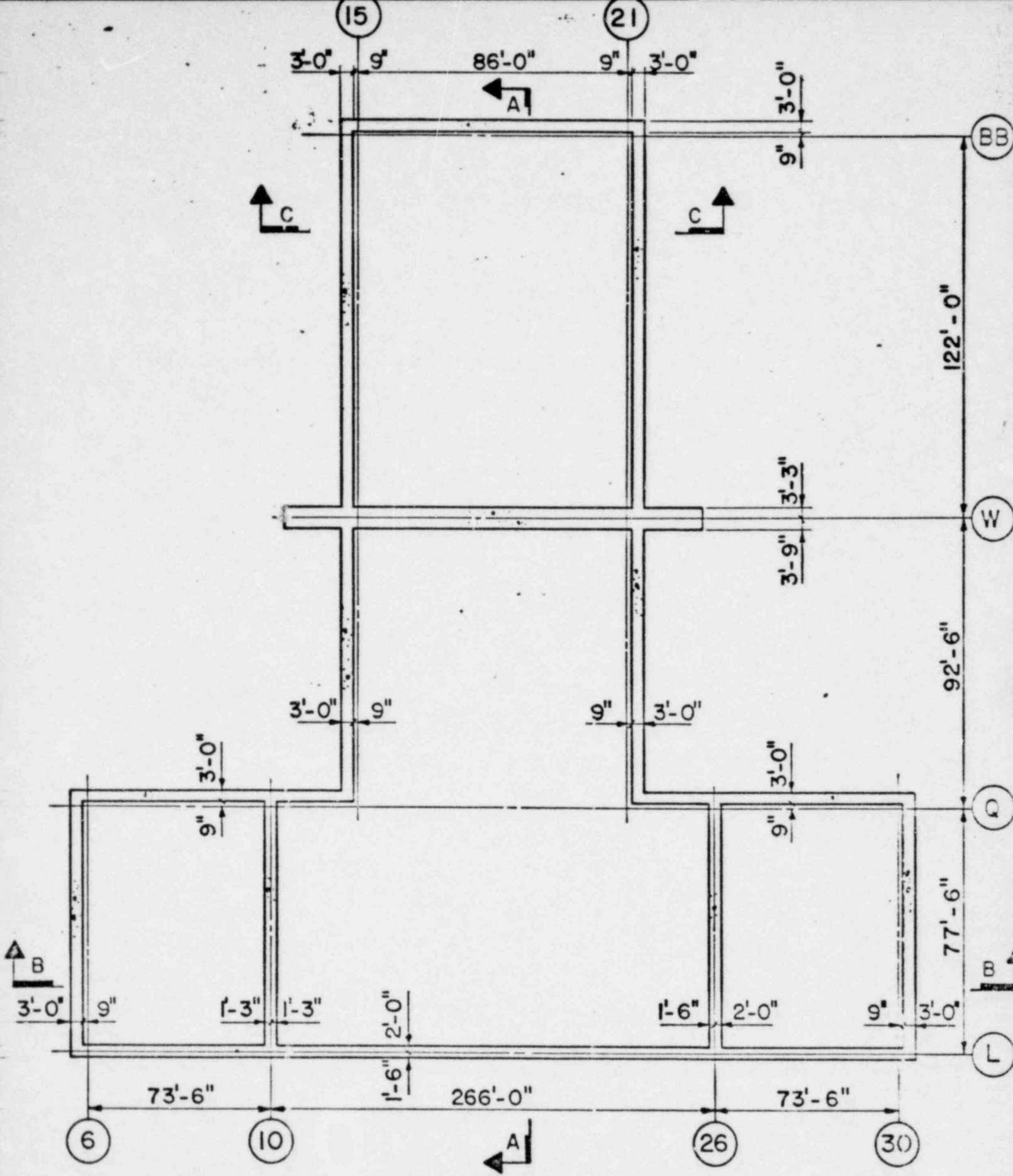
A Braidwood lake screen house foundation plan, floor framing plans, and elevation views are shown in new Figures 3.8-74 through 3.8-78.

A plan view and section of a typical Byron deep well enclosure are shown in new Figure 3.8-79.

A safety valve room plan view and elevation views are shown in new Figures 3.8-80 through 3.8-82.

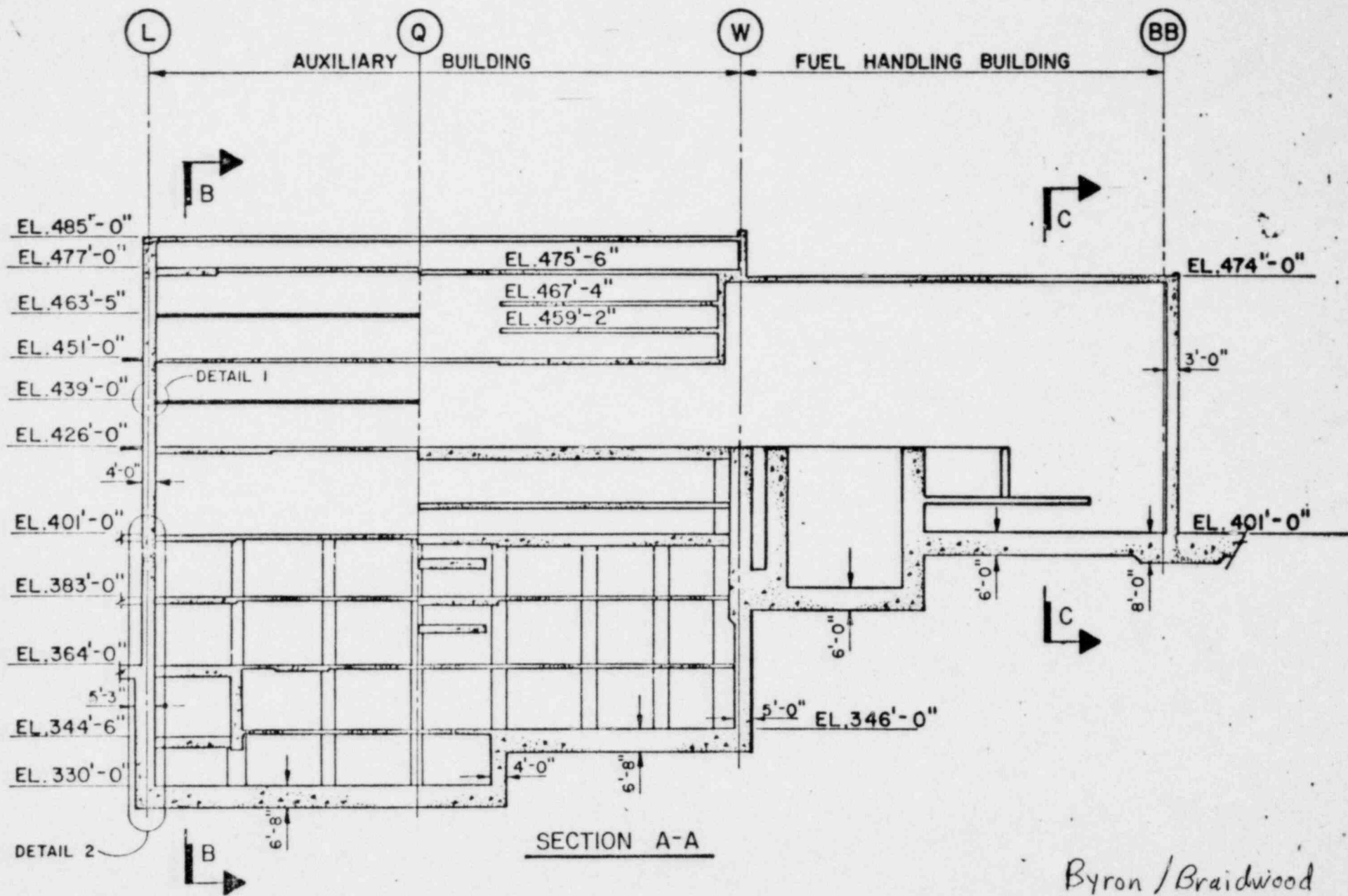
An elevation view and typical dome section of the refueling water storage tank are shown in new Figures 3.8-83 and 3.8-84, respectively.

These figures, along with the information in Subsection 3.8.4, provide sufficient detail for a structural review.

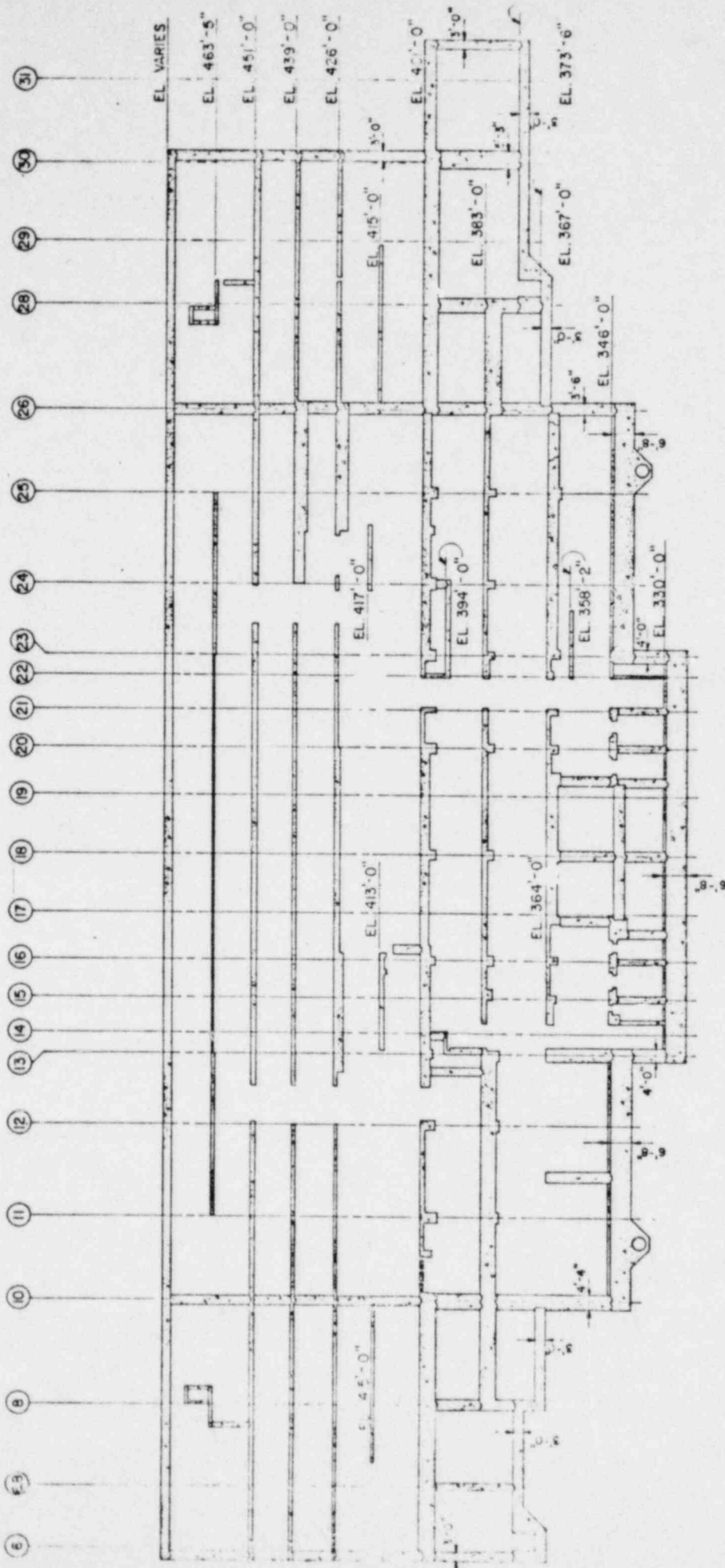


AUXILIARY - FUEL HANDLING BUILDING COMPLEX SHEAR WALLS AT ELEVATION 401'-0"

BYRON →
 BRAIDWOOD →
 N ←



Byron / Braidwood
 Figure 3.8-53
 Auxiliary - Fuel
 Handling Building Section

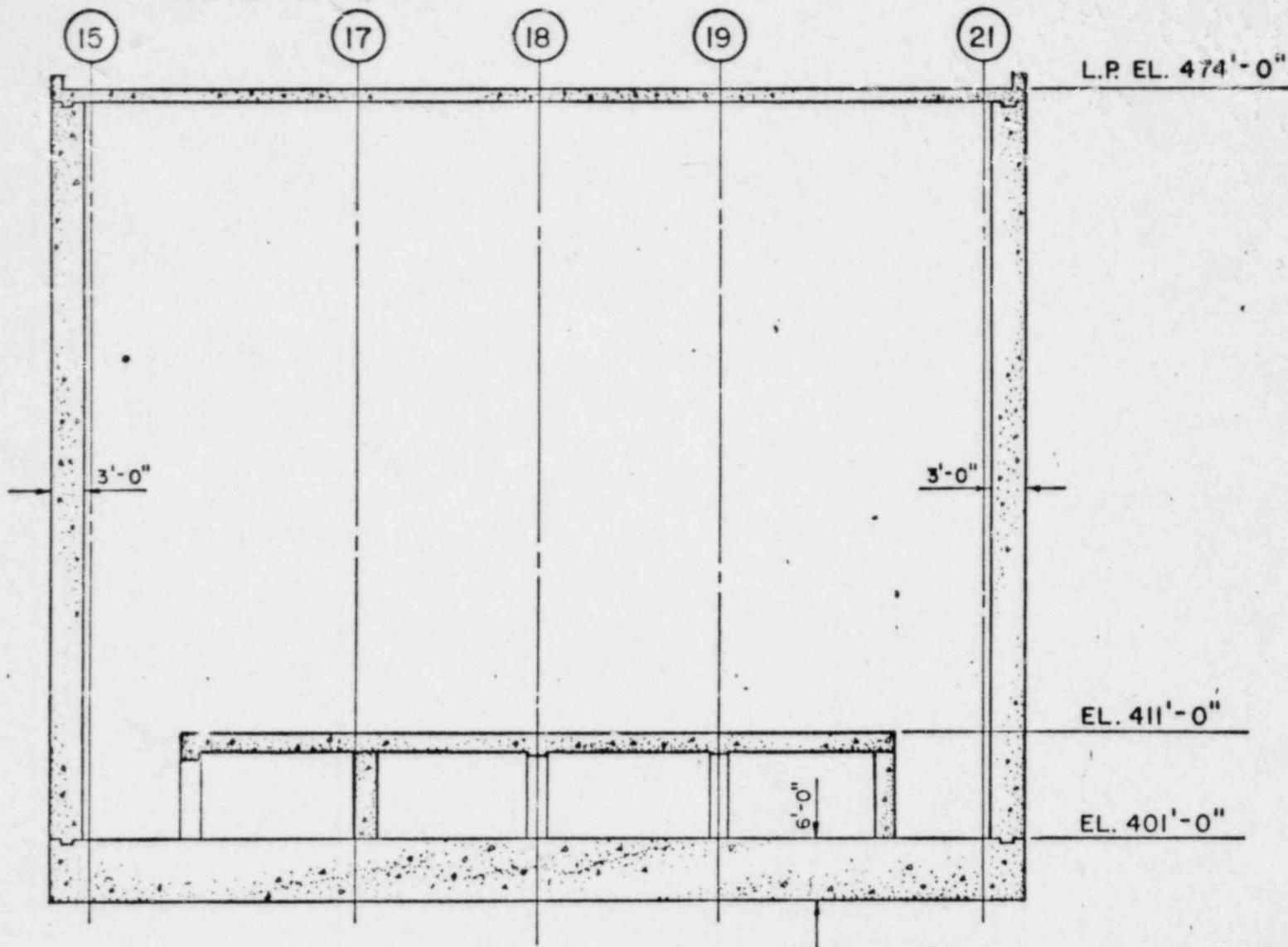


AUXILIARY BUILDING
SECTION B-B

Byron/Braidwood

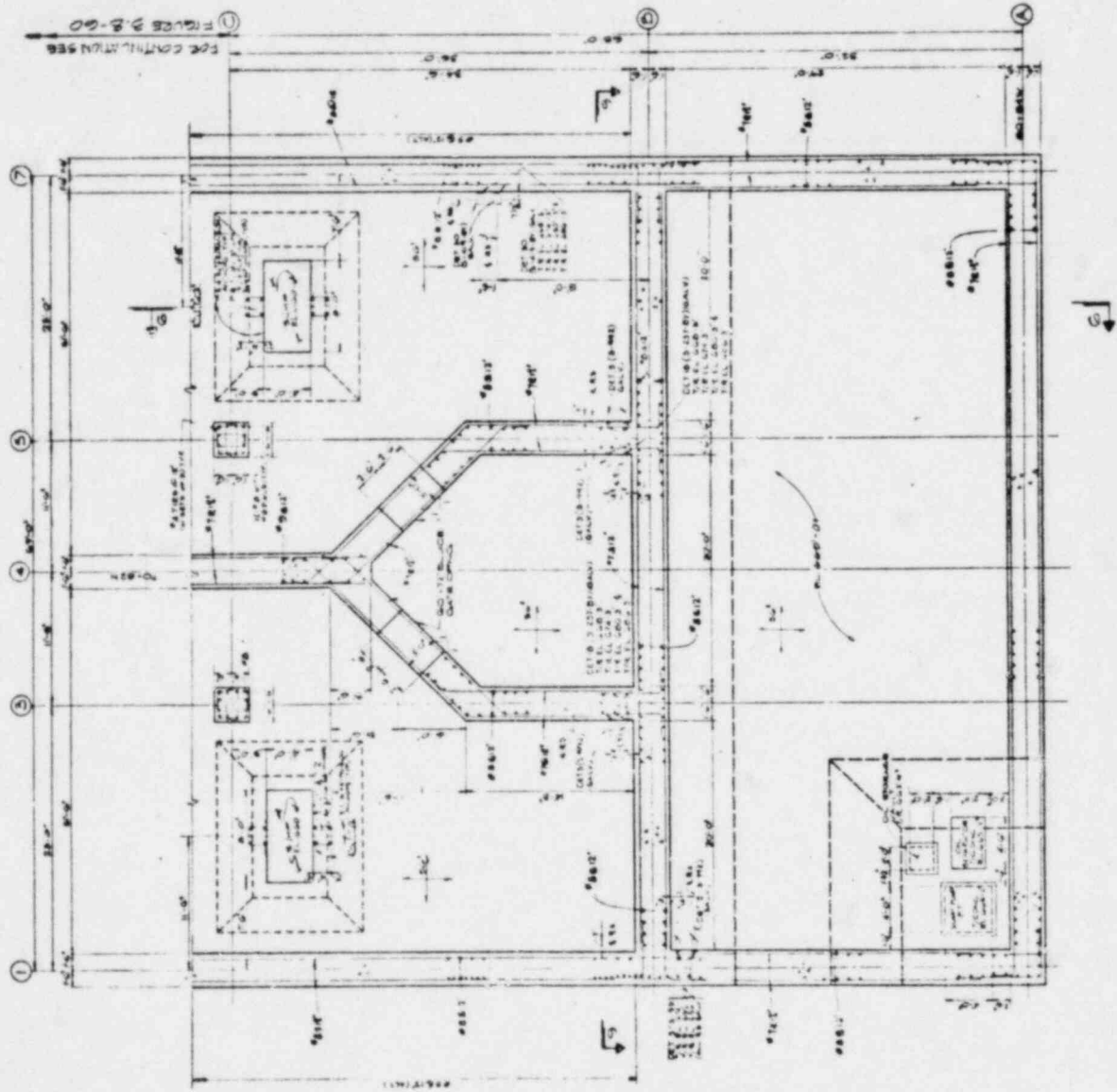
FIGURE 3.8-57

Auxiliary Building
Section



FUEL HANDLING BUILDING
SECTION C-C

Byron/Braidwood
 FIGURE 3.8-58
 Fuel Handling Building Section



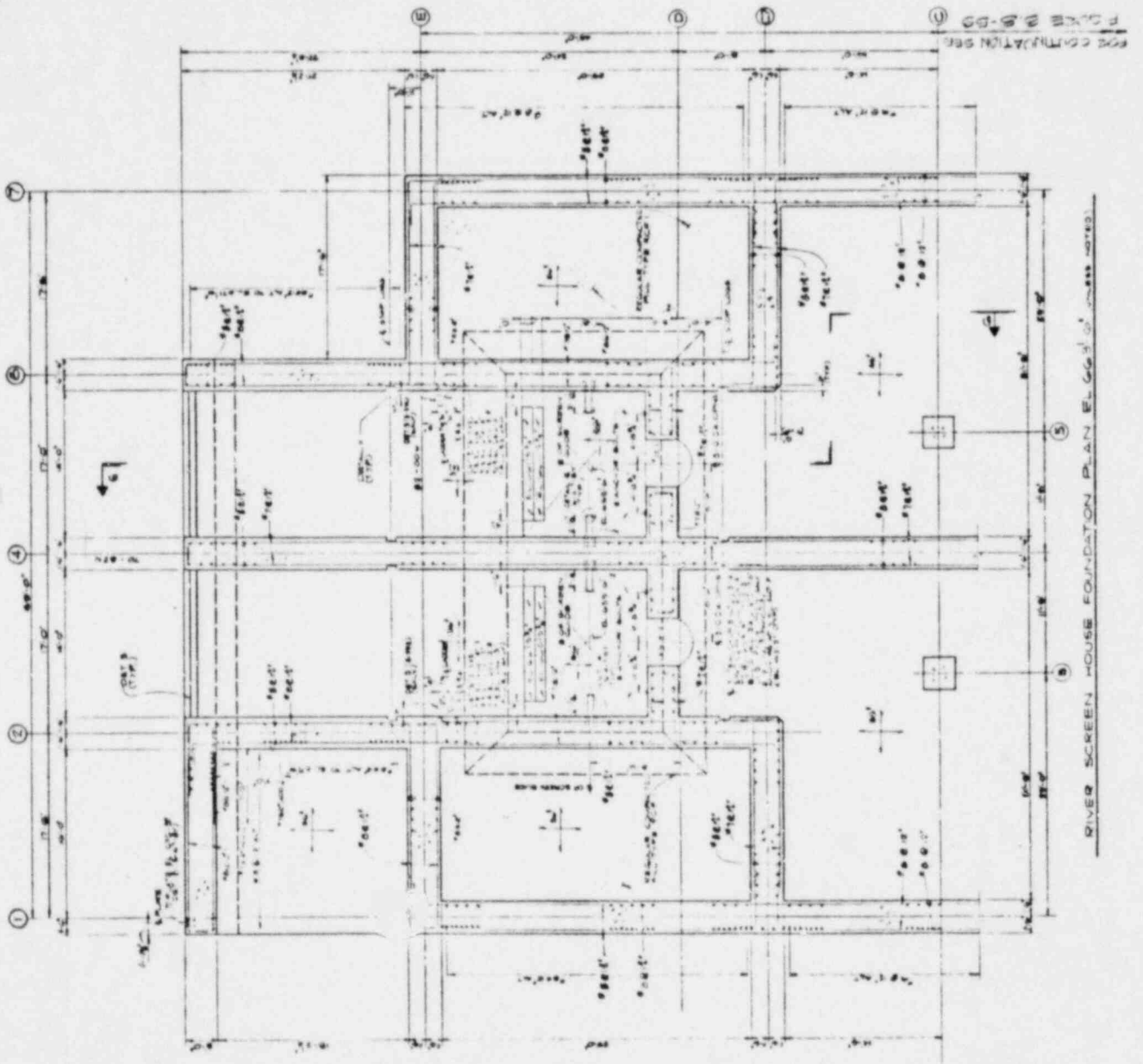
RIVER SCREEN HOUSE FOUNDATION PLAN (L.S. 603) (L.S. 603A) (L.S. 603B)

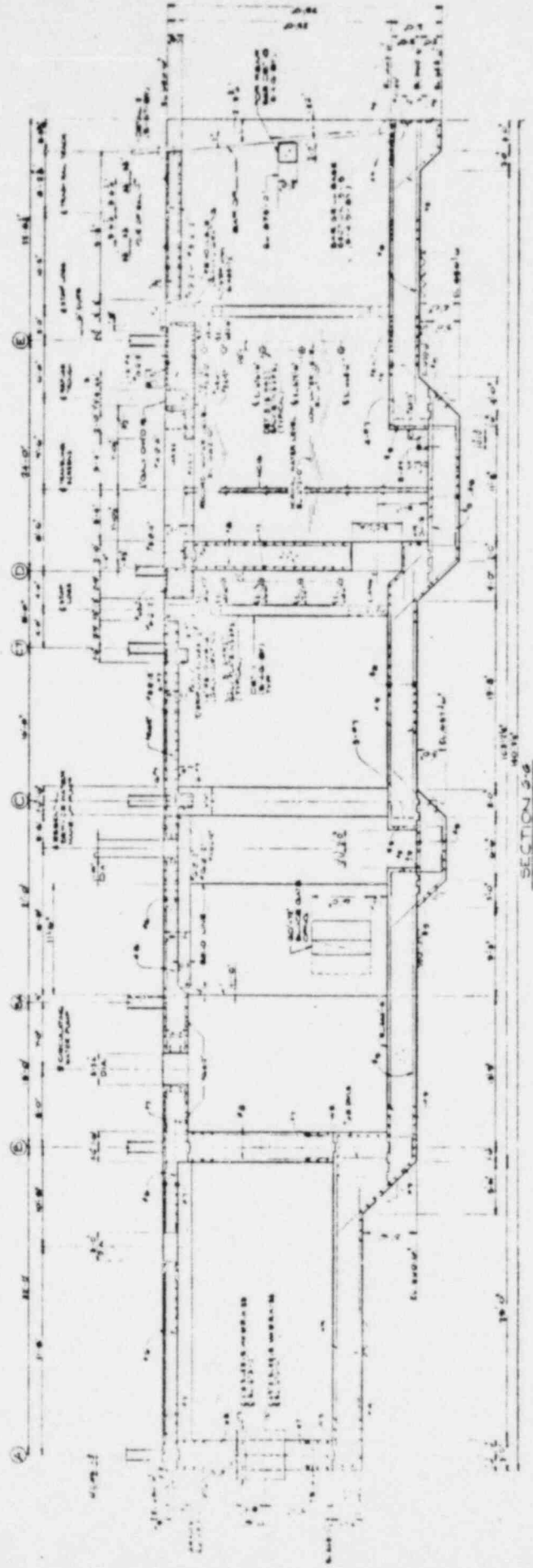
Byron
 FIGURE D.B-59
 River Screen House
 Foundation Plan

Byron

FIGURE B B-22

River Screen House
Foundation Plan (Cont'd)



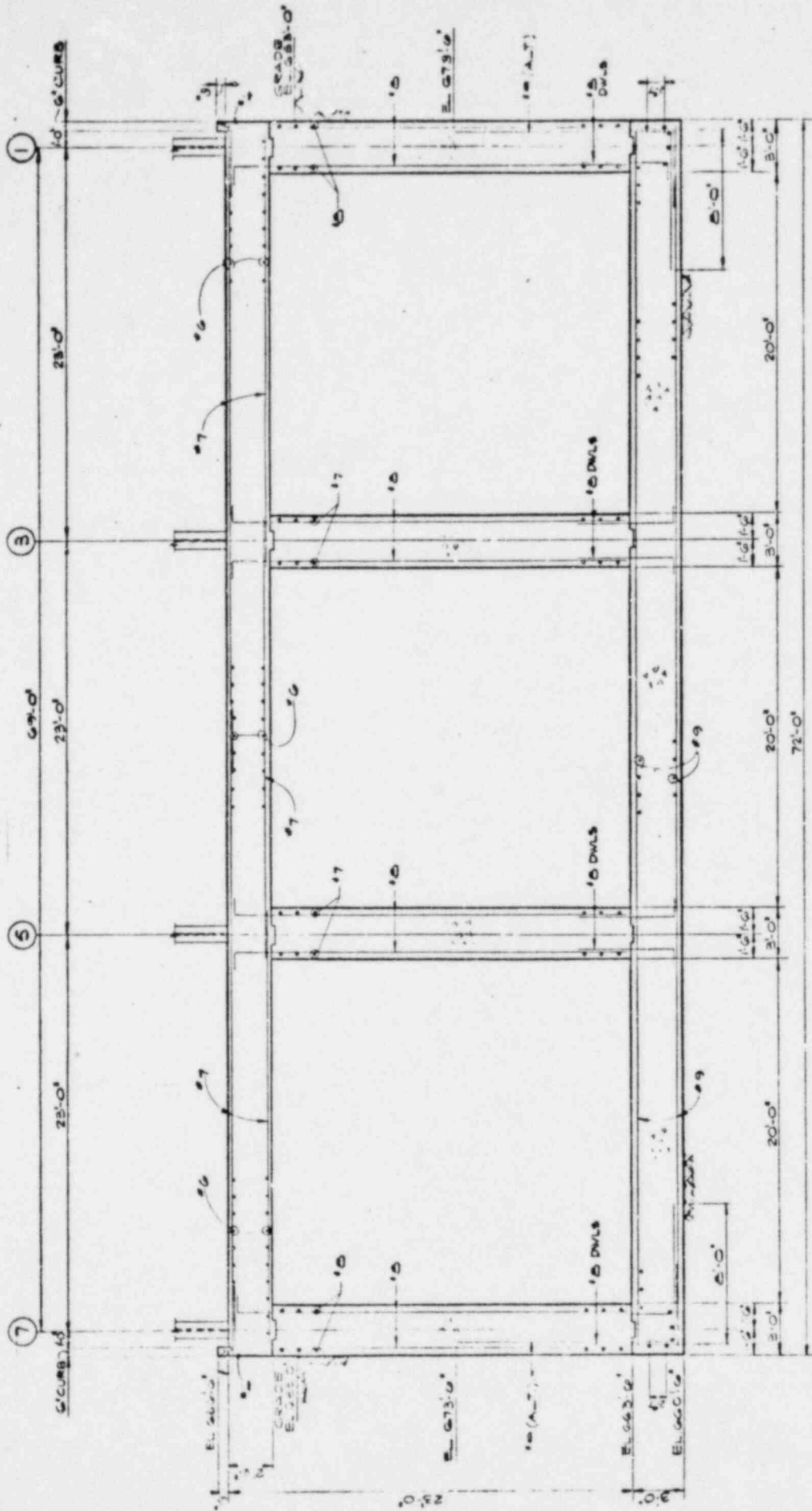


Byron

FIGURE 0.0.01

River Screen House

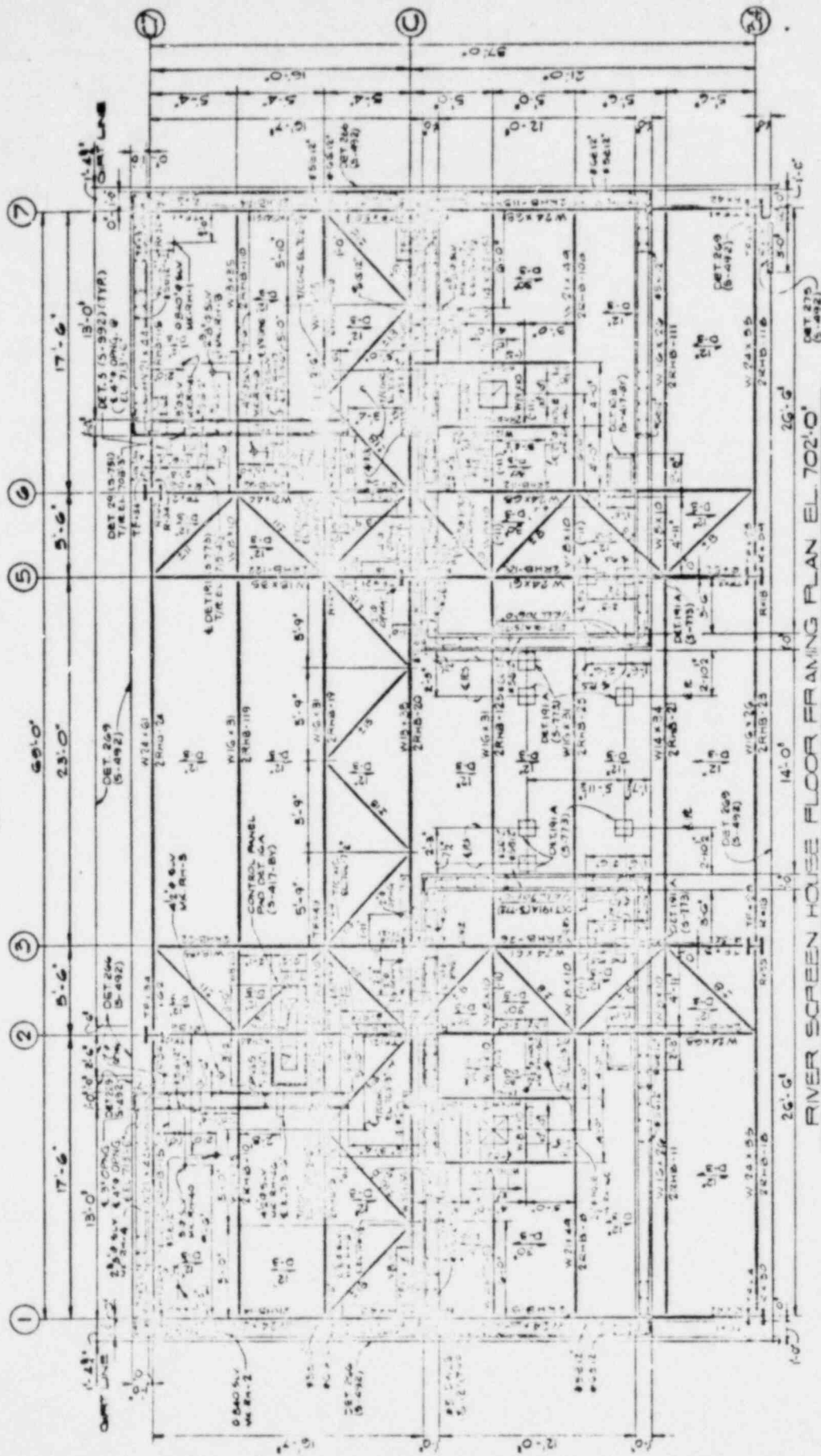
Section



SECTION 9-9

Byron
FIGURE B-B-02

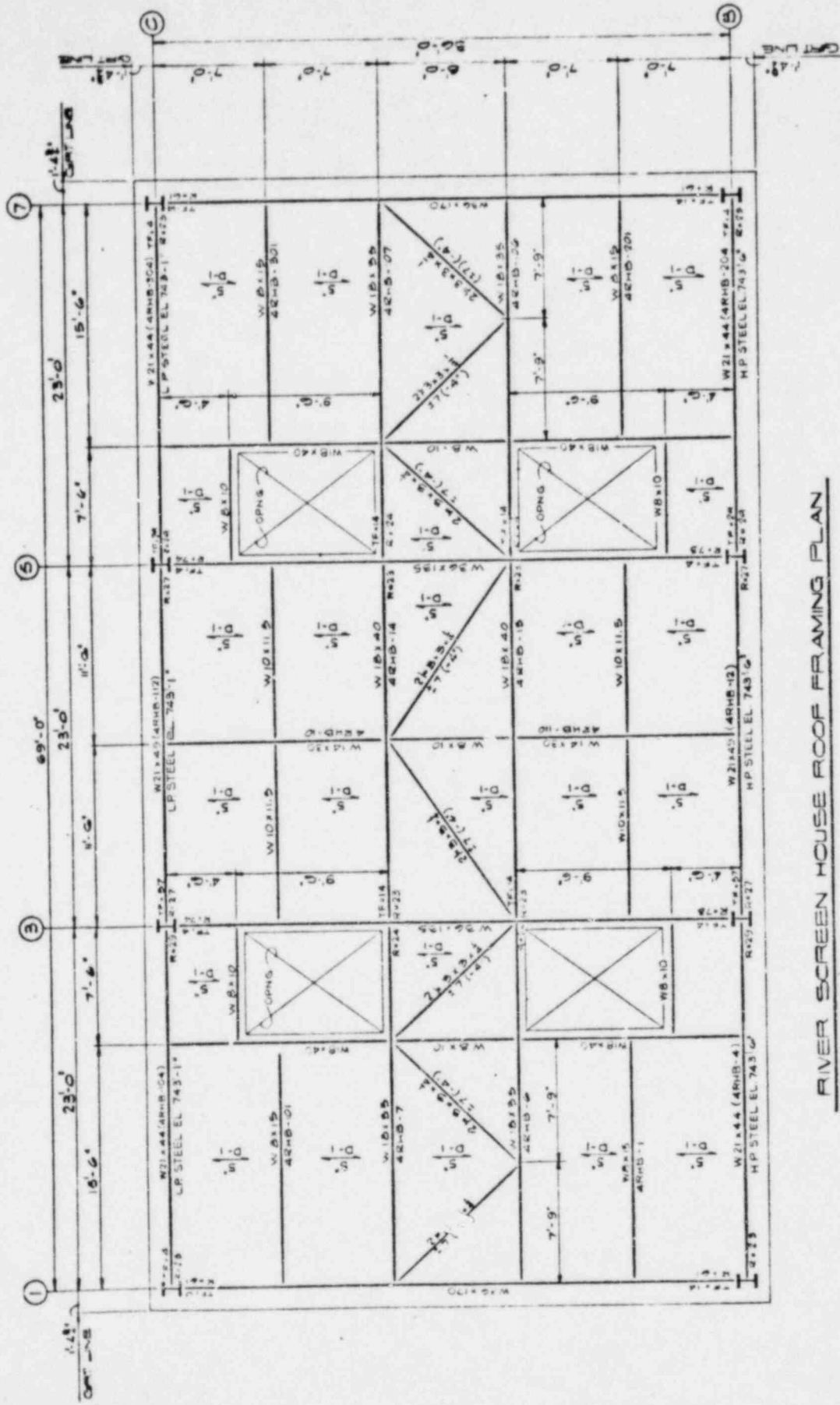
River Screen House
Section



RIVER SCREEN HOUSE FLOOR FRAMING PLAN EL 702'-0"

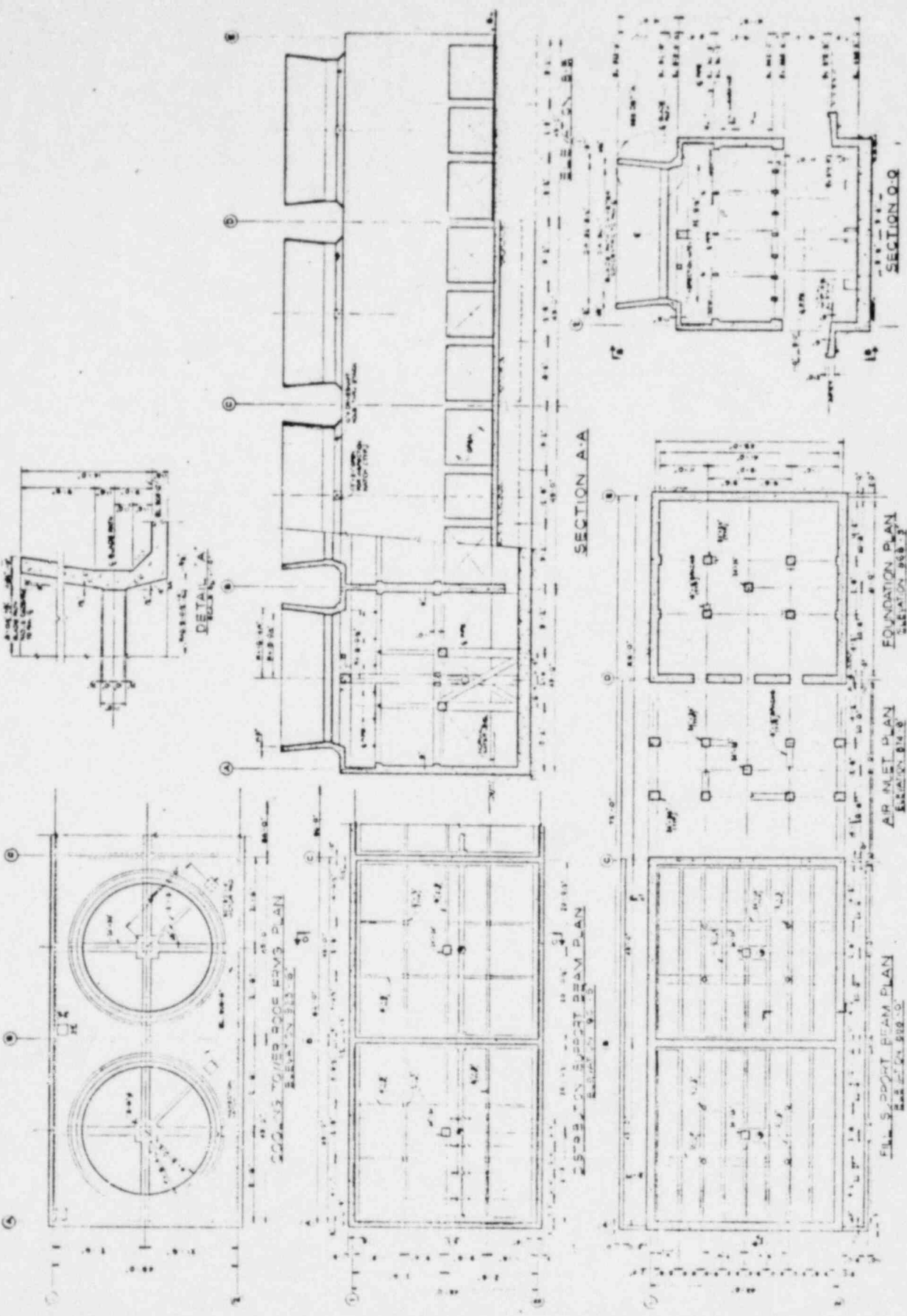
TOP OF STEEL EL 701'-0" (UNLESS NOTED)
ALL DIAGONAL BRACINGS SHALL BE 2x3x3x1/4 EL700'-8 1/2"

Byron
River Screen House
Floor Framing Plan
FIGURE 2.8-6B



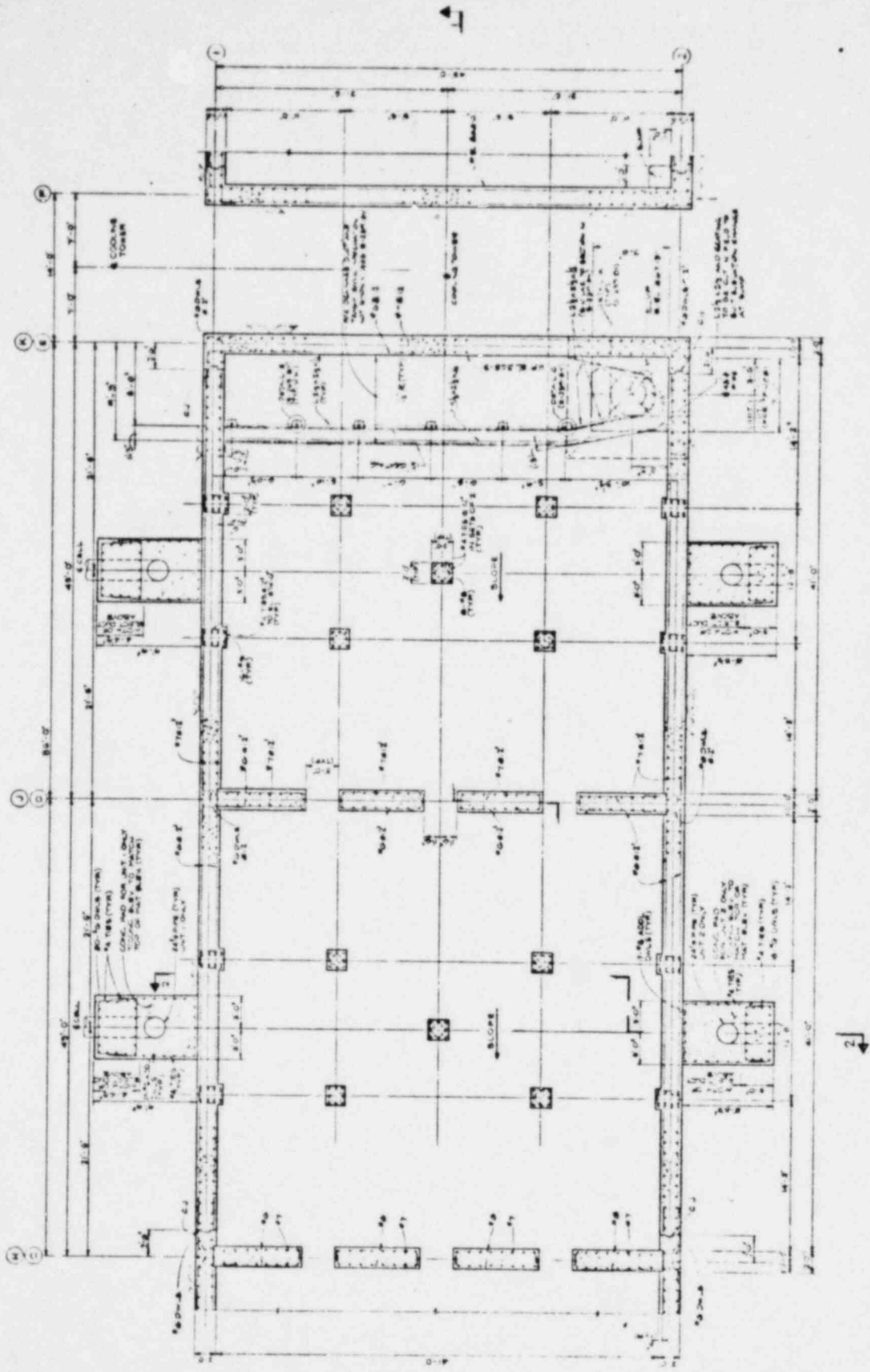
RIVER SCREEN HOUSE ROOF FRAMING PLAN

Byron
 FIGURE 3.8-04
 River Screen House
 Roof Framing Plan



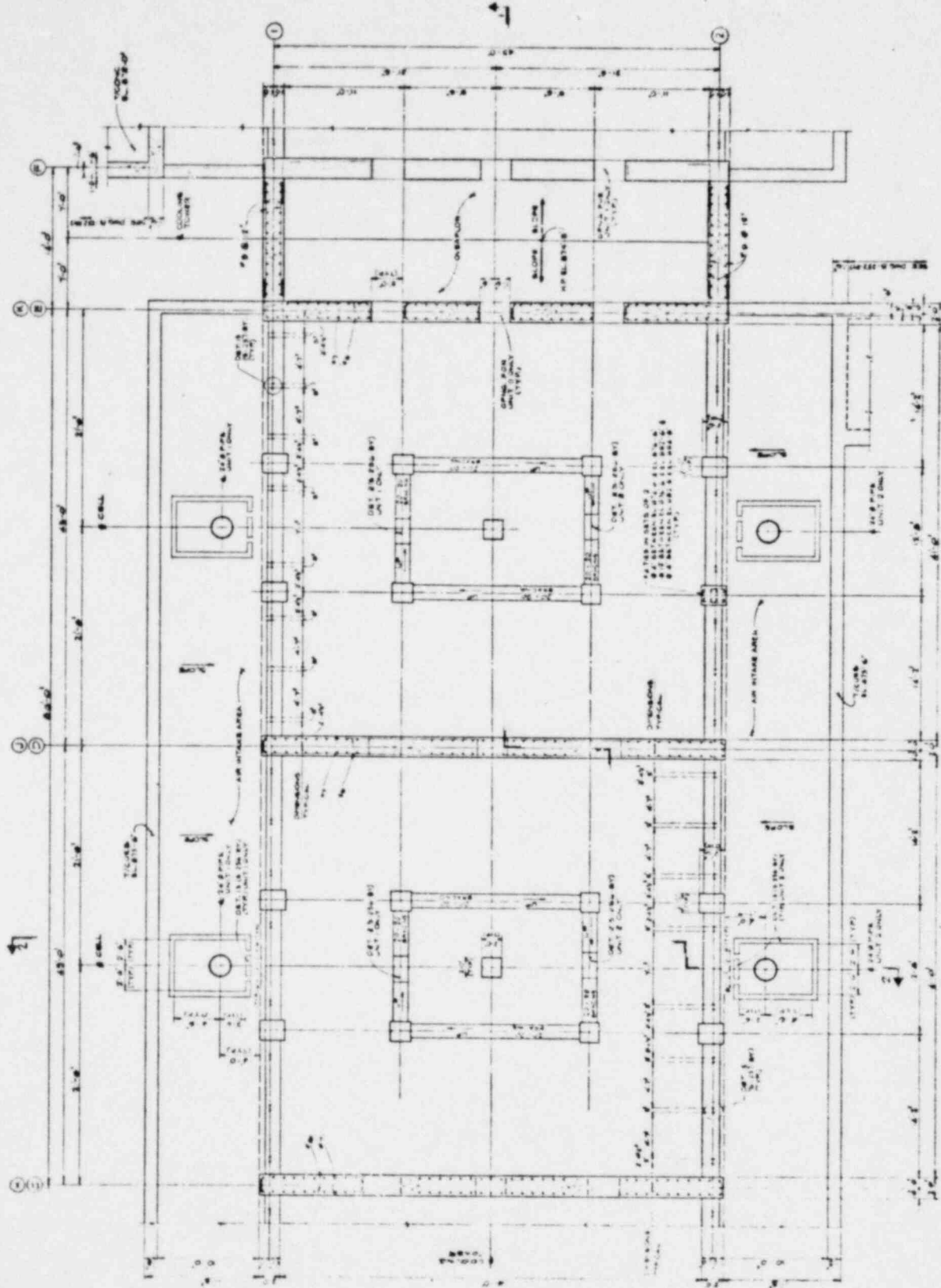
Byron
 FIGURE 2.3-25
 Essential Service
 Cooling Tower

ESSENTIAL SERVICE COOLING TOWER



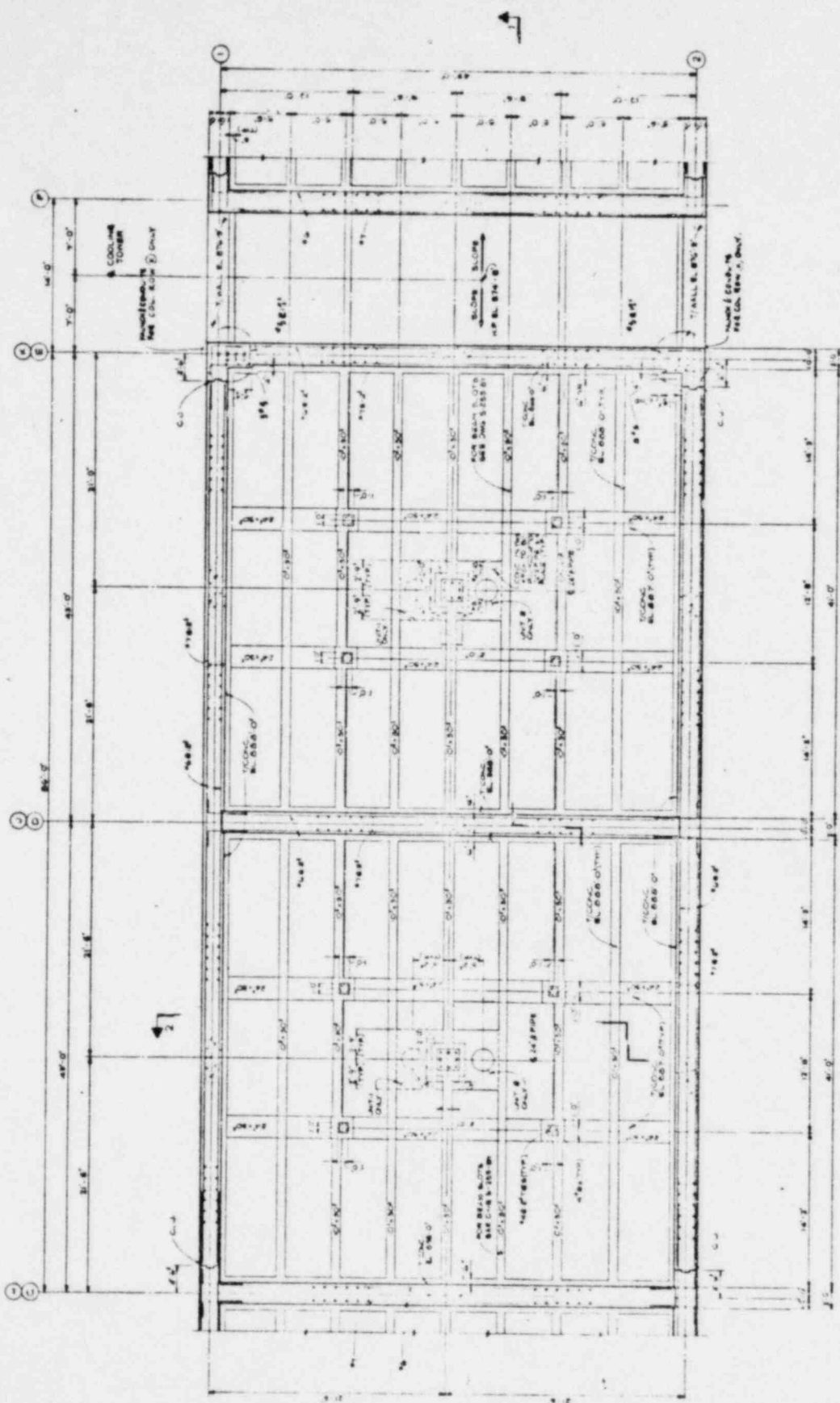
ESSENTIAL SERVICE COOLING TOWER
 FOUNDATION PLAN ELEVATION 868'3"

Byron
 FIGURE 3866
 Essential Service
 Cooling Tower



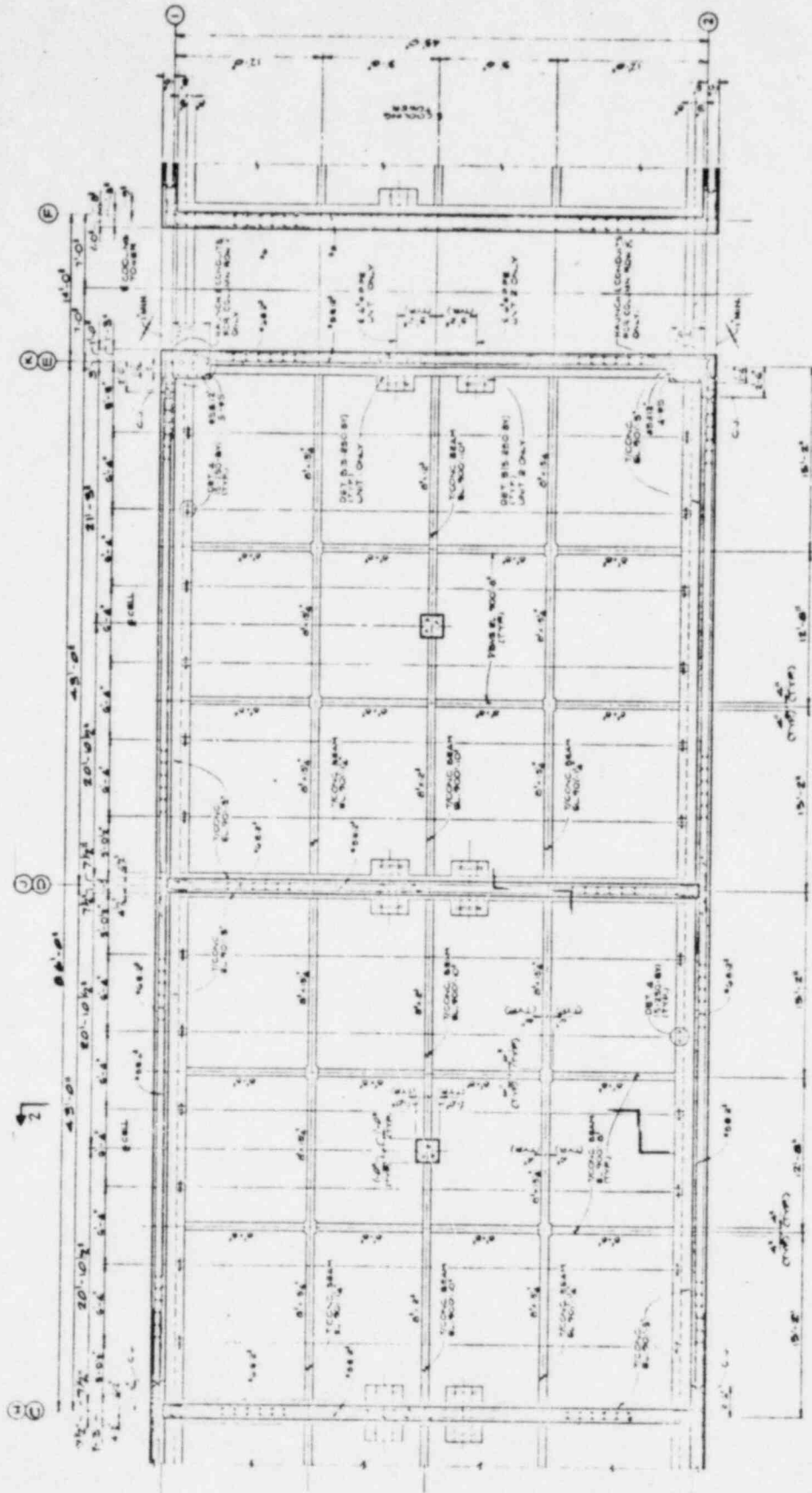
ESSENTIAL SERVICE COOLING TOWER
AIR INLET PLAN - EL. 675.0

Byron
FIGURE 2.2-67
Essential Service Cooling
Tower Air Inlet Plan



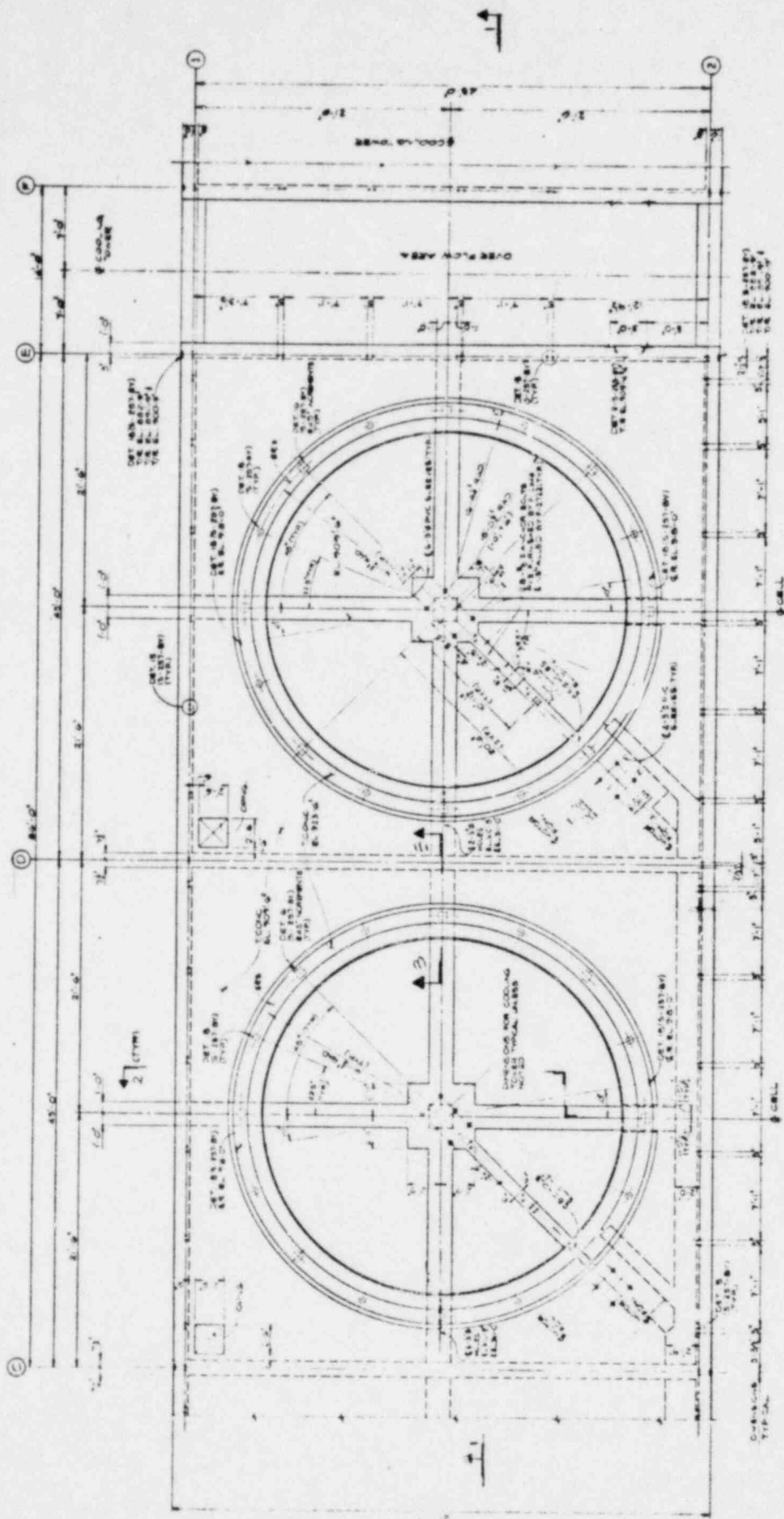
21
 ESSENTIAL SERVICE COOLING TOWER
 FILL SUPPORT - BEAM PLAN
 ELEVATION 888'-0" (FIN)

Byron
 FIGURE 3.0-08
 Essential Service
 Cooling Tower
 Fill Support - Beam
 Plan



2 | ESSENTIAL SERVICE COOLING TOWER
 (1774) | DISTRIBUTION SUPPORT BEAM PLAN

Byron
 FIGURE 2.8-60
 Essential Service
 Cooling Tower
 Distribution Support
 Beam Plan

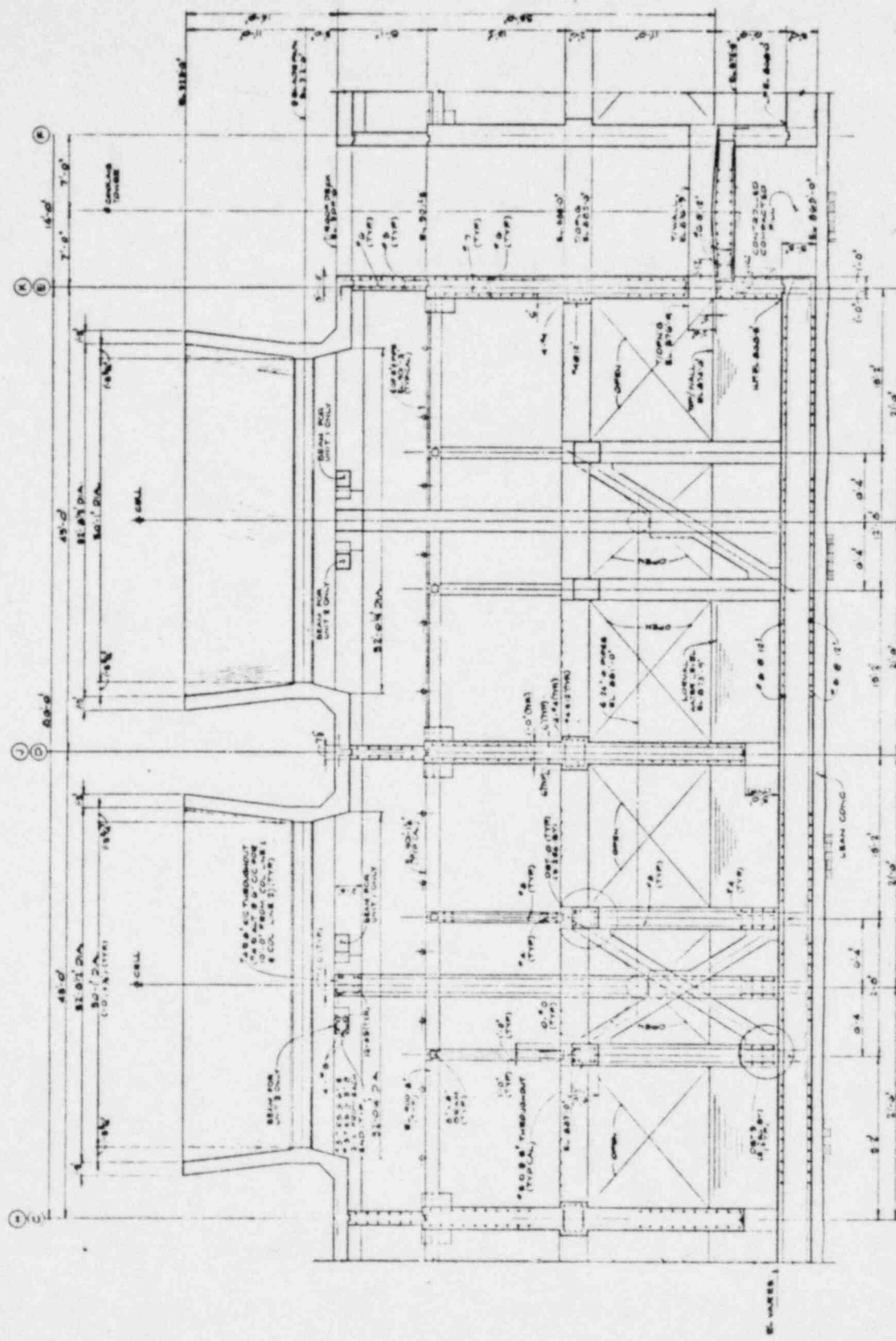


ESSENTIAL SERVICE COOLING TOWER ROOF FRAMING PLAN

Byron

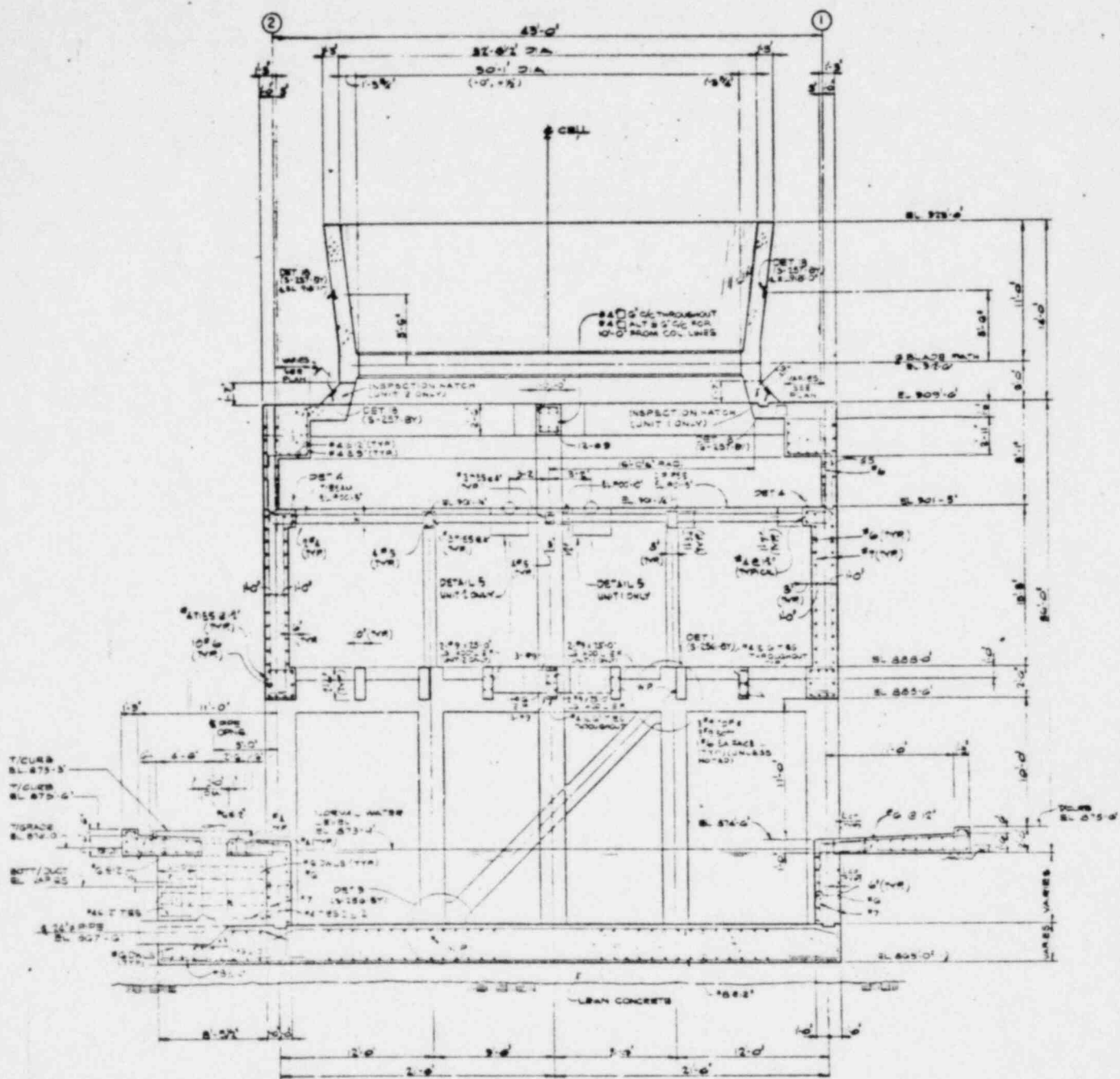
FIGURE 2.8-70

Essential Service
Cooling Tower Roof
Framing Plan



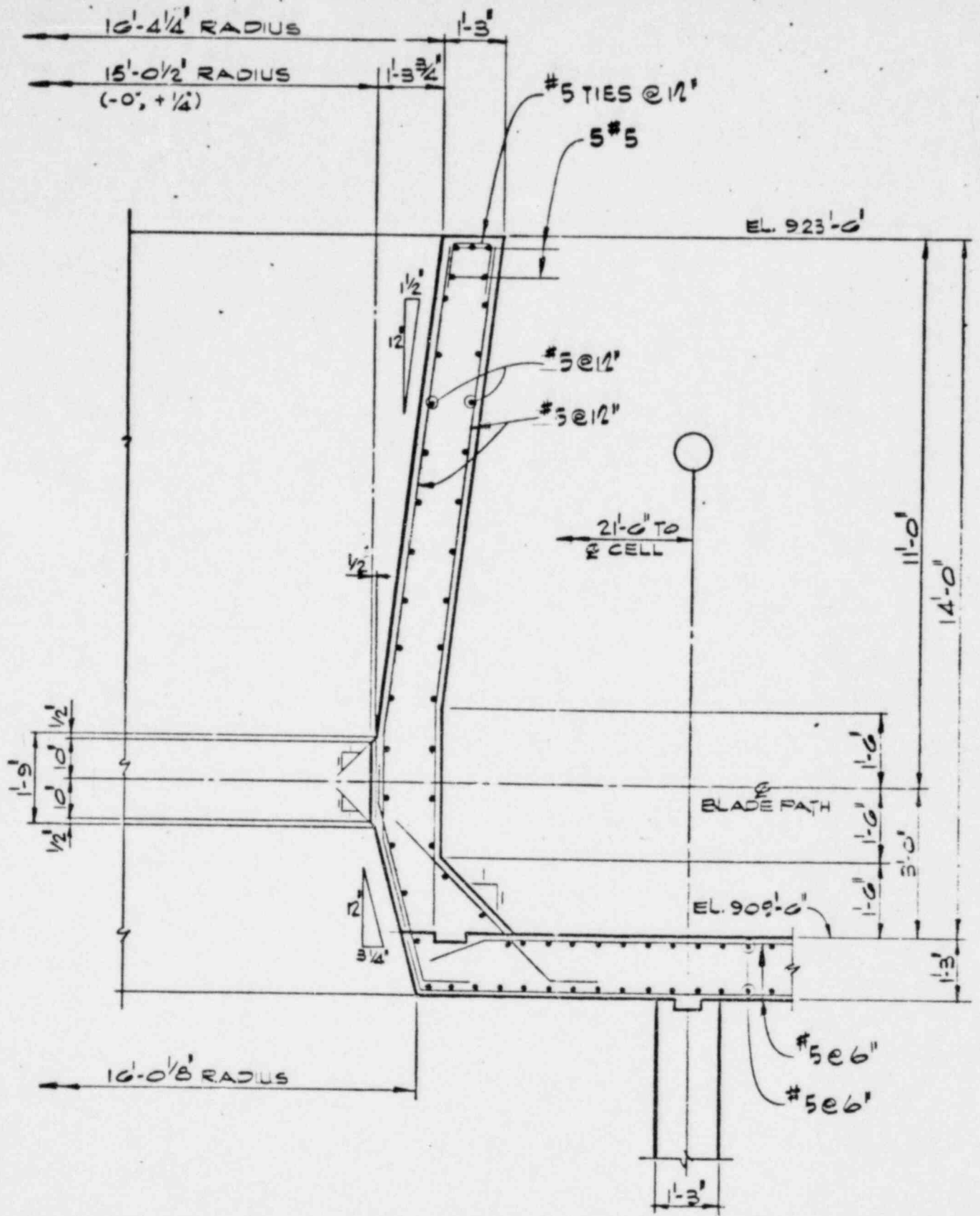
Byron
 FIGURE B.5.5-71
 Essential Service Cooling
 Tower Section

SECTION 1-1



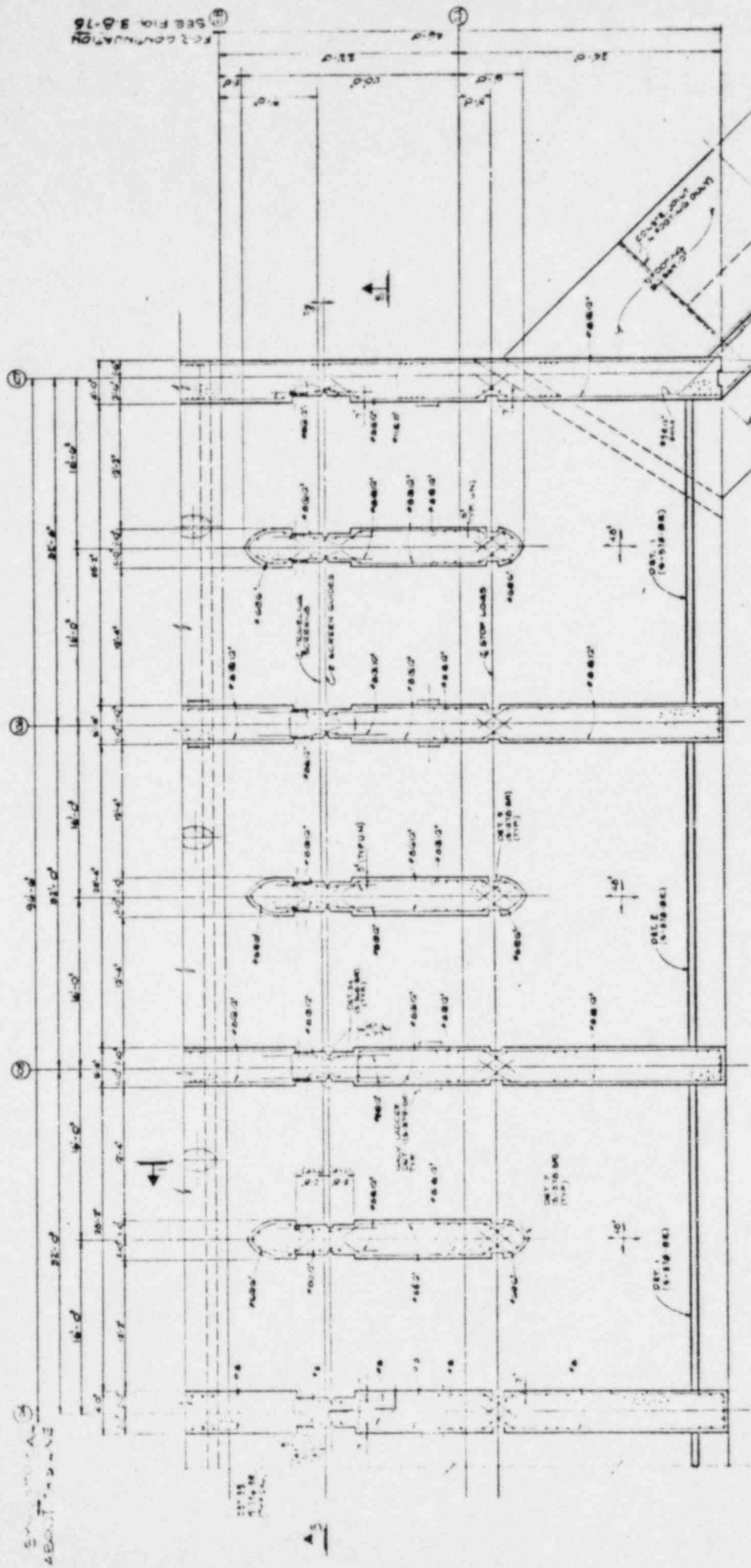
SECTION 2-2

Byron
 FIGURE 3.8-72
 Essential Service Cooling
 Tower Section



SECTION 3-3

Byron
 FIGURE 3.3-73
 Essential Service Cooling Tower Section

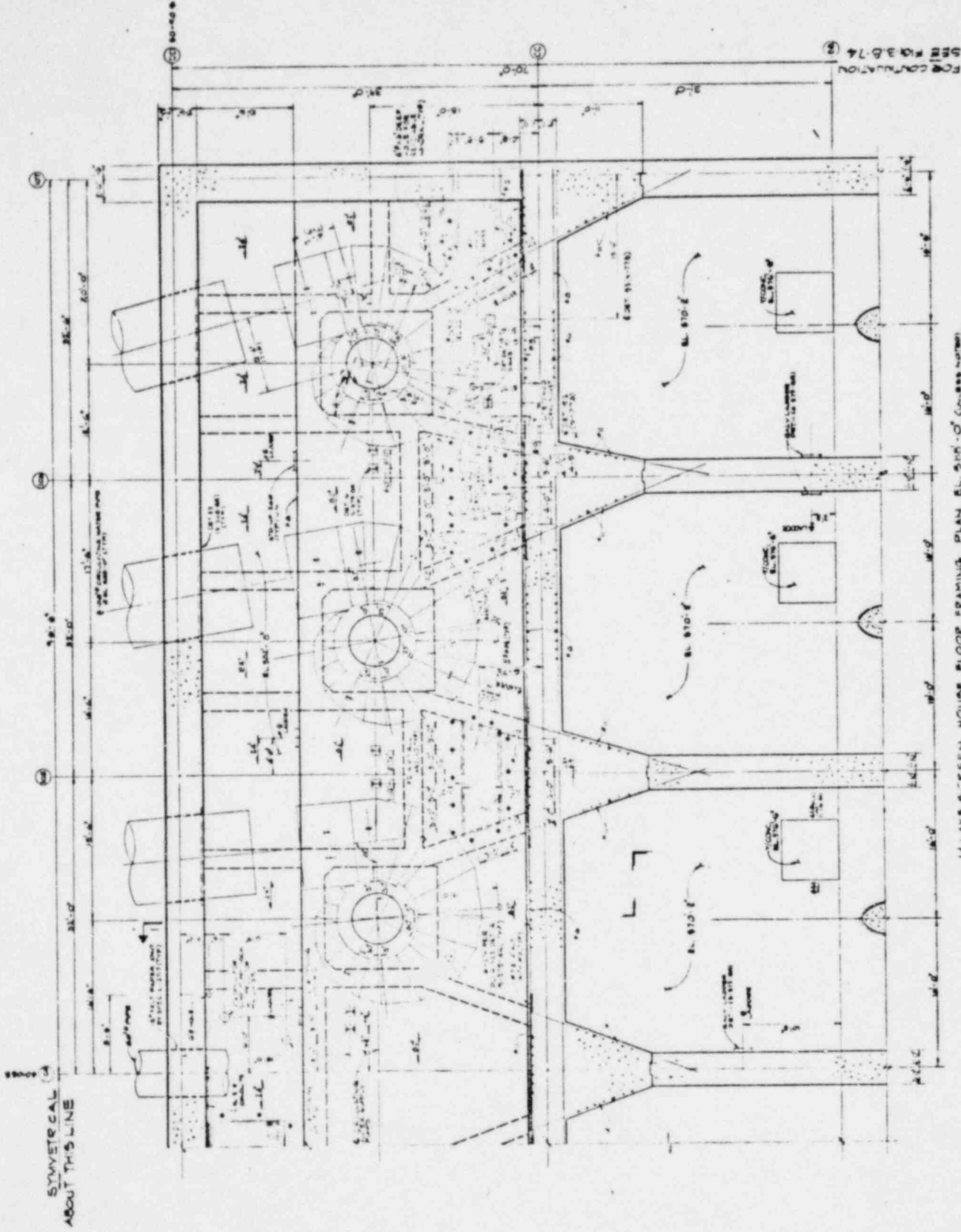


SECTION A-A
 ABOUT PIER 1

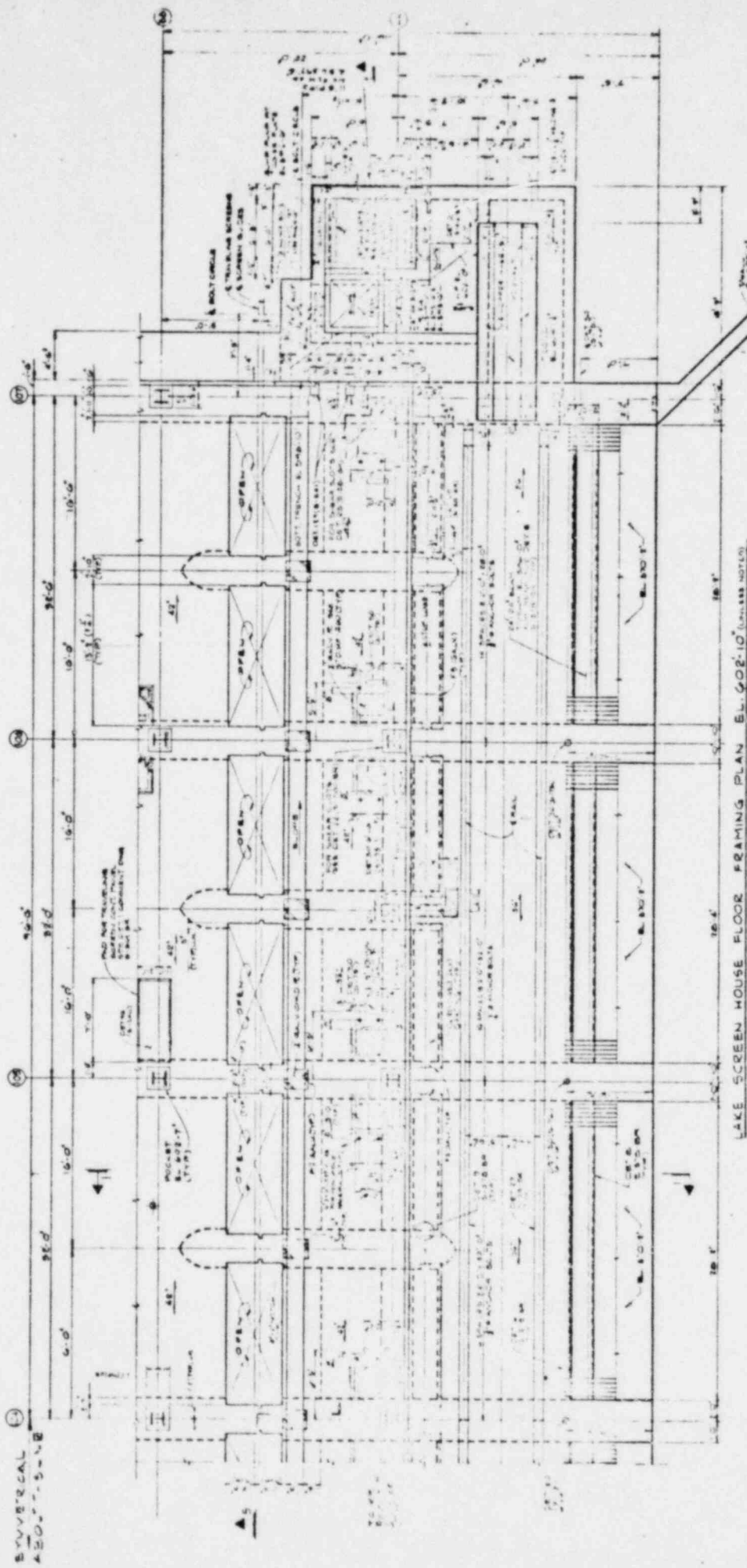
SEE FIG. 5-B-15

LAKE SCREEN HOUSE FOUNDATION PLAN EL. 570'-2"

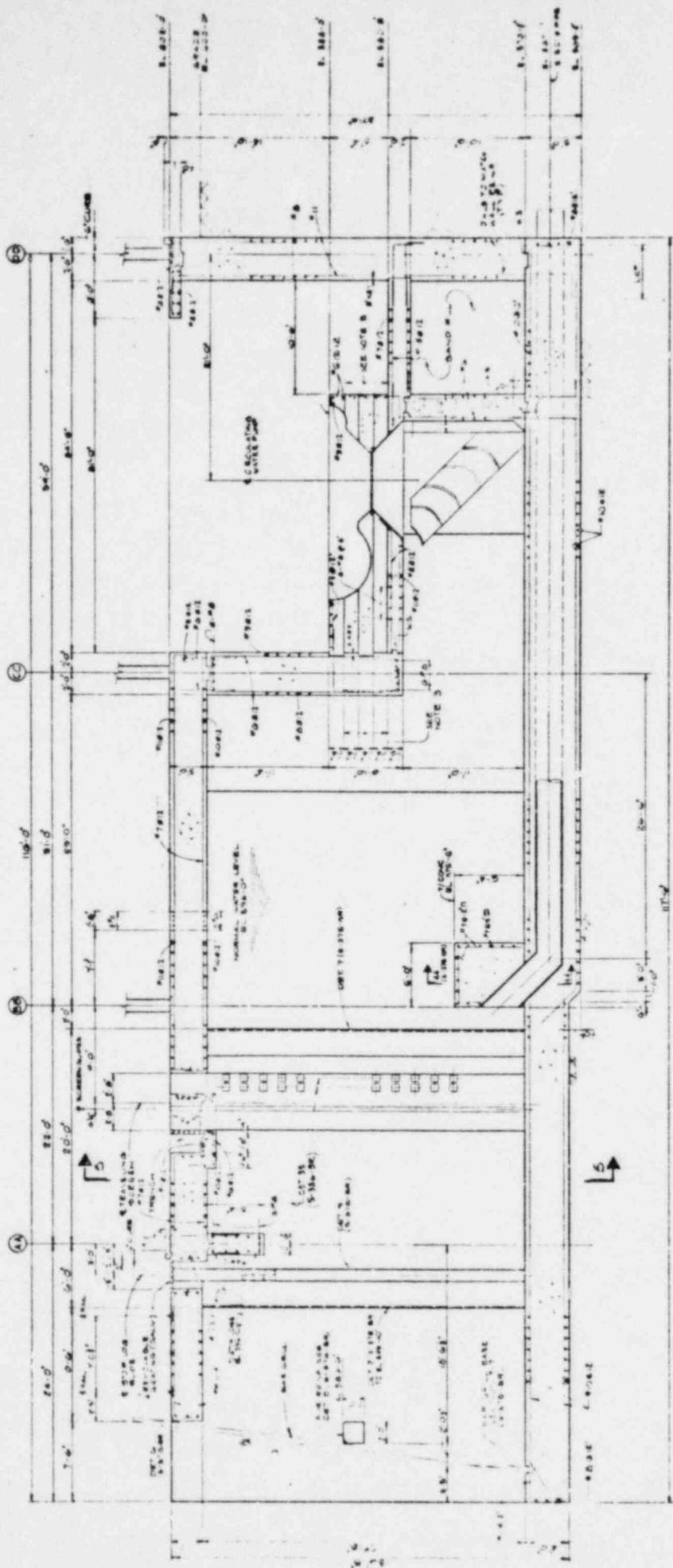
Braidwood
 LAKE SCREEN HOUSE
 FOUNDATION PLAN



Braidwood
 Figure 3.8-75
 Lake Screen House
 Floor Framing Plan

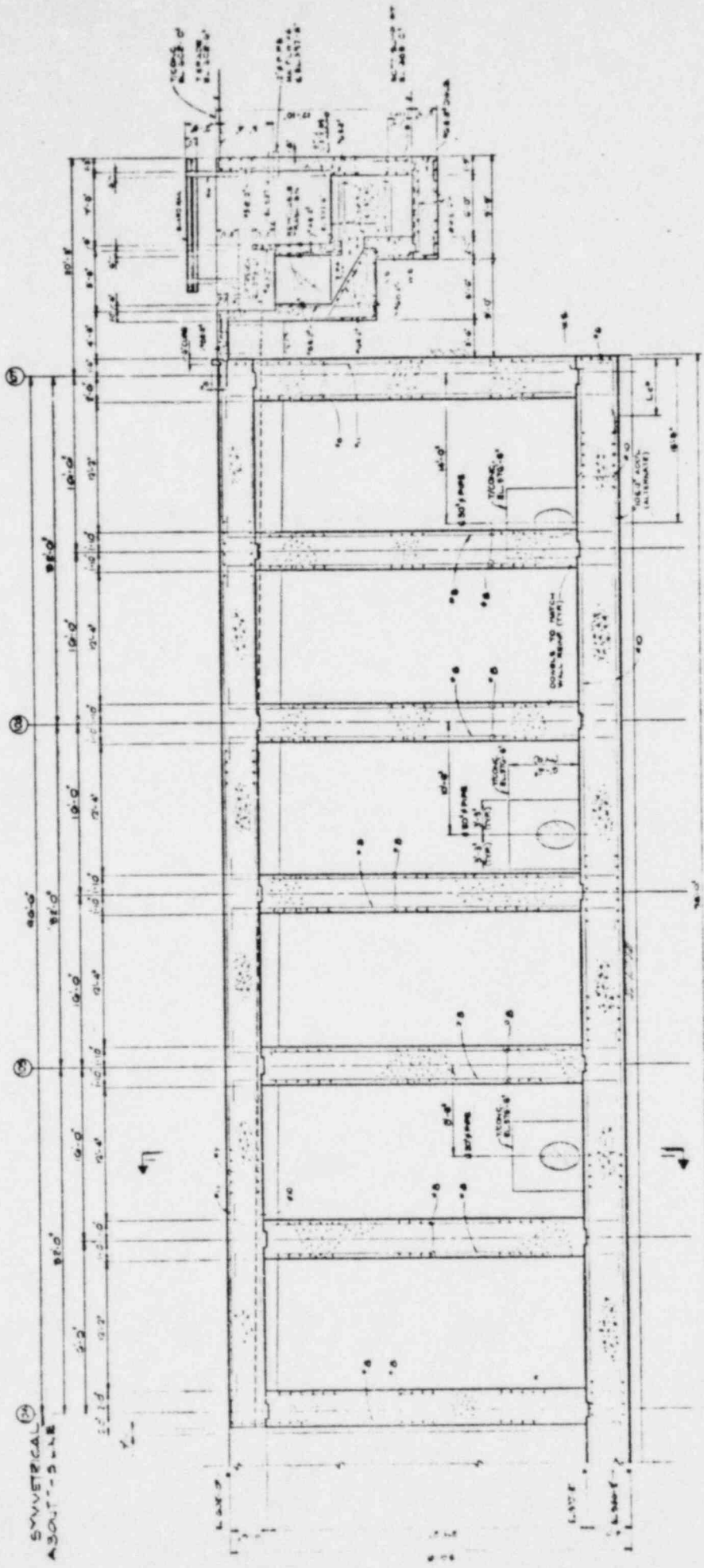


Braidwood
 FIGURE 3.5-7b
 Lake Screen House
 Floor Framing Plan



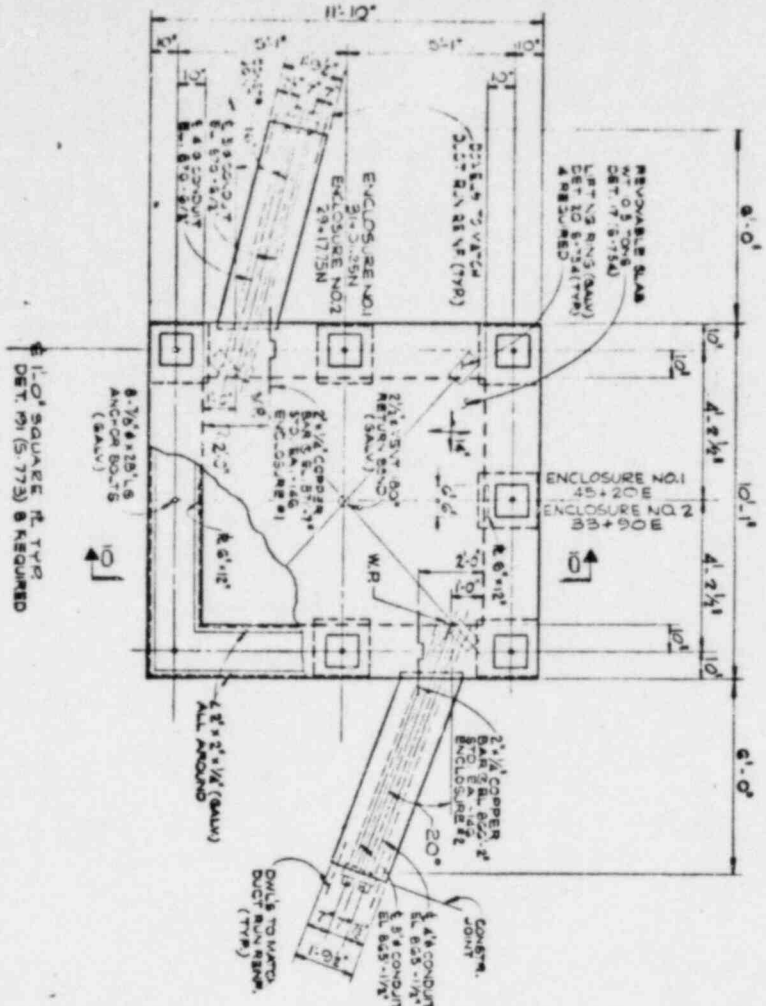
Braidwood
 FIGURE 38-77
 Lake Screen House
 Section

SECTION I-I

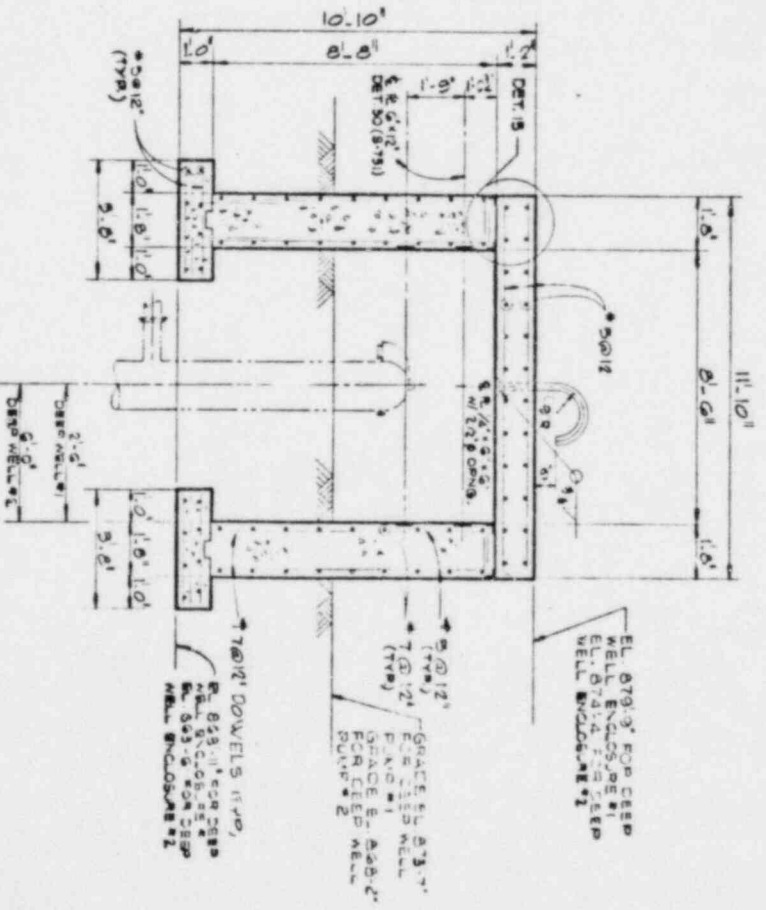


Braidwood
 FIGURE 3.8-70
 Lake Screen House
 Section

SECTION 3-5



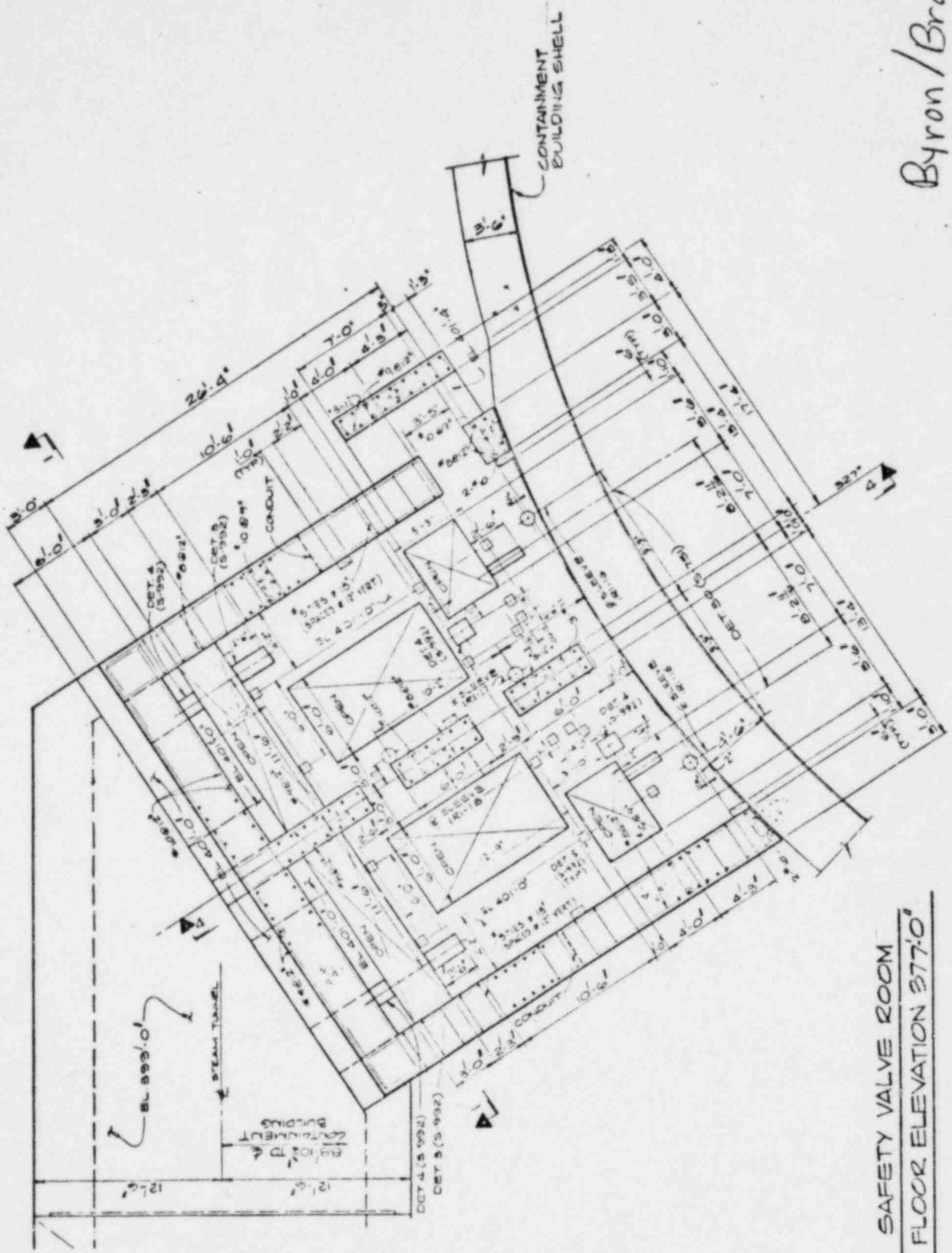
PLAN



SECTION 10-10
SCALE: 3/8" = 1'-0"

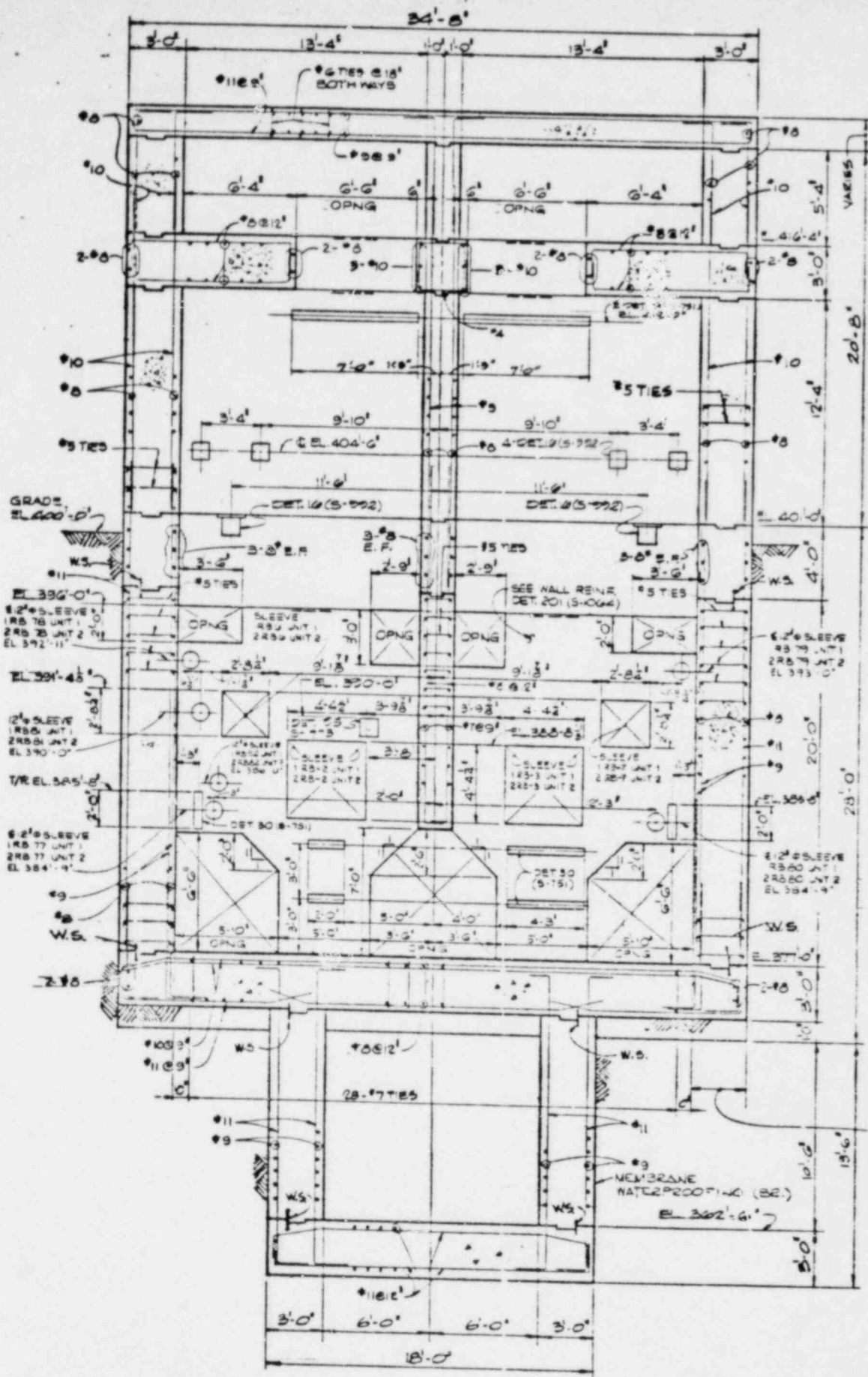
DEEP WELL ENCLOSURES

Byron
FIGURE 3.2-79
Deep Well
Enclosures



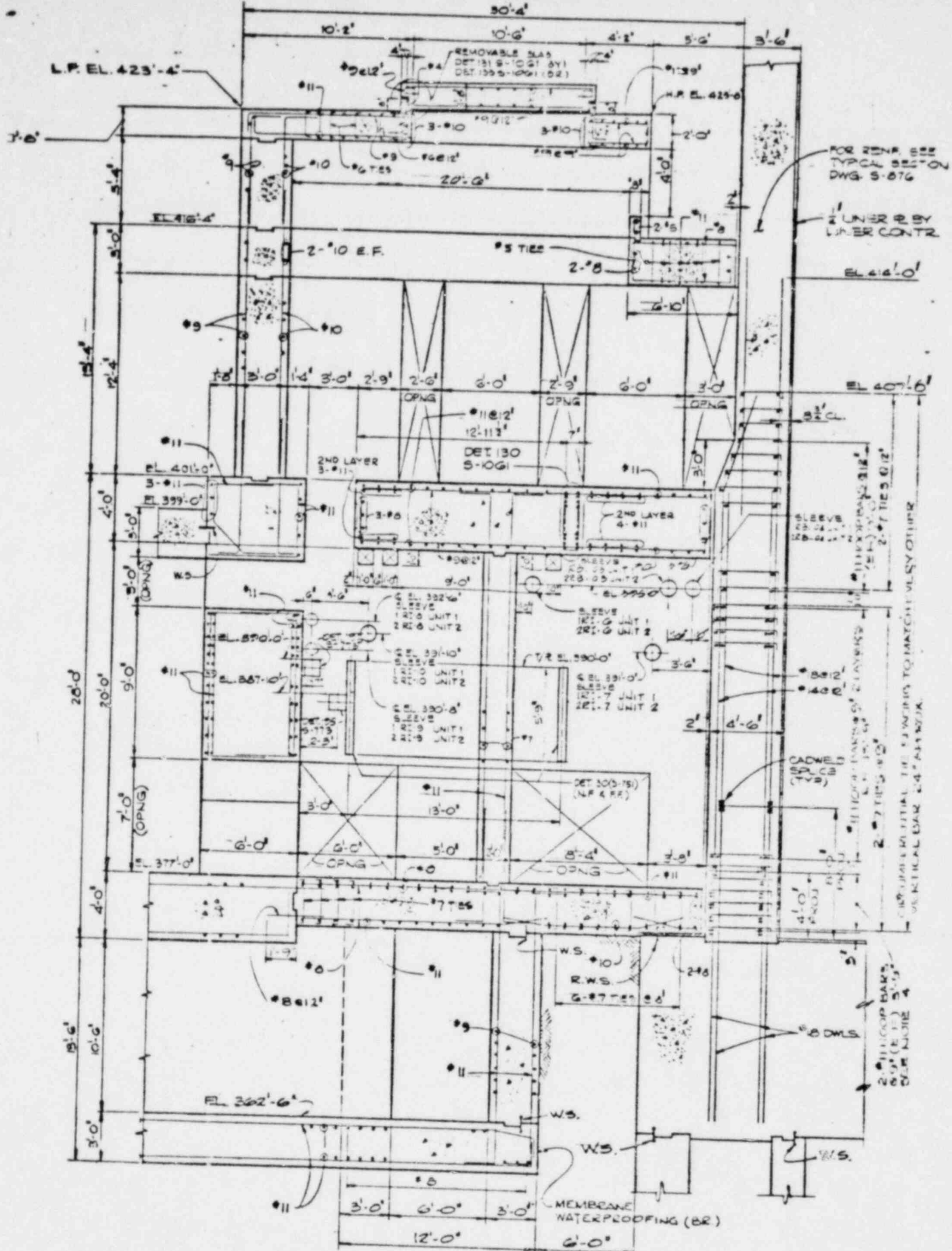
SAFETY VALVE ROOM
FLOOR ELEVATION 377'0"

Byron/Braidwood
 FIGURE B.B.80
 Safety Valve Room



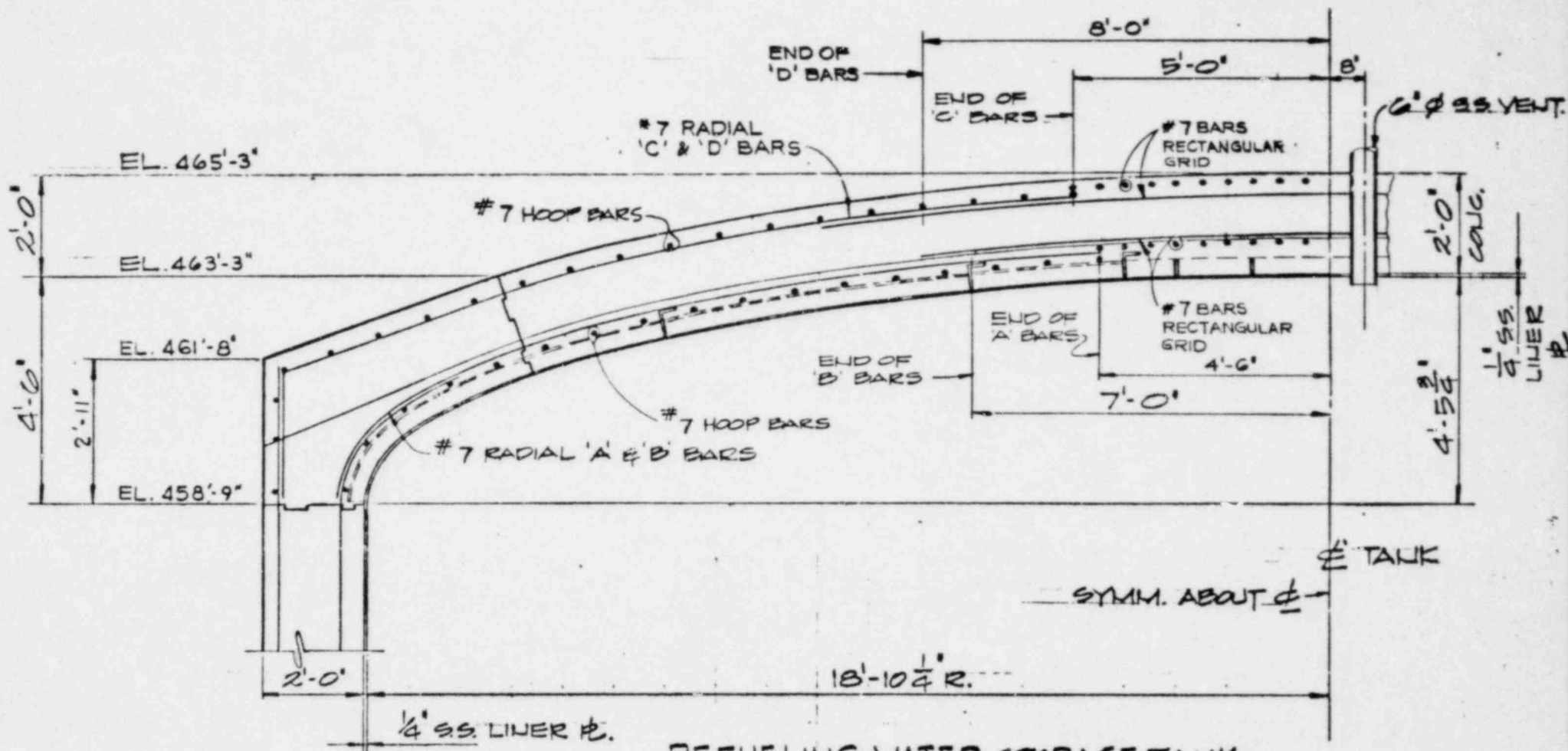
SECTION 1-1

Byron / Braidwood
 FIGURE 3.8-81
 Safety Valve Room Section



SECTION 4-4

Byron/Braidwood
 FIGURE 3.8-32
 Safety Valve Room Section



REFUELING WATER STORAGE TANK
TYPICAL DOME SECTION

Byron/Braidwood
FIGURE 3.8-84
RWST Typical Dome Section

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 4A: Additional information in response to
Question 130.06 (Seismic Reassessment)

1. Average actual material strengths.

S&L to provide methodology of calculating the average actual material strengths.

Rebar - The average actual material strength was calculated from the yield strength values given in the certified material test reports. The heat number and the yield strength values of all the reinforcing steel was listed and averaged. To assure this average value did not exceed 70% of the ultimate strength, 100 randomly selected heat numbers were listed. The ultimate strength values of these heats were recorded and averaged. The average actual material strength did not exceed 70% of the average ultimate strength.

Structural Steel - A random sample of 287 of the approximate total of 700 certified material test reports for A-36 material were listed with the heat number and yield strengths. These yield strength values were averaged. A similar sampling of 107 of the approximate total of 200 reports were listed and averaged for A-588 or A-572 Grade 50 material. The averaged yield strength values were compared to the respective average ultimate strength values of approximately 100 test values. The average yield strength values did not exceed 70% of the average ultimate strength values.

Concrete - The average compressive strengths at 91 days for both 3500 psi and 5500 psi concrete has been calculated. This average is computed based on all the in-process tests taken during construction.

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 4A: Additional information in response to
Question 130.06 (Seismic Reassessment)

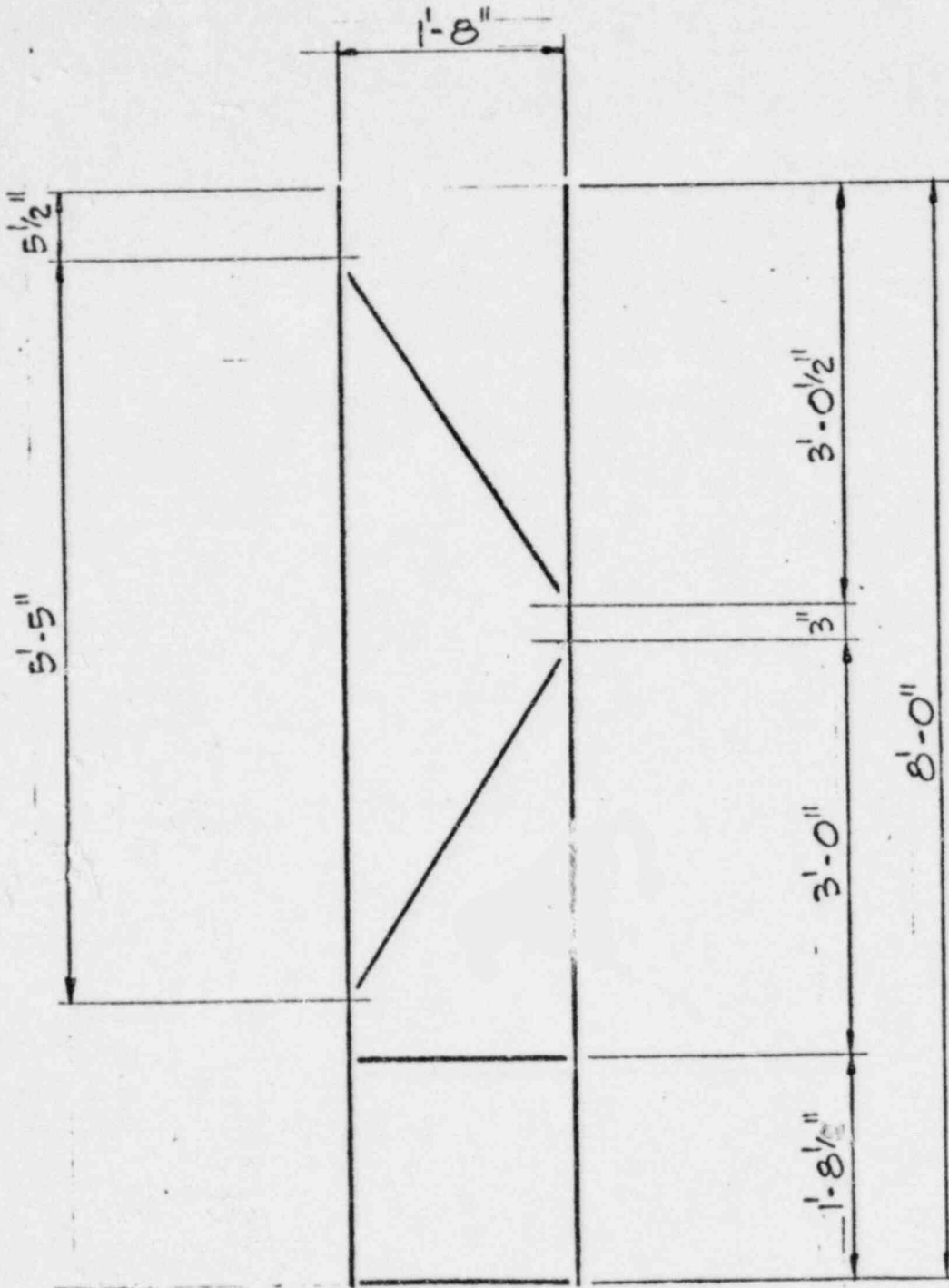
2. HVAC Supports

S&L to assess the behavior of the HVAC supports which have been determined to exceed the yield strength under an SSE event but will not collapse.

HVAC support structure number 3902 (sketch attached) as a whole will behave elastically and will not collapse under the specified loads for the following reasons:

- a) The maximum deflection at the free end of the structure is .353 inch, which gives a reasonably small value of $1/270$ for the Δ/L ratio.
- b) The structure as a whole will experience stresses below yield, except for the 5-1/2 inch-long leg near the support, where the member will experience a local inelastic joint rotation due to an increase in the moment of 47% above the elastic limit. This increase results in a joint rotation ductility factor of 1.95, which is considered very low.
- c) The largest b/t ratio of the members is 7, compared to the 8.5 permitted by Section 2.7 of the AISC Specification for projecting elements in compression.
- d) An increase of about 10% to 12% in the yield strength of the material under dynamic loading has not been utilized.

BYRON/BRAIDWOOD
ACTION 4A
HVAC SUPPORTS (Cont'd.)



HVAC SUPPORT STRUCTURE #3902

ALL MEMBERS: L 1 1/2 x 1 1/2 x 3/16 (F_y = 36 KSI)

SCALE: 3/4" = 1'-0"

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 4A: Additional information in response to
Question 130.06 (Seismic Reassessment)

3. Unique differences for the Marble Hill Model

S&L to list the unique structural features on the Marble Hill plant.

The unique structural differences are as follows:

- 1) One 40-foot long wall in the Auxiliary Building at elevation 401'-0" is 1'-0" thick at Byron/Braidwood and 2'-0" thick at Marble Hill.
- 2) Select columns in the Marble Hill Auxiliary Building at elevations 346'-0", 364'-0", and 383'-0" have a concrete strength of 5500 psi rather than 3500 psi.

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 4A: Additional information in response to
Question 130.06 (Seismic Reassessment)

4. Selection Method for OBE Reassessment of Structural
Steel Beams

S&L to provide methodology in selecting structural steel
beams for reassessment due to an OBE event.

The eleven structural steel beams in the Auxiliary Building
and the eight structural steel beams in the Containment
Building were selected for the OBE comparison because these
beams showed the highest stresses under the SSE reassessment.

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

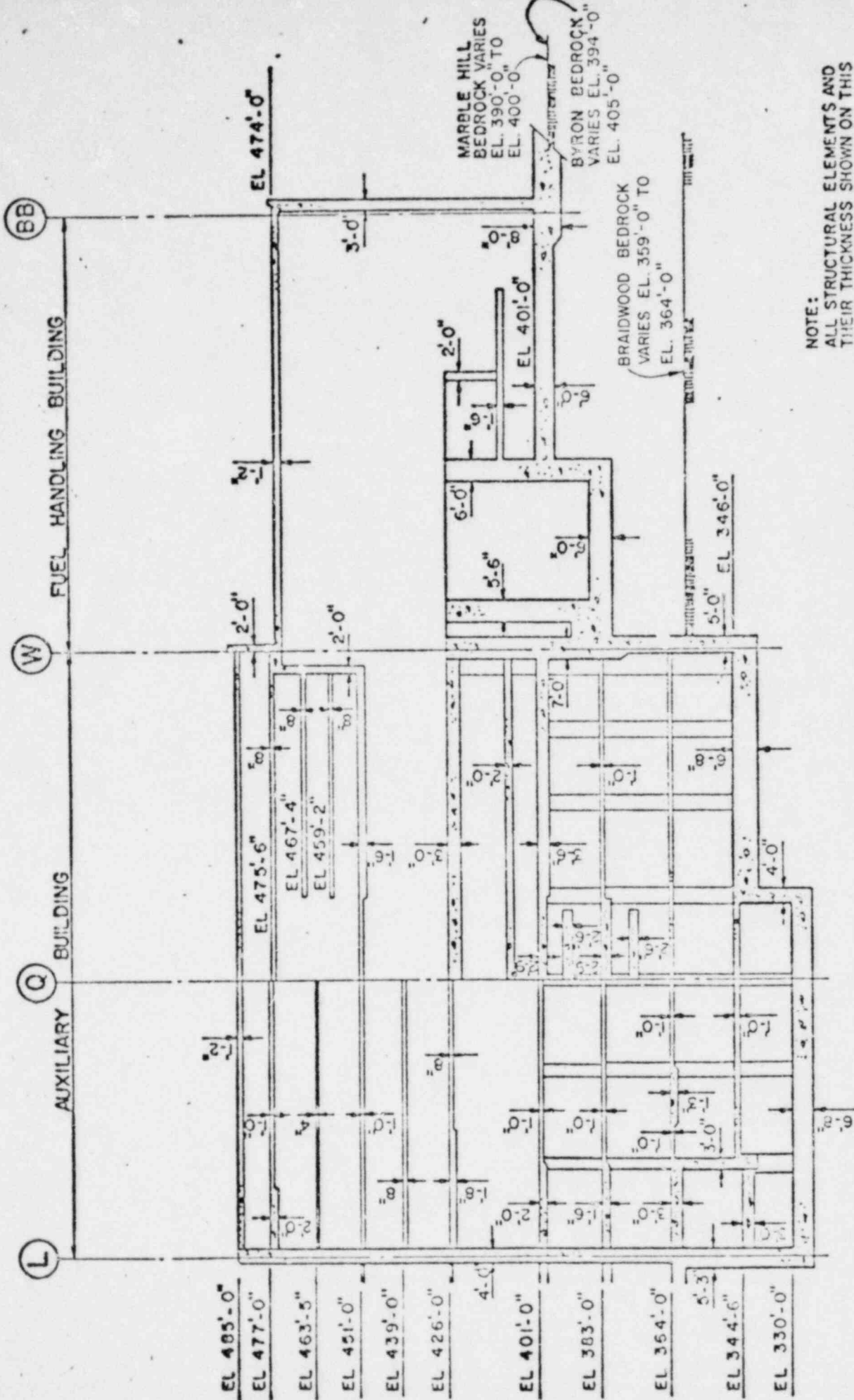
Action Item 4A: Additional information in response to
Question 130.06 (Seismic Reassessment)

5. Figure 130.06-2

S&L to provide the bedrock elevation for the Byron, Braidwood and Marble Hill sites. S&L also to provide the extent of backfill under the Fuel Handling Building at the Braidwood site.

Refer to attached Figure 130.06-2 for bedrock elevations.

At the Braidwood site, compacted fill with a minimum relative density of 85% was placed on the glacial till which is at elevation 378'-0". Refer to FSAR Figure 2.5-76 Section 2-2.



NOTE:
 ALL STRUCTURAL ELEMENTS AND
 THEIR THICKNESS SHOWN ON THIS
 SKETCH ARE SAME FOR
 BYRON/BRAIDWOOD AS WELL AS
 MARBLE HILL PROJECTS.

AUXILIARY AND FUEL HANDLING BUILDINGS
 FIGURE 130.06-2

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 4A: Additional information in response to
Question 130.06 (Seismic Reassessment)

6. Unique areas of the Auxiliary-Fuel Handling Building Complex

S&L to explain the unique areas of the Auxiliary-Fuel Handling Building mat design as shown in Figure 130.06-3.

The unique areas of the mat are unique only for the reinforcing steel. Concrete strengths and thicknesses are the same. The Marble Hill design assumptions used were more conservative; therefore the reinforcing in these areas was increased. None of these areas using the Byron/Braidwood design show any over-stress during reassessment.

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 19: Two-inch thick premolded filler joint between
Containment and Auxiliary Building

S&L to provide physical properties of Ethafoam.

A copy of the manufacturer's product literature is being provided to the NRC which gives the physical properties of Ethafoam. Based on these properties, the forces generated on the structures due to relative seismic displacement were accounted for in the design of the Containment and Auxiliary Building walls.

Important Advantages Of ETHAFOAM* Brand Polyethylene Foam

- Resilient — resists multiple impacts
- Lightweight — saves handling and shipping costs — and it's buoyant
- Moisture resistant — closed cell structure resists water pickup and moisture transmission
- Chemical resistant
- Available in fabricated shapes through extensive fabricator network
- Stable over wide temperature range
- Compressible — select compressive strength
- Workable — easy to make prototypes
- Reusable

*Trademark of The Dow Chemical Company

ETHAFOAM Brand Polyethylene Foam

Because of its unusual combination of characteristics and properties, ETHAFOAM brand polyethylene foam is used extensively in:

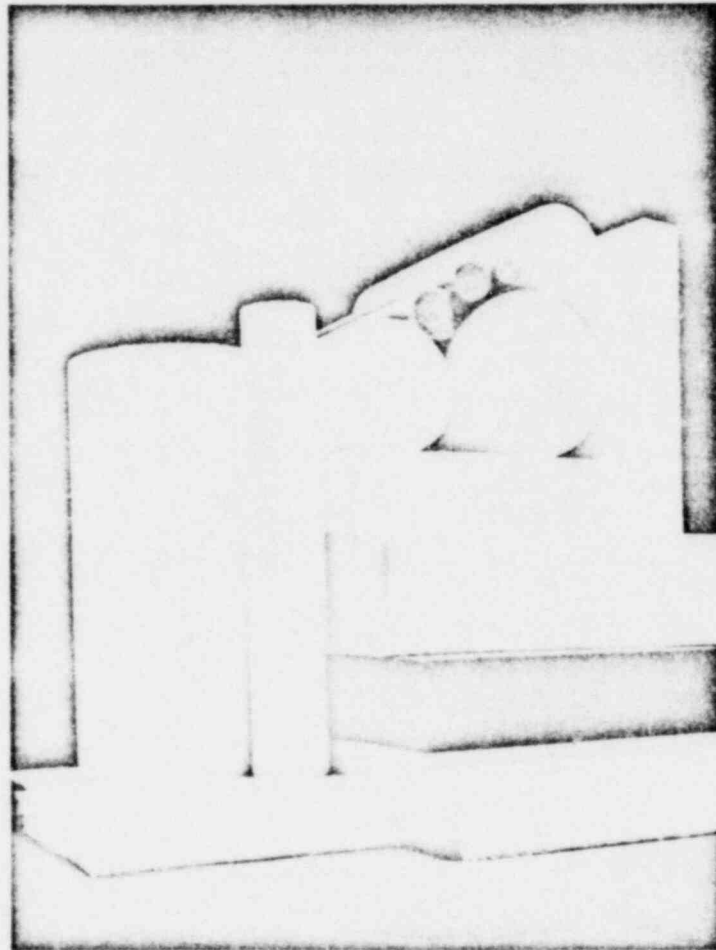
- Cushion Packaging
- Transportation Applications
- Water Sports Equipment
- Industrial Flotation
- Sporting and Recreation Goods
- Construction

In addition, engineers and designers continue to find new applications for ETHAFOAM where no satisfactory material was previously available.

ETHAFOAM expanded polyethylene foam is available in black and natural white, and in a selection of shapes and densities. A tough, flexible closed-cell material, it is:

- Energy-absorbent
- Resilient
- Lightweight
- Flexible over a wide temperature range
- Resistant to chemicals
- Buoyant
- Easy to fabricate

This booklet is intended to provide basic information on the wide range of applications and the outstanding properties of this unique foam material.



Markets And Applications

The past and present applications listed below in which ETHAFOAM has been selected for use, illustrate the versatility of ETHAFOAM brand products, and may suggest potential new uses. **Any applications, however, should be tested thoroughly by the user to determine the suitability of the material for a specific use.**

Cushion Packaging

- Corner Pads
- Pads and Saddles
- Encapsulation
- Case Inserts
- Overwrap Sheeting
- Bracing and Blocking
- Display Cases

Transportation

- Dust and Water Gaskets
- Carpet Underlayment
- Protective Separators
- Aircraft Seating
- Child Seat Cushioning
- Car Roof Underlayment

Government Applications

- Munition Wadding and Plugs
- Filler Pads
- Missile Containerization
- Instrument Cushioning

Water Sports Equipment

- Life Vests
- Ski Belts
- Kickboards
- Games
- Water Lounges
- Rings
- Canoe Liners, Seats, Sponsons
- Pool Cove and Liner Backing
- Boat Flotation

Industrial Flotation

- Oil Booms
- Flotation Collars (Pipelines)
- Buoys
- Ship Fenders

Sporting And Recreation Goods

- Gym and Floor Exercise Mats
- Wrestling Mats
- Wall Pads
- Padding — Sports Equipment
- Toboggan Pads
- Ski Lift Seat Pad
- Golf Bag and Backpack Straps
- All-Terrain Vehicle Flotation
- Handlebar Padding
- Athletic Field Pads and Markers
- Archery Targets
- Sleeping Bag Pads

Construction

- Sealant Backer
- Pressure Relief Joint Filler
- Closure Strips
- Curing Blanket
- Gym Floor Underlayment
- Underground Cable Wrap
- Equipment Vibration Dampening
- Seismic Joint Filler
- Cold Water Pipe Insulation

Appliances

- Gaskets
- Tubing
- Vibration Pads
- Grommets
- Tape

Packaging

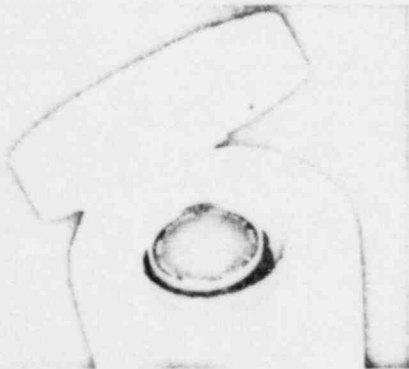
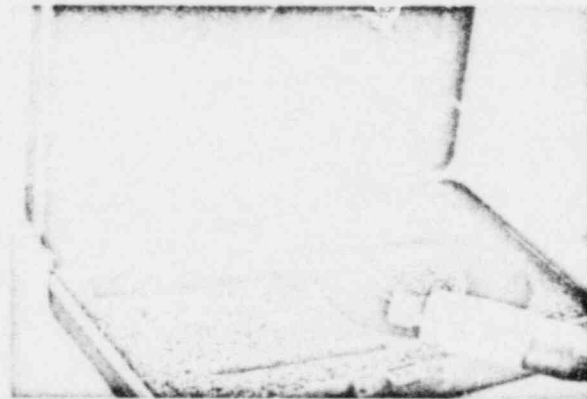
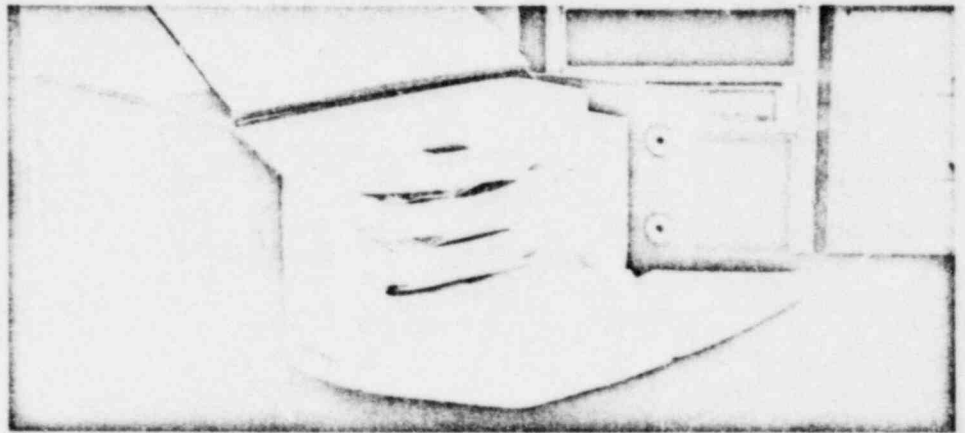
An increasing number of manufacturers are using ETHAFOAM as a cushioning medium in protective packages. ETHAFOAM brand polyethylene foam consists of tiny closed cells, which absorb impact shocks while holding the packaged product in position. Because it is flexible and resilient, the foam also dampens vibrations during handling and shipping.

The resistance of ETHAFOAM brand plastic foam to moisture and water vapor transmission is another important advantage when moisture-sensitive products are to be packaged. In addition the foam is noncorrosive to the packaged product.

The savings gained by using ETHAFOAM can be significant. The material is clean and lightweight and generally may be used repeatedly with minimal loss of effectiveness. Also, it can be fabricated with little tooling expense. These characteristics all contribute to lower costs through reduced transportation charges, lower expenditures for tooling, packing labor and housekeeping,

and reduced waste. Names and addresses of experienced fabricators are available from any of the Dow sales offices listed on the back cover.

For a complete discussion of the use of ETHAFOAM expanded polyethylene in packaging, and information on designing packages with this material, consult Dow's technical bulletin Form No. 172-221, *Designing Packages to Survive Shipping and Handling with ETHAFOAM Brand Polyethylene Foam.*



Sporting And Recreation Equipment

ETHAFOAM brand polyethylene foam is widely used in recreation equipment. Among its advantages are strength, toughness, light weight, good moisture resistance and cushioning properties.

ETHAFOAM has been selected by manufacturers for use in cushioning systems for sports and recreation equipment to help soften impacts or blows.

ETHAFOAM has also been selected for use in athletic equipment, and to provide cushioning in snowmobile seat composites.

Energy-absorbing systems in sports equipment seek to absorb impacts and reduce the prospect of injury. The suitability of any material for use in such systems and the system itself should be evaluated carefully and thoroughly, since no cushioning system can provide absolute protection. In any activity involving motion or height, an individual may receive impacts that exceed the capability of the cushioning system or may be injured from landing in an out-of-control position.

Protective equipment and padding can help reduce the possibility of injuries, but is imperative in any sports activity to promote safety awareness in the participants. Participation in sports without

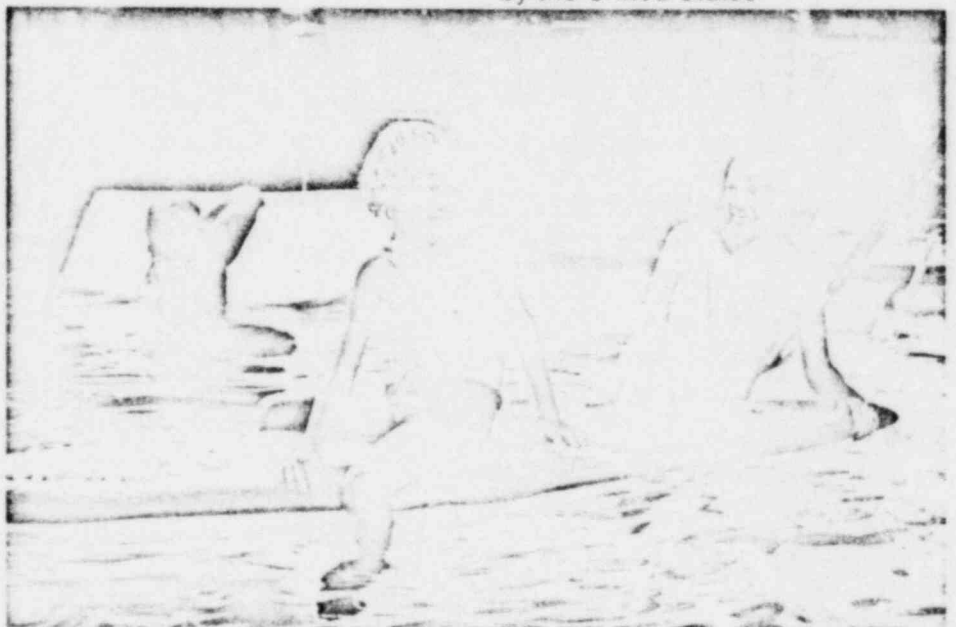
proper training and supervision can be dangerous and should be discouraged.

Water Sports And Aquatic Accessories

ETHAFOAM 220 is about thirty times as light as water.

Because of this excellent buoyancy, and its resilience and toughness, ETHAFOAM brand foam is widely used in equipment for water sports and boating.

Planks and rounds of ETHAFOAM 220 and sheets of ETHAFOAM 221 and ETHAFOAM 222 expanded polyethylene have been approved by the United States

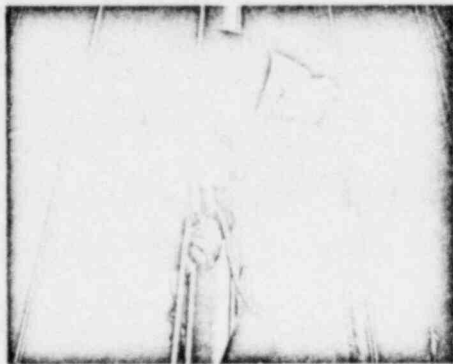


Coast Guard Underwriters Laboratories, Inc. for use in personal flotation devices.

games includes tether ball, ring toss, and water basketball.

ETHAFOAM plastic foam is used in producing floating lounge chairs and marine devices, such as floats and buoys. It also finds use in canoe parts, such as liners, seats and sponsons.

For swimmers, water skiers and boaters, water ski belts, buoyant jackets, surfboards, and kickboards are made using ETHAFOAM brand polyethylene foam.



Transportation

The automotive industry is another user of ETHAFOAM brand plastic foam. Components of cars, trucks, aircraft, and other vehicles are fabricated from this tough, resilient material.

The flexibility and compressibility of this foam permit it to conform to surface variations.

Dust and water gaskets of ETHAFOAM are economical and efficient. Such gaskets are durable, even in contact with grease, oil, and most chemicals.

making closure strips used with corrugated metal sheet. It is also effective as thermal insulation in specialty low-temperature applications such as pipe, tanks and heat exchangers.

When applied as an underlayment in various floor systems, ETHAFOAM reduces noise.

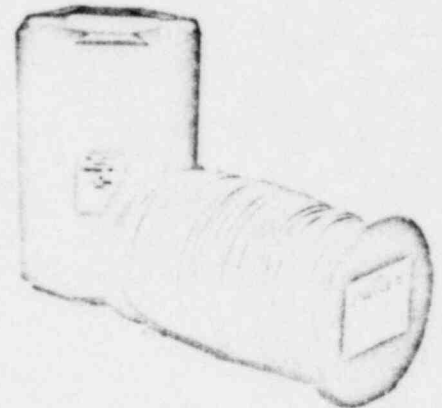
In dam construction, its toughness, water resistance, ability to insulate and light weight have established ETHAFOAM 221 and ETHAFOAM 222 plastic foam sheet as an ideal concrete-curing blanket.

Construction

ETHAFOAM brand polyethylene foam is the logical choice for many construction applications because of its insulating value, flexibility and compressibility. For certain other applications, its resistance to water or chemicals or to the adverse effects of temperature variation are factors of prime importance.

In building construction, the shape and depth of sealants applied to joints can be controlled with resilient, non-staining backer rods of ETHAFOAM SB.

ETHAFOAM is an excellent nonporous, flexible material for



Properties And Characteristics*

Compression Characteristics

Compressive strength

Compressive strength is a measure of the firmness of the product or the amount of compression it experiences under a given load. Typical compressive strength values for various ETHAFOAM brand products at different percentages of deflection are shown in Figure 1 and Table 1.

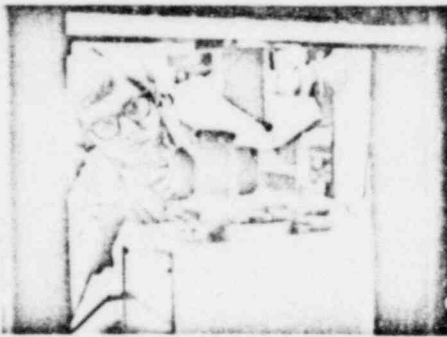
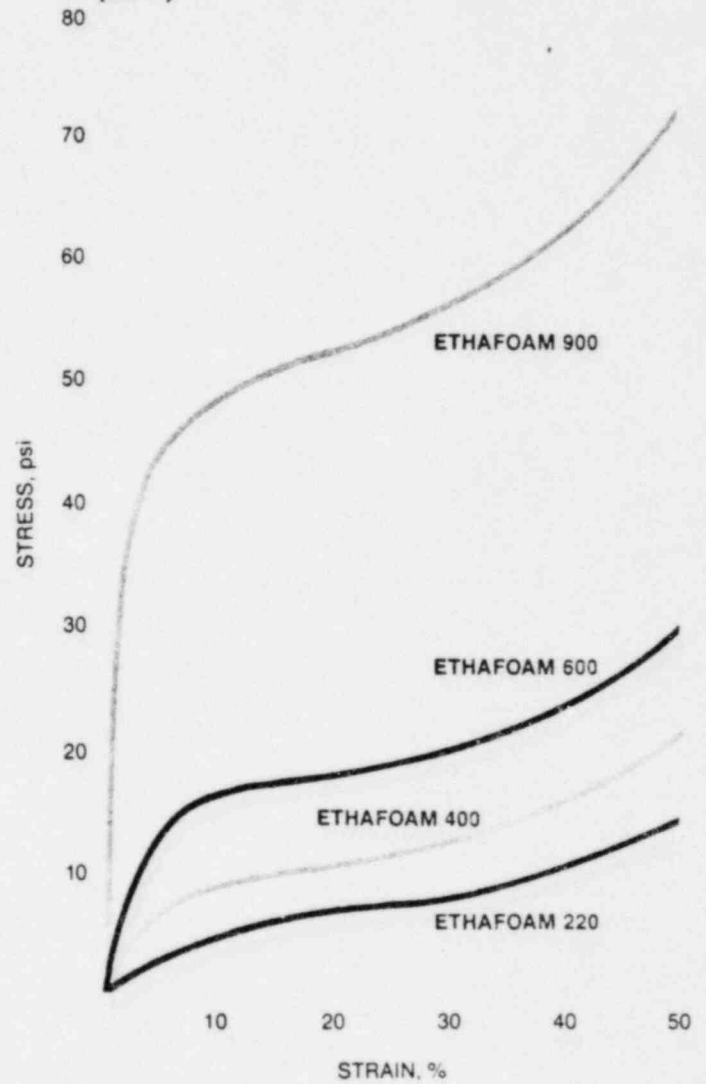


FIGURE 1 — Typical Compressive Strength Curves for Expanded Polyethylene Plank (73°F.)*



*Not to be considered specification values.

TABLE 1—Typical Properties* of ETHAFOAM Products

		ETHAFOAM 200	ETHAFOAM 220	ETHAFOAM 250	ETHAFOAM 300	ETHAFOAM 350	ETHAFOAM 400
Density (PCF)	ASTM D 3575 Test C	1.6	2.7	2.2	4	6	9
Cell Size (MM)	ASTM D 3576	1/8"=2.0	1/8"=1.0				
	Dow Modified	1/4"=2.2 1/2"=2.4	1/4"=1.2 1/2"=1.4	1.3	1.2	1.1	1.0
Compressive Strength (PSI)	ASTM D 3575 Test B						
	at 5%	—	2.5	3.0	6.5	15	44
	at 10%	—	3	5	8.5	17	48
	at 25%	3	6	8	12	20	54
	at 50%	—	15	15	21	33	75
Compressive Creep (% Deflection)	ASTM D 3575 Test BB						
	Loaded @ specified PSI static load for 1000 hrs.						
	75°F	—	—	<5 @ 1.5 PSI	<5 @ 3 PSI	<5 @ 4 PSI	<4 @ 20 PSI
	160°F	—	—	<5 @ .25 PSI	<6 @ 1 PSI	<5 @ 2 PSI	<7 @ 7 PSI
Tensile Strength (PSI)	ASTM D 3575 Test E	60	100	50	65	80	130
Tensile Elongation (%)	ASTM D 3575 Test E	60	80	60	70	70	100
Tear Strength (lb./in.)	ASTM D 3575 Test D	—	30	15	25	30	50
Flexural Modulus (PSI)	ASTM D-790	—	—	300	500	700	1100
Buoyancy (PCF)	ASTM D 3575 Test AA	55-60	55-60	55-60	55-60	53-58	50-55
Thermal Conductivity (BTU— in./hr. ft. ² °F) at 75°F mean temperature ..		1/8"=0.3					
	ASTM D 3575 Test EE	1/4"=0.4	1/4"=0.3	0.4	0.4	0.4	0.4
	Method B	1/2"=0.4	1/2"=0.4				
Thermal Stability (% Shrinkage)	ASTM D 3575 Test F						
	Conditioned @ specified temp. with no load:						
	24 hrs	—	-1.0 @ 165°F	-10 @ 165°F	-1.0 @ 165°F	-1.0 @ 165°F	-1.0 @ 165°F
	48 hrs	—	-1.5 @ 165°F	-2.4 @ 165°F	-1.5 @ 165°F	-1.0 @ 165°F	-1.0 @ 165°F

*Not to be considered specification values.

FIGURE 2 — Typical Compressive Creep of ETHAFOAM 220 Plastic Foam 4" x 4" x 2" Sample*

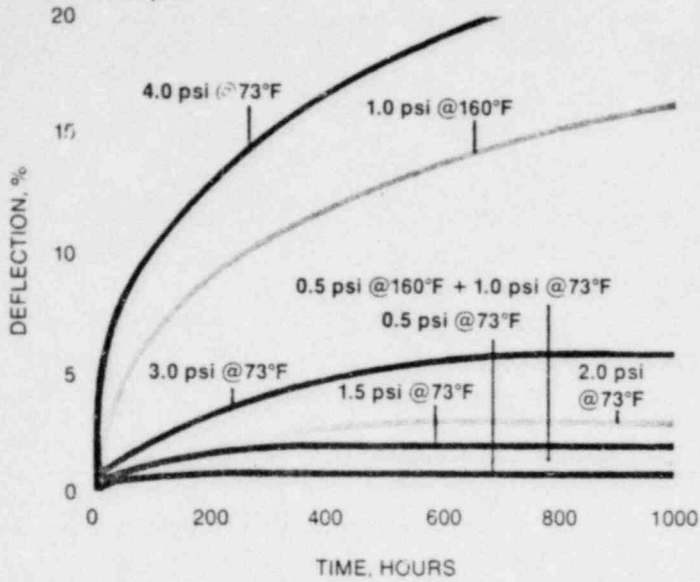


FIGURE 3 — Typical Compressive Creep of ETHAFOAM 400 4" x 4" x 2" Sample*

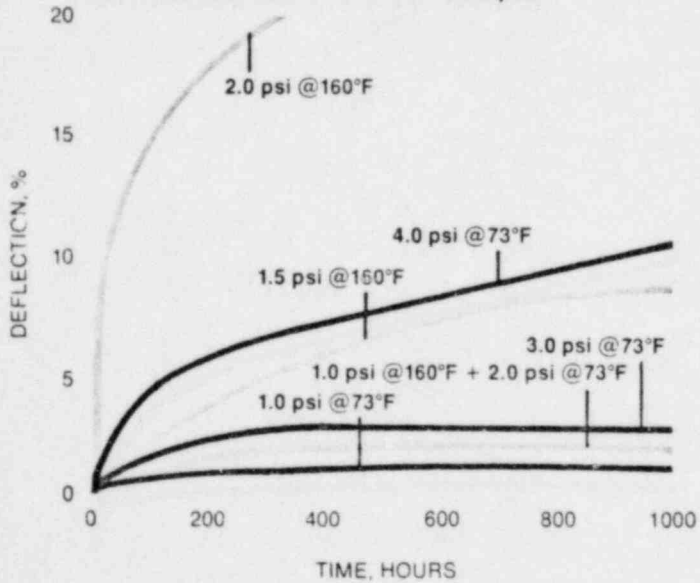


FIGURE 4 — Typical Compressive Creep of ETHAFOAM 600 Plastic Foam 4" x 4" x 4" Sample*

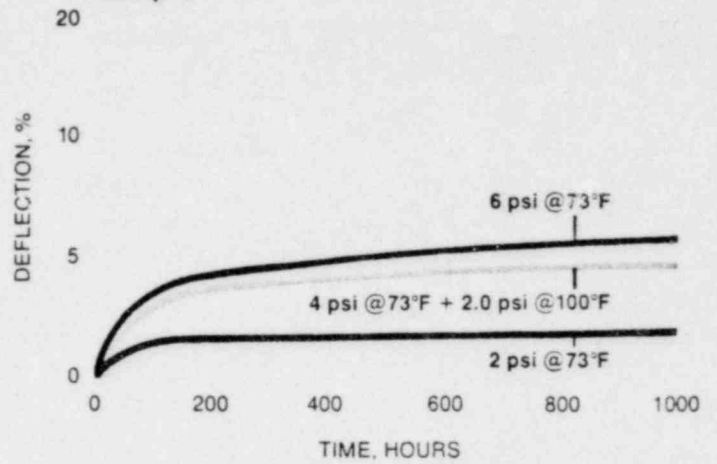
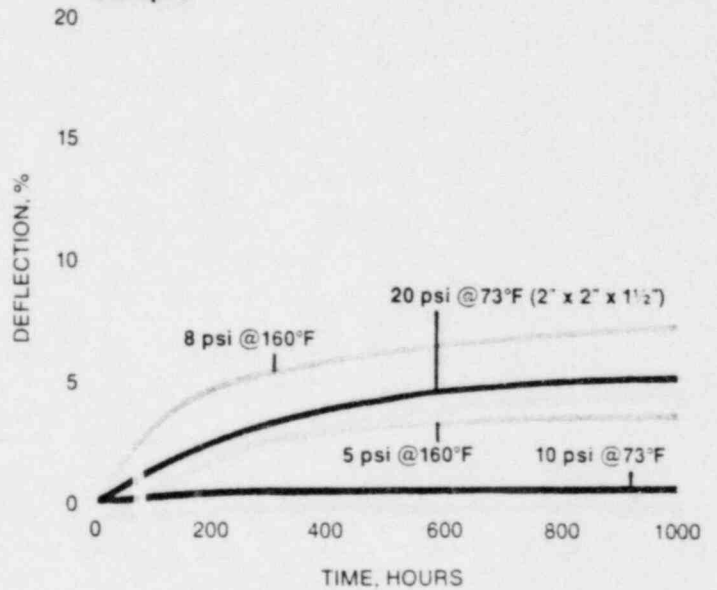


FIGURE 5 — Typical Compressive Creep of ETHAFOAM 900 Plastic Foam 4" x 4" x 2" Sample*

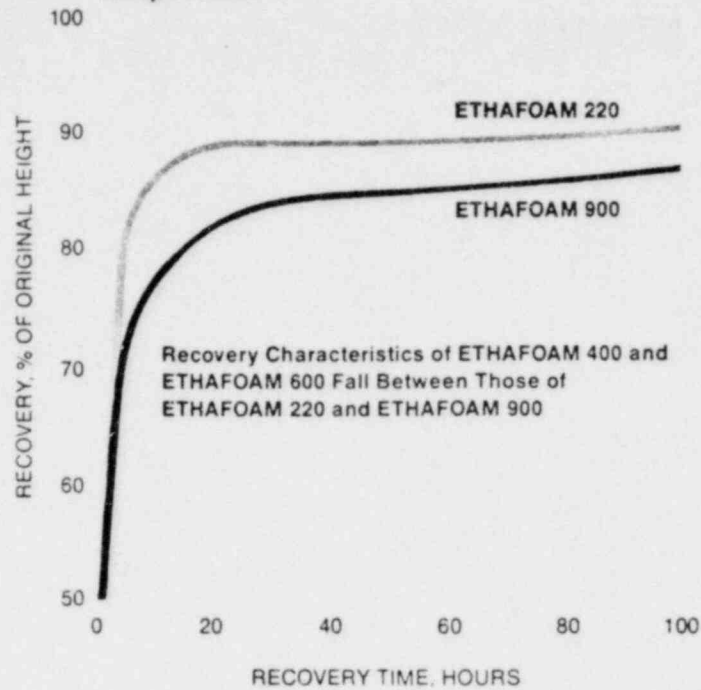


Compressive creep When low-density materials are loaded continuously over a period of time, they tend to "creep"; that is, they lose some of their original thickness. Creep characteristics, which are determined by the variables of load, time and temperature reflect the long-term load-carrying ability of a material and affect its cushioning ability. Typical compressive creep characteristics of ETHAFOAM brand products are shown in Figures 2 through 5.

Compressive Set-Recovery

Compressive set is a measure of the thickness a material fails to recover after it has undergone compressive creep and the load has been removed. To express this relationship in another way, percent recovery plus percent compression set equals 100 percent. Figure 6 shows the recovery characteristics of ETHAFOAM 900 and ETHAFOAM 220 after compression to 50% of their original thickness for 22 hours at room temperature.

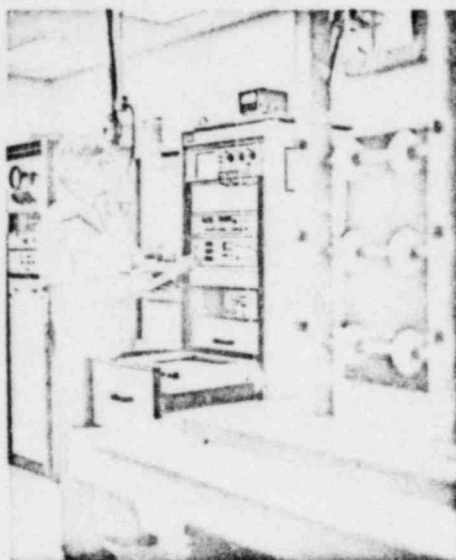
FIGURE 6 — Typical Recovery of ETHAFOAM (At Room Temperature) After 50% Compression for 22 Hours @ Room Temperature*



* Not to be considered specification values.

Tensile and Tear Strength

ETHAFOAM brand plastic foam is a tough material. Typical tensile and tear properties for ETHAFOAM brand products are listed in Table 1, page 7.



Thermal Properties

Thermal Conductivity The amount of heat transmitted through a foam is determined by many factors, including the nature of the base material, the cell size and the degree of closed-cell structure. Permanence of good thermal insulating value often depends on the ability of the foam to resist the effects of water and moisture vapor. Typical conductivity values are listed in Table 1, page 7.

Thermal Stability The thermal stability of ETHAFOAM plastic foam is suitable for most applications. Without load, ETHAFOAM products are used successfully in many applications in which they are exposed intermittently to temperatures up to 180°F. Typical dimensional stability data are given in Table 1. Like other thermoplastic materials, polyethylene foam becomes more flexible at high temperatures and more rigid at low temperatures.

Water Resistance

Water Absorption Even when it is totally immersed, ETHAFOAM plastic foam absorbs a negligible amount of water. In fact, essentially all the water pickup measured in immersion tests is accounted for by the open cut cells on the surface (see Table 1).

Water Vapor Transmission

ETHAFOAM plastic foam provides an excellent barrier to the transmission or penetration of water vapor. ETHAFOAM products have perm ratings of less than 0.4 perm-inch, as shown in Table 1. When a skin is left on the surface, the water vapor transmission can drop to 0.2 perm-inch.

Buoyancy The light weight and excellent water and water vapor resistance of ETHAFOAM brand plastic foam make it an excellent material for flotation applications. See Table 1 and page 4.

WARNING — ETHAFOAM brand polyethylene foam is combustible and can be ignited by contact with a flame. Therefore it should not be exposed to flames or other ignition sources. The burning characteristics of ETHAFOAM vary significantly depending on the amount of material present and other combustion conditions. Large quantities of ETHAFOAM, as might be found in storage, can burn rapidly and produce dense smoke. Under normal combustion conditions, carbon monoxide is generated. Additional toxic substances may be released under less than full combustion conditions. In firefighting situations, dense smoke should be avoided and respirators used.

Chemical-Solvent Resistance

ETHAFOAM plastic foam is resistant to solvents and most other chemicals at room temperature and contains no water-soluble constituents. As a buoyancy material in water, it is unaffected by contact with fuel oil and other hydrocarbons. However, if the foam is immersed for a long period of time in certain solvents, including gasoline, it will swell, absorb solvent, and lose strength. Also, at elevated temperatures the foam becomes more susceptible to attack by certain solvents. Acids and alkalis normally have no effect on ETHAFOAM polyethylene foam, but strong oxidizing agents may cause degradation, especially at high temperatures. No universal solvent for ETHAFOAM plastic foam is known.

Light Stability

Extended exposure to sunlight causes degradation of ETHAFOAM. This degradation is first noted in a yellowing of the surface. After longer periods of exposure, some loss of physical properties will occur. The degree of aging depends on the climate. For example, three months of exposure to the summer sun in Arizona will produce some degradation of ETHAFOAM, while exposure to sunlight for a whole year in Michigan has very little effect.

For applications under direct sunlight, where long-term performance is required, a protective coating should be applied over the foam. The surface of polyethylene is not naturally receptive to coatings, but flame treatments and other treatments, such as those used on

polyethylene film to improve the acceptance of paints and inks, can be used to prepare ETHAFOAM for the application of coatings and adhesives. Coatings such as vinyl, to protect foam against ultraviolet degradation, increase toughness and weather resistance or to create decorative effects can be applied to ETHAFOAM. Such coatings are available from:

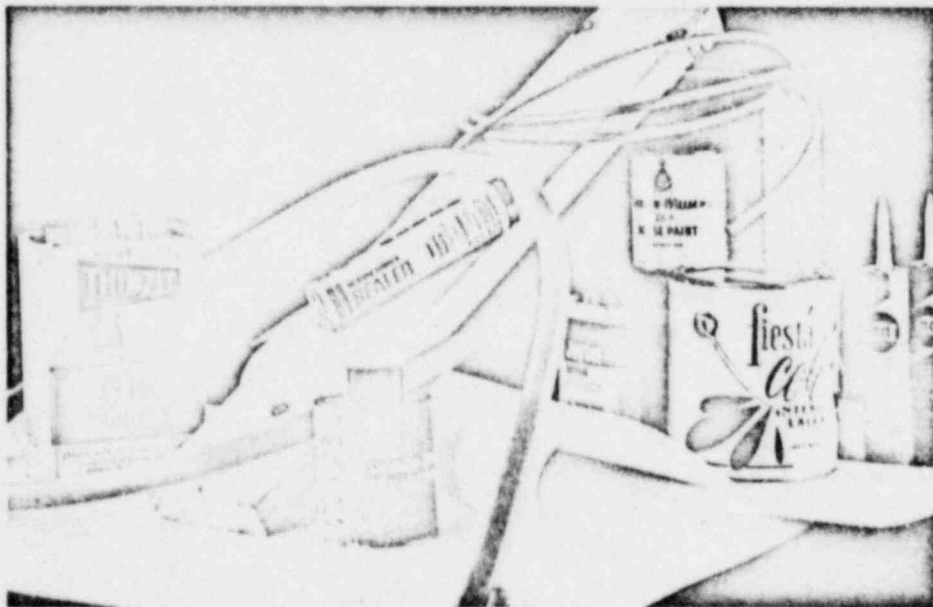
Flexabar Corp.
140 Walnut Street
Northvale, NJ 07647

United Coatings
1130 E. Sprague
Spokane, WA 99202

Bee Chemical Co.
Lansing, IL 60438

Food and Drug Packaging Compatibility

Some forms of ETHAFOAM 220, ETHAFOAM 221, ETHAFOAM 222, ETHAFOAM 400, ETHAFOAM 600, and ETHAFOAM 900 expanded polyethylene foam comply with the Food Amendment of the U.S. Food Drug and Cosmetic Act when used unmodified and according to good manufacturing practices for food packaging applications. Contact your nearest Dow sales office for further information.



Fabrication

Ease of fabrication is an important advantage of ETHAFOAM brand plastic foam. It can be skived to precise thickness, cut and shaped to form custom parts, and joined to itself or other materials without a major investment in equipment.

Authorized Fabricators

Generally, the most economical, quickest and most reliable way to secure special shapes or custom parts made from ETHAFOAM is to employ the services of an authorized fabricator. Fabricators are equipped not only with conventional tools, but also with special devices, such as machines with blades or bits that operate with a slicing action, splitters with scalloped blades that cut up to 50 feet per minute without producing dust, electrically heated resistance wires, and contoured heated molds. For names and addresses of fabricators please consult the nearest Dow sales office.

Bonding

ETHAFOAM plastic foam can be adhered to itself by use of heat alone. The two surfaces are heated simultaneously with hot air or by use of a plate heated to about 350°F. When the surface of the foam begins to soften, the pieces are quickly joined with moderate pressure. Only a short cooling time is necessary. This simple method produces an excellent durable bond. The surface of the heating plate can be coated with Teflon fluorocarbon resin or Dow Corning 1890 protective sealer to facilitate release of the melted foam.

Note When heat is used to cut or form ETHAFOAM plastic foam, adequate ventilation must be provided to carry smoke or fumes away from the breathing zone of workmen.

CAUTION When large quantities of ETHAFOAM plastic foam are stored or fabricated, small quantities of the blowing agent released from the foam may tend to accelerate corrosion of heaters and boilers. Corrosion can eventually create holes in the combustion chamber, leading to the release of combustion gases and carbon monoxide, which would endanger the health of persons in the area. Heating equipment should be inspected regularly during every heating season to check for pinholes or larger defects in the combustion chamber.

Further information on ETHAFOAM products is available from the Dow Sales Offices listed on the back cover of this bulletin. For more detailed information on the use of ETHAFOAM in packaging, ask for Form No. 172-221, Designing Packages to Survive Shipping and Handling with ETHAFOAM Brand Polyethylene Foam.

TABLE 2 — Adhesives for ETHAFOAM

1. Contact Adhesive	Structural Applications Requiring High Bond Strength Foam To Foam, Wood, Metal etc. Packaging Applications	Pliogrip AD965; Neolite all purpose adhesive Polyfoamstix 1579 Bondmaster G-590 Scotch Grip Industrial Adhesive 4693 or 4729 Armstrong 520	Goodyear Tire & Rubber Company Akron, Ohio Adhesive Products Corporation Bronx, New York PPG Bloomfield, N.J. 3M Company St. Paul, Minnesota Armstrong Cork Company Lancaster, Pennsylvania
2. Pressure Sensitive	Gaskets & Sealer Strips	Dri Tac 1519	Adhesive Products Corporation Bronx, New York
3. Double Faced Self-Adhesive Films	Gaskets & Sealer Strips	Fox Film S-510 Series	Morgan Adhesives Company Stow, Ohio Fasson Products Painesville, Ohio
4. Release Paper	Used with Press Sensitive Adhesives		Riegel Paper Company N.Y., N.Y.
5. Hot Melt	See Contacts Easy to apply Short open time	Thermogrip 212; Thermogrip 1398 Hot Grip	USM Corporation Cambridge, Massachusetts Adhesive Products Corporation Bronx, New York

NOTE: Adhesive roll coaters available from Black Bros. Co. Inc., Mendota, Illinois.
Hot Melt applicators available from USM Corporation.

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 23: Masonry Walls

The NRC provided required action for the masonry walls at Byron/Braidwood Stations in a letter from B. J. Youngblood to L. O. Del George of Commonwealth Edison Company, dated November 30, 1981. The following required action was requested:

- a) All unbuilt walls will be reinforced concrete or reinforced masonry designed to comply with SEB Masonry Wall Criteria, Revision 1. A list of unbuilt walls as of October 20, 1981 should be provided.
- b) Perform a systematic mapping of structural cracks on all safety related walls and discuss safety implications of any existing cracks and possible disposition.
- c) Confirm in a letter stating full conformance to QA requirements of Appendix B to CFR Part 50. Actual audit by NRC QA Branch may be necessary for key walls.
- d) Provide calculations for each type of Category I masonry walls for staff's review to verify conformance with the intent of "SEB Criteria for Safety-Related Masonry Wall Evaluation" July 1981, Revision 1.
- e) Provide detailed assessment of behavior of walls assuming existence of reasonable crack patterns for the safety related walls for the staff's review and acceptance. The assessment should be realistic and adequate in describing actual wall behavior under SSE shaking. The assessment should utilize any existing test data pertinent. Pending staff's review, need for additional analysis, limited tests, and fixes may arise to demonstrate the adequacy of walls. The CEC OL should not be contingent upon completion of any additional analysis testing or fixes. This will be treated as a license condition and any remedial action (testing and fixes) will have to be completed to the satisfaction of the NRC staff prior to the resumption of power operation after the first refueling.

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Response:

- a) Enclosure 1 lists the masonry walls not built at Byron Station as of October 20, 1981. Enclosure 2 lists the masonry walls not built at Braidwood Station as of October 20, 1981. All of these walls will be reinforced concrete or reinforced masonry in accordance with "SEB Interim Criteria for Safety-Related Masonry Walls Evaluation", Revision 1.
- b) At this time there are no masonry walls with structural cracks at either Byron Station or Braidwood Station. If structural cracks appear in safety-related masonry walls in the future, an inspection program will be initiated to survey these walls and map any structural cracks.
- c) The design drawings and specifications for safety-related masonry walls require all work to be performed and inspected in accordance with an approved quality assurance program that is in conformance with the requirements of Appendix B to 10 CFR Part 50. Specific QA/QC documentation is available for NRC review at Byron Station and Braidwood Station.
- d) Calculations for the following three representative walls are attached as Enclosure 3 for the staff's review:
 1. Wall 4A-88, Elevation 392'-6", Auxiliary Building
 2. Wall 7A-19, Elevation 439'-0", Auxiliary Building
 3. Wall 5A-134, Elevation 401'-0", Fuel Handling Building

These walls, and all other safety-related walls for Byron/Braidwood Stations, have been designed in accordance with the requirements of NCMA-1974 and the load and load combinations as described in Table C of the calculations. There are no significant deviations between NCMA-1974 and the "SEB Criteria for Safety Related Masonry Wall Evaluation", Revision 1, July 1981.

- e) Commonwealth Edison Company has reviewed the data obtained from the Shaking Table Study of Single Story Masonry Houses performed at the Earthquake Engineering Research Center (1) at the University of California, Berkeley, to determine crack patterns under dynamic

loading. The test results for vertically spanning non-load bearing walls subject to out-of-plane loads, as summarized in Reference 1, indicates that a horizontal crack develops in the top third height for walls without openings. This horizontal crack is attributed to tension in the wall perpendicular to the bed joint due to the vertical wall moment. For walls with openings, diagonal and/or horizontal cracks were initiated either near the top or bottom corner of the opening. It should be noted that horizontally spanning walls were not tested at Berkeley. The vertically spanning non-load bearing walls subject to in-plane loads developed horizontal cracks near the base of the wall and horizontal and/or diagonal cracks at the ends of the door and window lintels. This cracking was associated with the rigid body rocking of the wall.

All safety related concrete masonry walls at Byron/Braidwood have been designed to span horizontally. It is the opinion of Commonwealth Edison Company that horizontal cracks would not develop under dynamic conditions due to out-of-plane loading, since the stresses perpendicular to the bed joint will always be less than one-half those parallel to the bed joint. However, if a horizontal crack is postulated in the safety related walls at Byron/Braidwood, the walls will retain their integrity, since the walls are designed to span horizontally. Should a diagonal crack occur locally adjacent to an opening, the joint reinforcement which is provided in alternate masonry courses would prevent crack propagation, and would assure the structural integrity of the wall for out-of-plane loads.

The dynamic test data indicates that horizontal or diagonal cracks may form due to in-plane loading on the masonry walls. Diagonal cracks may form locally at the openings. However, the joint reinforcement provided in alternate masonry courses would prevent crack propagation, and would assure structural integrity of the walls. It is the opinion of Commonwealth Edison Company, therefore, that only horizontal crack formation may have an effect on structural integrity of masonry walls used at the Byron/Braidwood Stations. However with the presence of a horizontal crack due to in-plane loads, the behavior of the wall remains unchanged from the initial design, and the wall will retain its structural integrity to carry out-of-plane loads, since the walls are designed to span horizontally.

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LIST OF REFERENCES

1. Clough, R. W., Mayes, R. L, and Gulkan, P., "Shaking Table Study of Single-Story Masonry Houses", Volume 3: Summary, Condlusions and Recommendations, Earthquake and Engineering Research Center, Report No. UCB/EERC-79/25, September, 1979.

Byron Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-208	346'-0"	25	N.1 & N.8
		N	N.24 & 24.8
		24.8	N & P.9
		24.5	P.8 & Q
		Q.1	24 & 24.5
		S.5	19 & 19.6
A-210	346'-0"	19	S.5 & Q
		Q.8	19 & 19.6
A-211	346'-0"	S.7	21 & 23
A-212	343'-0"	U.5	21 & 23.4
		X.5	21.5 & 23.5
A-213	354'-0"	X.6	21.1 & 21.5
		15	U.4 & U.5
A-220	364'-0"	U.4	14.9 & 15.3
		15.3	U & U.4
		U	15.5 & 15.6
		U.2	15.5 & 15.6
		15.5	U & U.2
		20.3	U & U.2
		U	20 & 20.3
		U.2	20 & 20.3
		21	U.4 & U.5
		U.4	21.1 & 20.8
		20.8	U & U.4
		21.8	N.9 & P.5
		N.9	21.8 & 23.5
		24	N.5 & N.9
		N.5	24 & 24.8
25	N.9 & N.5		
N.5	25 & 25.6		
25	L & L.2		
25.3	L & L.2		
A-223	364'-0"	L.2	25 & 25.3
		19	Q & Q.9
		19.6	Q.8 & 5.3
		19	S.5 & U
		19	U & V
		U.2	21 & 18.6
		18.6	S.9 & U.5
		18.6	U.9 & V.3
		19.1	V.3 & V.5
19.6	V.3 & W		
A-225	364'-0"	Y	12.5 & 13.8
		Y	21.2 & 18.5

Byron Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>			
		<u>On</u>	<u>Between</u>		
A-229	383'-0"	L.5	10 & 10.6		
		L.4	10.6 & 10.8		
		11.3	L.5 & Q.1		
		12.1	N & P.9		
		13	M.1 & P.6		
		14	M.5 & P.5		
		P.5	14 & 14.5		
		15	P.2 & P.6		
		14.8	M & N		
		A-230	383'-0"	18.8	L & M.1
M.1	18.8 & 20				
M.1	20 & 21				
20.1	L.8 & M				
L.8	20.1 & 21				
24	P & P.3				
P.3	24 & 24.2				
24.2	P.3 & P.5				
19.8	V.7 & V.9				
A-232	383'-0"	U.5	19.2 & 21		
		19	S.5 & S.9		
		S.5	19 & 19.8		
		19.8	Q.8 & S.5		
		S.2	19 & 19.8		
		A-233	391'-0"	12	N & Q
				13	M & Q
A-234	392'-6" 394'-6" 375'-6"			U	15 & 17
		Q	15 & 21		
		Q	15 & 21		
A-239	401'-0"	N	23 & 24		
		23.8	N.7 & P.8		
		24.8	N.7 & P.8		
		25.2	N.7 & P.8		
A-242	401'-0"	V.2	15 & 15.5		
		15.5	U.8 & V.2		
		V.2	20.5 & 21		
		20.5	U.8 & V.2		
		17.4	Q & V.8		
		18.6	Q & V.9		
		19	Q & V.2		
Q	16.5 & 16.9				

Byron Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-242	401'-0"	16.5	Q & Q.4
		Q	19.1 & 19.5
		19.5	Q & Q.4
		Q.1	15 & 15.5
		Q.5	15 & 15.5
		15.5	Q.1 & Q.5
		Q.1	20.5 & 21
		Q.5	20.5 & 21
		20.5	Q.1 & Q.5
A-243	401'-0"	Q.1	21 & 20.8
		Q.5	21 & 20.8
		20.8	Q.1 & Q.5
		23	Q.1 & S
		S	23 & 24.5
		23.4	Q.1 & Q.2
		23.6	Q.1 & Q.2
		Q.2	23.4 & 23.6
A-246	409'-6"	17.4	Q & V.9
		18.6	Q & V.9
		P.5	26.4 & 26.6
		26.4	L.5 & P.5
		L.5	26.8 & 28.8
		28.8	L.1 & L.5
A-249	414'-0"	22.2	U.6 & V.9
		U.6	22.2 & 23.8
A-251	414'-0"	L.5	24.3 & 24.6
		L.8	24.2 & 24.5
		24.5	L & N
A-254	426'-0"	N	23.1 & 23.9
		M	24 & 24.3
		L.9	24 & 24.5
		24.3	L.9 & M
A-257	426'-0"	U.8	16.2 & 17.1
		17.1	U.4 & U.8
		19	Q.1 & S.2
		S.2	19 & 19.9
		19.9	Q.1 & S
		S	19.9 & 20.8
		16.2	U.4 & U.9

Byron Station
Unbuilt Walls

Enclosure 1

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-267	451'-0"	L.1 11.1	11 & 11.2 L & L.1
A-268	451'-0"	23 P.9 23.5 24.8 L.1	Q & P.8 23 & 23.5 P.9 & Q L & L.1 24.8 & 25
A-270	451'-0"	S	12 & 15
A-272	451'-0"	S	21 & 24
A-277	467'-0"	S	12 & 15
A-279	467'-0"	S	21 & 24

Braidwood Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-206	330'-0"	Q.3	13.1 & 14
		Q.3	21.9 & 23
A-208	346'-0"	22.2	L & L8
		23.2	L & L.4
		Q.1	24 & 24.5
A-209	346'-0"	U.4	12.5 & 15
		S.7	13 & 15
A-211	346'-0"	U.5	21 & 23.4
		S.7	21 & 23
A-212	343'-0"	X.6	14 & 14.7
		X.5	12 & 14
		X.6	21.2 & 21.5
		X.5	21.5 & 23.5
A-219	364'-0"	P	10.5 & 10.9
		10.5	P & P.4
A-220	364'-0"	25	L & L.2
		L.2	25 & 25.3
		25.3	L & L.2
A-225	364'-0"	Y	12.5 & 13.8
		Y	21.2 & 23.5
A-226	355'-4"	L.6	19.3 & 20
		20	L.7 & L.7.5
		L.7.5	20 & 21.9
		21.9	L & L.7
A-229	383'-0"	P.1	10 & 10.6
		L.5	10 & 10.6
		L.4	10.6 & 10.8
		L.6	10.6 & 10.7
		11	N.1 & N.9
		11.3	L.5 & Q.1
		12	L & L.5
		12.1	L.8 & P.9
		12.5	M & N.5
		13	M.1 & P.9
		L.8	13.1 & 14.1
		L.8	15.1 & 15.9
		15.1	L & L.8

Braidwood Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-229	383'-0"	M	13.2 & 14.3
		14.3	M & N
		14	M.2 & P.5
		P.5	14 & 14.5
		15	P.2 & P.6
		18	L.5 & M
A-230	383'-0"	19.9	N.5 & P
		21.9	N & P
		22.9	N & P
		P.1	23.2 & 23.9
		24	N.1 & 0.9
		18.8	L & M1
		18	L.5 & M
		L.8	20.1 & 21
		M.1	18.8 & 20
		M.1	20 & 21
		N	22.9 & 23.9
		N	24.2 & 24.7
		20.1	L.8 & M
A-232	383'-0"	16.5	V.7 & V.9
		19.8	V.7 & V.9
A-233	391'-6"	P.1	10.1 & 10.5
		L.7	10.1 & 10.5
		11.4	L.6 & Q
		12	L.6 & Q
		N.4	11 & 11.4
A-234	375'-6"	Q	17 & 19
	394'-6"	Q	15 & 21
A-237	401'-0"	7.5	N.4 & P.4
		P.5	7.5 & 7.7
		P	7.5 & 7.7
		N.5	7.5 & 7.7
A-239	401'-0"	20	L & M
		23.8	N.7 & P.8
		24.2	L.7 & M
		M	24.2 & 24.5
		24.9	N.7 & P.8
		25.2	N.7 & P.8

Braidwood Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-229	383'-0"	M	13.2 & 14.3
		14.3	M & N
		14	M.2 & P.5
		P.5	14 & 14.5
		15	P.2 & P.6
		18	L.5 & M
A-230	383'-0"	19.9	N.5 & P
		21.9	N & P
		22.9	N & P
		P.1	23.2 & 23.9
		24	N.1 & O.9
		18.8	L & M1
		18	L.5 & M
		L.8	20.1 & 21
		M.1	18.8 & 20
		M.1	20 & 21
		N	22.9 & 23.9
		N	24.2 & 24.7
		20.1	L.8 & M
		A-232	383'-0"
19.8	V.7 & V.9		
A-233	391'-6"	P.1	10.1 & 10.5
		L.7	10.1 & 10.5
		11.4	L.6 & Q
		12	L.8 & Q
		N.4	11 & 11.4
A-234	375'-6"	Q	17 & 19
	394'-6"	Q	15 & 21
A-237	401'-0"	7.5	N.4 & P.4
		P.5	7.5 & 7.7
		P	7.5 & 7.7
		N.5	7.5 & 7.7
A-239	401'-0"	20	L & M
		23.8	N.7 & P.8
		24.2	L.7 & M
		M	24.2 & 24.5
		24.9	N.7 & P.8
		25.2	N.7 & P.8

Braidwood Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-240	401'-0"	28.7 F.5 P N.5	N.5 & P.5 28.3 & 28.7 28.3 & 28.7 28.3 & 28.7
A-241	401'-0"	V	12.5 & 14.7
A-242	401'-0"	Q.1 Q.5 15.5 17 17.4 18.6 19 Q.1 Q.5 20.5 V	15 & 15.5 15 & 15.5 Q.1 & Q.5 S & V Q & V.9 Q & V.9 S & V 20.7 & 21 20.7 & 21 Q.1 & Q.5 21 & 24
A-243	401'-0"	V	21.2 & 23.6
A-247	415'-0"	7.1 L.5 28.9	L & L.5 7.1 & 7.6 L & L.5
A-251	414'-0"	M L.8	24.1 & 24.5 24.3 & 24.7
A-255	426'-0"	28.3	L & L.4
A-256	426'-0"	V 14.9	12 & 15 U.2 & V.1
A-258	426'-0"	V 21.2	21 & 24 U.2 & V.1
A-264	439'-0"	P.6	29.2 & 30
A-268	451'-0"	18.5	P.5 & P.9
A-271	451'-0"	S.3 S.3	15.2 & 17.8 18.2 & 19.8

Braidwood Station
Unbuilt Walls

<u>Drawing</u>	<u>Elevation</u>	<u>Wall Location</u>	
		<u>On</u>	<u>Between</u>
A-274	451'-0"	S	12 & 15
A-275	459'-2"	S	17.7 & 18
		Q.8	17.7 & 18
		S.3	15.1 & 16.7
		S.3	16.7 & 17.8
		20.5	Q & S
		S.3	18.2 & 19.2
		S.3	19.2 & 20.8
		15.3	Q & S
A-276	459'-0"	S	21 & 24
A-277	467'-0"	S	12 & 15
A-278	467'-4"	S.3	16.7 & 17.8
		S.3	15.3 & 16.7
		S.3	18.2 & 19.2
		S.3	19.2 & 20.2
		S.4	20.2 & 20.8
		15.5	Q & S
		17.7	S & Q.9
		18	S & Q.9
		S	17.7 & 18
		Q.9	17.7 & 18
		S	18 & 21
		20.4	Q & S
A-279	467'-0"	S	21 & 24
A-282	477'-0"	17	L & Q
		18	L & N.5
A-283	477'-0"	19	L & Q
		Q	18 & 19
		C	18. & 18.9
A-312	401'-0"	18	Y & Z.5
		18.5	Y & Y.9
		19	Y & Z.5
		Z.6	17.1 & 18.5
		Z.6	18.6 & 20.1
		18.6	Z.6 & Z.9
		Z.9	17.9 & 18.6
		Y.4	20.3 & 20.7
		Y.5	20.5 & 21.
		Y.4	15 & 15.2
		Y.3	15.1 & 15.3
		Z.5	18.3 & 18.7

Design Summary

I. Material Properties

A. Masonry Units

Density: 145 pcf (solid block)
105 pcf (hollow block)

Compressive strength of unit (f_c'): 1800 psi (solid block)
1000 psi (hollow block)

Ultimate Compressive strength of masonry (f'm):
1350 psi (solid & hollow)

B. Mortar: Type M

Compressive strength of mortar (m_o): 2500 psi

II. Allowable Stresses (per NCMA)

A. OBE Load Combination

Flexural tension parallel to bed joint (f_{t_{||}}):
78 psi (solid block)
46 psi (hollow block)

Flexural tension perpendicular to bed joint (f_{t_⊥}):
39 psi (solid block)
23 psi (hollow block)

Shear (f_v): 34 psi (solid & hollow block)

B. SSE Load Combination (overstress factor = 1.67)

Flexural tension parallel to bed joint (f_{t_{||}}):
130 psi (solid block)
77 psi (hollow block)

Flexural tension perpendicular to bed joint (f_{t_⊥}):
65 psi (solid block)
38 psi (hollow block)

Shear (f_v): 57 psi (solid & hollow block)

III. Frequency Calculations (Ref. S&L SD&DD Report #25)

A. Simply supported Walls

$$f = \frac{56}{2 \pi} \sqrt{\frac{EmIm}{144 W_w (Lw)^4}} \quad (\text{cps})$$

B. Cantilivered Walls

$$f = \frac{20}{2\pi} \sqrt{\frac{E_m I_m}{144 W_w (L_w)^4}} \quad (\text{cps})$$

IV. Nomenclature

- A = Net unit area of wall in horizontal section (in²/ft)
- E_m = Modulus of Elasticity = 1000 fm' (psi)
- f_c' = Compressive strength of individual masonry (psi)
- f'_m = Ultimate compressive strength of masonry (psi)
- h_w = Height of wall (ft)
- I_m = Moment of inertia of masonry (in⁴/ft)
- L_w = Horizontal length of wall (ft)
- M_o = Compressive strength of mortar (psi)
- S^o = Section modulus of masonry (in³/ft)
- tw = Thickness of wall (in)
- W_w = Weight of masonry wall (lb/sf sq wall)

V. Load Combinations

Table C

Load Combination	Load				Load Factors							Notes
	D	L	R _o	P _o	T _o	E _o	R _a	T _a	P _a	E _{ss}		
Normal	1.0	1.0	1.0	1.0	1.0							
Normal/ Severe Env.	1.0	1.0	1.0	1.0	1.0	1.0						(1)
Abnormal.	1.0	1.0					1.0	1.0	1.5			
Normal/Ex- treme Env.	1.0	1.0	1.0	1.0	1.0					1.0		(2)
Abnormal/ Severe Env.	1.0	1.0				1.25	1.0	1.0	1.25			
Abnormal/Ex- treme Env.	1.0	1.0					1.0	1.0	1.0	1.0		

Notes

- (1) Governing OBE Load Comb.
- (2) Governing SSE Load Comb.

Controlling Load combinations are normal/severe environmental and normal/extreme environmental.

No walls are subjected to LOCA or SRV loads.

VI. Damping Values

A. Seismic Damping Values For Walls

OBE: 4%

SSE: 7%

B. Seismic Damping Values for Attachments to Walls

OBE: 4%

SSE: 7%

VII. Allowable Stresses for Steel Wall Supports (Columns & Tees)

OBE: AISC Allowable

SSE: 1.6 x (AISC Allowable) not to exceed 0.95 Fy

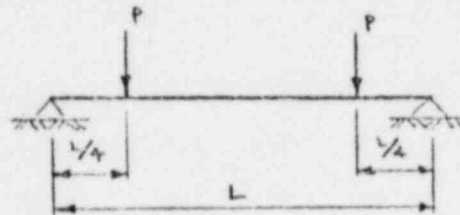
VIII. Attachments to Walls

A. Contingency load to account for 2" ϕ and under control instrumentation piping and 4" ϕ and under electrical conduit:

P = 180 lbs per foot of height for solid block walls

P = 135 lbs per foot of height for hollow block walls

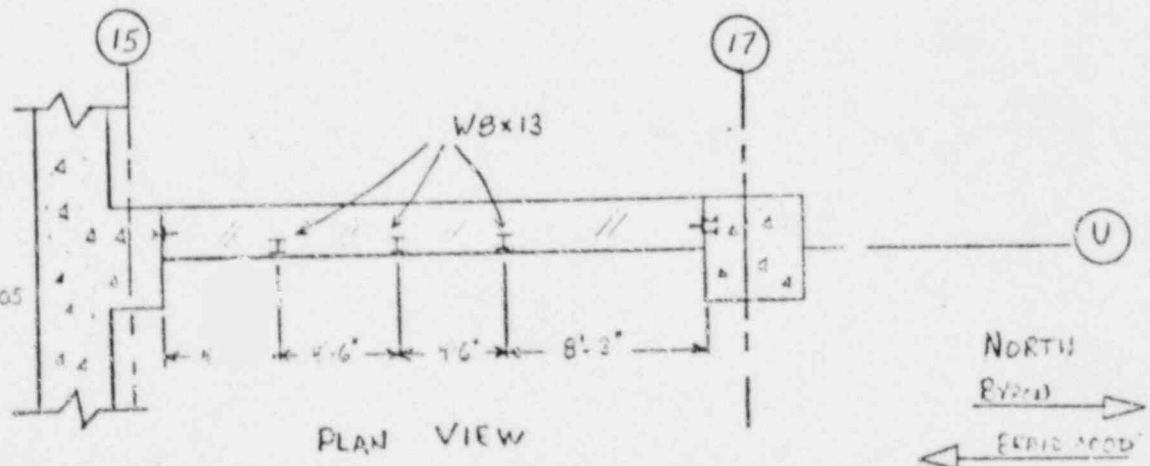
B. For design purposes, consider the load as follows:



This configuration results in maximum moment, and maximum shear at the supports.

C. Use peak seismic acceleration for attachment loads.

WALL NO 4A 88
 FLOOR EL. 372'-6"
 DRAWING NO. A-234, S-705
 ORIENTATION N-S



Wall Parameters

$$\begin{aligned}
 tw &= 1'-11 \frac{5}{8}" \text{ (solid)} \\
 Lw &= 8'-3" \\
 hw &= 6'-2" \\
 \text{Density} &= 145 \text{ pcf} \\
 f'c &= 1800 \text{ psi} \\
 f'm &= 1350 \text{ psi} \\
 A &= (23 \frac{5}{8}")(12") = 283.5 \text{ in}^2/\text{ft} \\
 I &= \frac{(12")(23 \frac{5}{8}")^3}{12} = 13186 \text{ in}^4/\text{ft} \\
 S &= \frac{(12")(23 \frac{5}{8})^2}{6} = 1116.3 \text{ in}^3/\text{ft} \\
 Ww &= \frac{(145 \text{ pcf})(23 \frac{5}{8})}{12 \text{ in/ft}} = 285.5 \text{ psf} \\
 Em &= 1000 f'm = 1.35 \times 10^6 \text{ psi}
 \end{aligned}$$

Considering 1'-0" beam strip from the 8'-3" simply supported span:

$$\begin{aligned}
 f &= \frac{56}{2\pi} \sqrt{\frac{Em I}{144 Ww Lw^4}} \\
 &= \frac{56}{2\pi} \sqrt{\frac{(1.35 \times 10^6)(13186)}{(144)(285.5)(8.25)^4}} = 86.2 \text{ cps}
 \end{aligned}$$

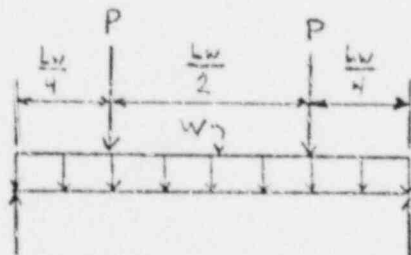
Period, $T = 0.0116$ seconds

"gh" values for wall frequency 86.2 cps $>$ 33 cps, rigid zone.

Use response spectra values for E. 401'-0"

4% Damping (OBE)
7% Damping (SSE)

		Rigid	Peak	Allowable Stress	
Spectra No.				$\frac{Fb}{psi}$	$\frac{Fv}{psi}$
107-OB-EW	0.195	0.53	78	psi	34
107-SS-EW	0.38	1.0	130	psi	57



Simply Supported Span

$$\begin{aligned}
 P &= 180 \text{ lbs/ft ht.} \\
 W &= 285.5 \text{ psf} \\
 Lw &= 8.25 \text{ ft}
 \end{aligned}$$

1. OBE

$$\begin{aligned}
 M_{max} &= (0.195)(1/8)(285.5)(8.25)^2 + (0.53)(1/4)(180)(8.25) \\
 &= 670.4 \text{ ft-lbs}
 \end{aligned}$$

$$V_{max} = (0.195)(1/2)(285.5)(8.25) + (0.53)(180) = 325.1 \text{ lbs}$$

$$f_b = \frac{M}{S} = \frac{(670.4)(12)}{1116.3} = 7.2 \text{ psi} < 78 \text{ psi (OK)}$$

$$f_v = \frac{P}{A} = \frac{325.1}{283.5} = 1.2 \text{ psi} < 34 \text{ psi (OK)}$$

2. SSE

$$M_{\max} = (0.38)(1/8)(285.5)(8.25)^2 + (1.0)(1/4)(180)(8.25) \\ = 1294.3 \text{ ft-lbs}$$

$$V_{\max} = (0.38)(1/2)(285.5)(8.25) + (1.0)(180) \\ = 627.5 \text{ lbs}$$

$$f_b = \frac{M}{S} = \frac{1294.3 \times 12}{1116.3} = 13.9 \text{ psi} < 130 \text{ psi (OK)}$$

$$f_v = \frac{P}{A} = \frac{627.5}{283.5} = 2.2 \text{ psi} < 57 \text{ psi (OK)}$$

CHECK COLUMNS

W8X13 $S_x = 9.91 \text{ in}^3$

Allowable Bending Stress (F_b)

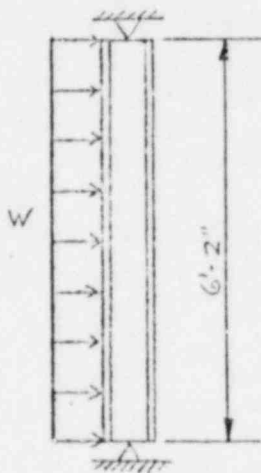
OBE 24ksi

(Continuous Lateral Support)

SSE $0.95F_y = 34.2 \text{ ksi}$

$$W_{obe} = 325.1 + 220.7 = 545.8 \text{ \#/ft}$$

$$W_{sse} = 627.5 + 424.1 = 1051.6 \text{ \#/ft}$$



4.5 Ft. Span $V_{\max} \text{ (OBE)} = (0.195)(1/2)(285.5)(4.5) + 0.53(180) \\ = 220.7 \text{ lbs}$

$$V_{\max} \text{ (SSE)} = (0.38)(1/2)(285.5)(4.5) + 1.0(180) \\ = 424.1 \text{ lbs}$$

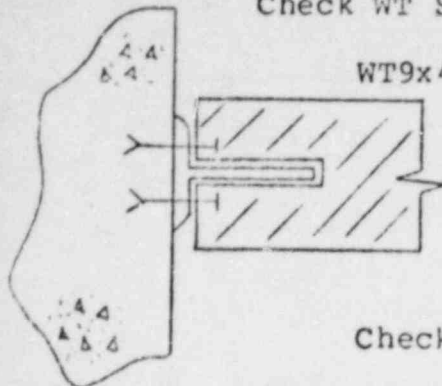
OBE $M_{\max} = 1/8(545.8)(6.167)^2 = 2594.7 \text{ ft-lbs}$

$$f_{b_u} = \frac{2594.7 \times 12}{1000 \times 9.91} = 3.1 \text{ ksi} < 24 \text{ ksi (OK)}$$

SSE $M_{\max} = 1/8(1051.6)(6.167)^2 = 4999.3 \text{ ft-lbs}$

$$f_{b_u} = \frac{4999.3 \times 12}{1000 \times 9.91} = 6.1 \text{ ksi} < 34.2 \text{ ksi (OK)}$$

Check WT Section and Expansion Anchors



WT9x42.5 with 3/4" \emptyset anchors @ 3'-10" O.C
 $A_{STEM} = 12" \times tw = 12 \times 0.526 = 6.3 \text{ in}^2$

OBE $V_{max} = 325.1 \frac{\text{lbs}}{\text{ft.}}$

8'-3" Span

SSE $V_{max} = 627.5 \frac{\text{lbs.}}{\text{ft.}}$

Check Shear on WT Stem, Use 1 Ft. Strip

1. OBE $f_v = \frac{325.1}{6.3 \text{ in}^2} \text{ lbs} = 51.5 \text{ psi} < 14,400 \text{ psi} = 0.4 \text{ Fy (OK)}$
2. SSE $f_v = \frac{627.5}{6.3 \text{ in}^2} \text{ lbs.} = 99.6 \text{ psi} < 23,200 \text{ psi} = 1.6 \times 0.4 \times \text{Fy (OK)}$

Anchor Bolts

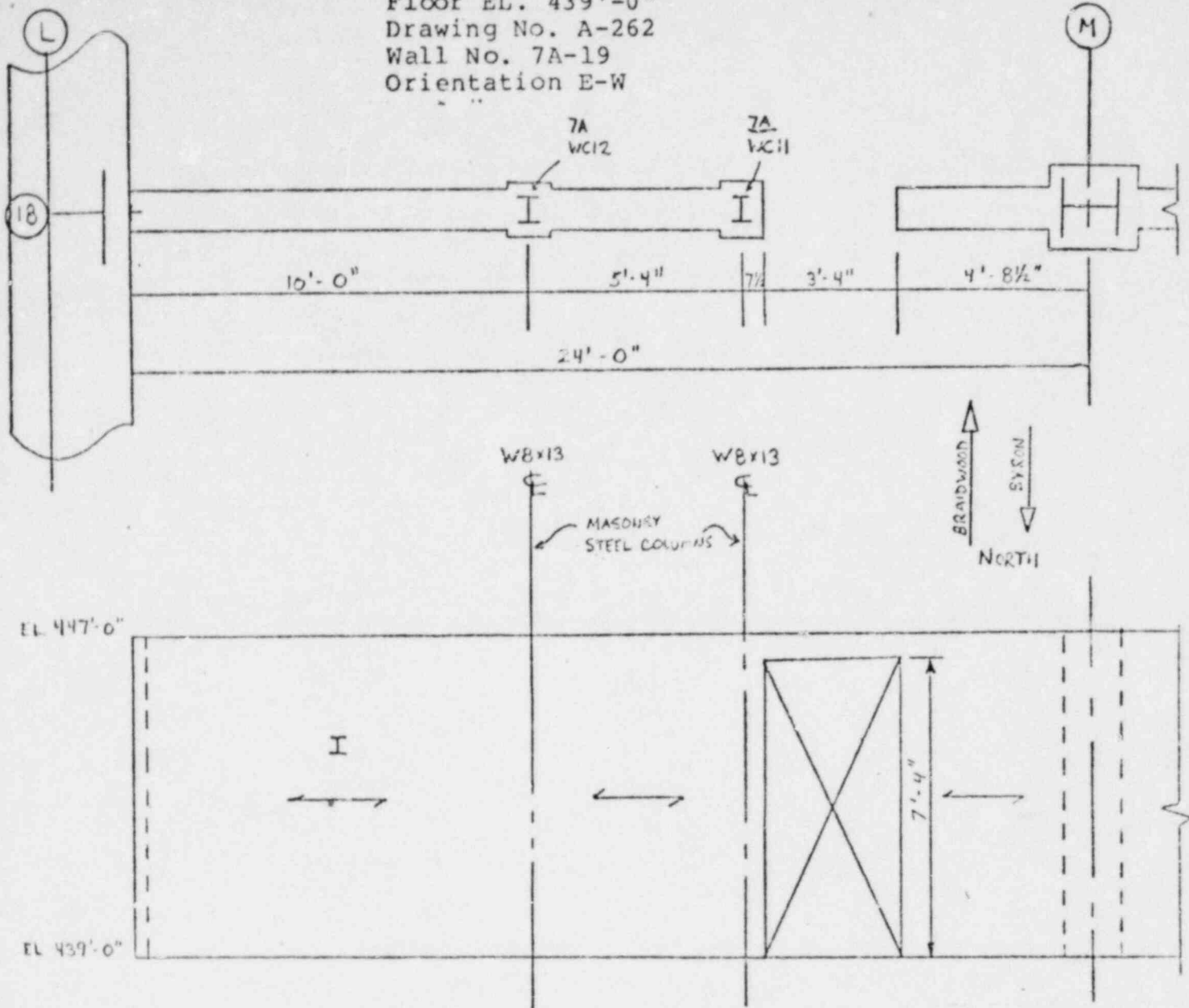
3/4" \emptyset Anchor Allowables

1. OBE $\frac{325.1 \frac{\text{lbs}}{\text{ft}} \times 6.167 \text{ ft}}{4 \text{ Bolts}} = 501.2 \text{ lb/Bolt} < 3,400 \text{ lb (OK)}$
2. SSE $\frac{627.5 \frac{\text{lbs}}{\text{ft}} \times 6.167 \text{ ft}}{4 \text{ Bolts}} = 967.4 \text{ lb/Bolt} < 4,230 \text{ lb (OK)}$

Check Shear on Reduced Wall Area Due to WT Connection (1 ft. Strip)

1. OBE $\frac{325.1 \text{ lb/Ft} \times 1 \text{ Ft}}{1/2(23.625) (12 \text{ in})} = 2.3 \text{ psi} < 34 \text{ psi (OK)}$
2. SSE $\frac{627.5 \text{ lb/Ft} \times 1 \text{ Ft}}{1/2(23.625) (12)} = 4.4 \text{ psi} < 58 \text{ psi (OK)}$

Floor EL. 439'-0"
 Drawing No. A-262
 Wall No. 7A-19
 Orientation E-W



WALL PROPERTIES:

$t_w = 11 \frac{5}{8}"$
 $L_w = 24'-0"$
 $h_w = 8'-0"$
 Density_w = 105 pcf
 $f'_c = 1800 \text{ psi}$
 $f'_m = 1350 \text{ psi}$

Horizontal Walls:

$A = 36.0 \text{ in}^2/\text{ft}$
 $I = 929.4 \text{ in}^4/\text{ft}$
 $S = 159.9 \text{ in}^3/\text{ft}$
 $W_w = 42.6 \text{ psf}$
 $E_m = 1000 f'_m = 1.35 \times 10^6 \text{ psi}$

Section I

Horizontal Simply Supported Span
Using 1'-0" Beam Strip $L_w = 10'-0"$

Controls Analysis

$$f = \frac{56}{2\pi} \sqrt{\frac{EmIm}{144W_w L_w^4}}$$

$$f = \frac{56}{2\pi} \sqrt{\frac{(1.35 \times 10^6)(929.4)}{144(42.6)(10.0)^4}} = 40.3 \text{ cps, } T = 0.025 \text{ sec}$$

 g_H Values from response Spectra @ EL. 439'-0" (Base of Wall)

	RIGID (f 33 CPS)	PEAK	
Spectra No: 109-OB-NS	0.25	1.3	4% damping
Spectra No: 109-SS-NS	0.55	1.7	7% damping

Allowable Stress

 F_b

OBE

46 psi

SSE

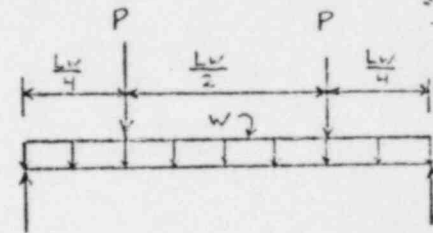
77 psi

 F_v

34 psi

57 psi

$$P = 135 \frac{\text{lbs}}{\text{ft. of ht.}}$$

For Attachment
Load to Hollow
Block Walls

$$W = 42.6 \text{ psf}$$

$$L_w = 10'-0"$$

$$1. \text{ OBE } M_{\max} = \frac{1}{8} (0.25) (42.6) (10)^2 + \frac{1}{4} (1.3) (135) (10) = 571.9 \text{ ft-lbs.}$$

$$f_{b||} = \frac{M}{S} = \frac{571.9 \times 12}{159.9} = 42.9 \text{ psi} < F_{b||} = 46 \text{ psi (OK)}$$

$$V_{\max} = \frac{1}{2} (0.25) (42.6) (10) + 1.3 (135) = 228.8 \text{ lbs.}$$

$$f_v = \frac{228.8}{36} = 6.4 \text{ psi} < 34 \text{ psi (OK)}$$

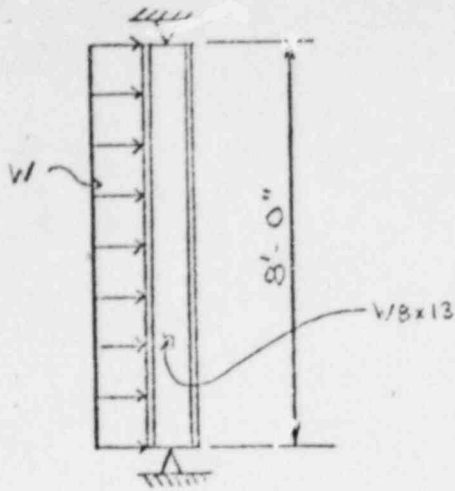
$$2. \text{ SSE } M_{\max} = \frac{1}{8} (0.55) (42.6) (10)^2 + \frac{1}{4} (1.7) (135) (10) = 866.6 \text{ ft-lbs}$$

$$f_{b||} = \frac{866.6 \times 12}{159.9} = 65 \text{ psi} < F_{b||} = 77 \text{ psi (OK)}$$

$$V_{\max} = \frac{1}{2} (0.55) (42.6) (10) + 1.7(135) = 346.7 \text{ lbs.}$$

$$f_v = \frac{346.7}{36} = 9.6 \text{ psi} < 57 \text{ psi (OK)}$$

Check Masonry Steel Columns w/ 10'-0" & 5'-4" Shear Forces Controls Analysis, Worst Loading



$$W = 228.8 + 203.9 = 432.7 \frac{\text{lbs}}{\text{ft}} \quad \text{OBE}$$

$$W = 346.7 + 291.9 = 638.6 \frac{\text{lbs}}{\text{ft}} \quad \text{SSE}$$

$$F_b \text{ OBE} = 24 \text{ ksi AISC (Cont. Lateral Support)}$$

$$F_b \text{ SSE} = 0.95 F_y = 34.2 \text{ ksi}$$

$$5'-4" \text{ Span } V_{\text{max}} = \frac{1}{2} (0.25) (42.6) (5.33) + 1.3 (135) = 203.9 \text{ lbs} \leftarrow \text{OBE}$$

$$V_{\text{max}} = \frac{1}{2} (0.55) (42.6) (5.33) + 1.7 (135) = 291.9 \text{ lbs} \leftarrow \text{SSE}$$

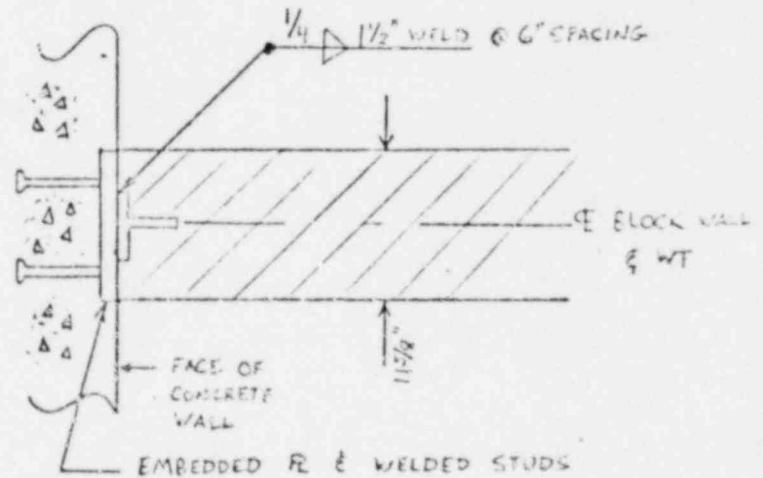
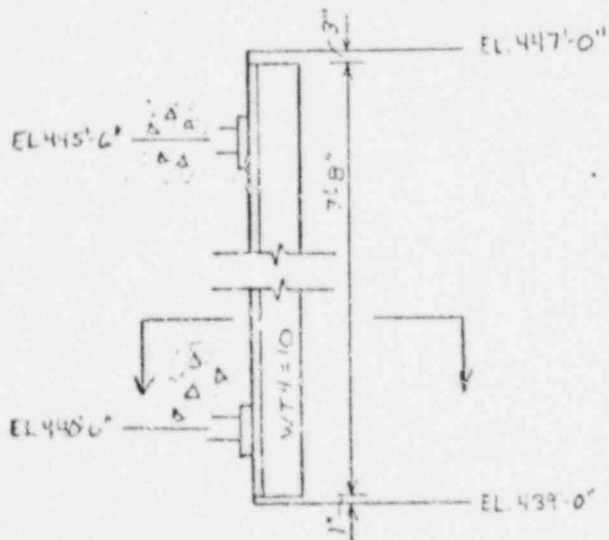
$$1. \quad \text{OBE } M = \frac{1}{8} (432.7) (8)^2 = 3461.6 \text{ lbs. ft.}$$

$$f_b = \frac{3461.6 \times 12}{1000 \times 9.91} = 4.2 \text{ ksi} < F_b = 24 \text{ ksi (OK)}$$

$$2. \quad \text{SSE } M = \frac{1}{8} (638.6) (8)^2 = 5108.8 \text{ ft. lbs.}$$

$$f_b = \frac{5108.8 \times 12}{1000 \times 9.91} = 6.2 \text{ ksi} < 34.2 \text{ ksi (OK)}$$

Check WT4x10 Section



Check Shear on WT4x10 Web

$$A = L_{wr} \times t_{web} = 7.66 \times 12 \times 0.248 = 22.8 \text{ in}^2$$

1. OBE $f_v = \frac{8' \times 0.228 \text{ k/ft}}{22.8 \text{ in}^2} = 0.08 \text{ ksi} < F_v = 14.4 \text{ ksi (OK)}$

2. SSE $f_v = \frac{8' \times 0.3467 \text{ k/ft}}{22.8 \text{ in}^2} = 0.13 \text{ ksi} < F_v = 1.6 \times 14.4 = 23 \text{ ksi (OK)}$

Check Shear Stress on Reduced Wall Area Due to WF Insert

1. OBE $V_{\max} = \frac{228.8 \text{ lbs}}{1/2(36 \text{ in}^2)} = 12.7 \text{ psi} < 34 \text{ psi (OK)}$

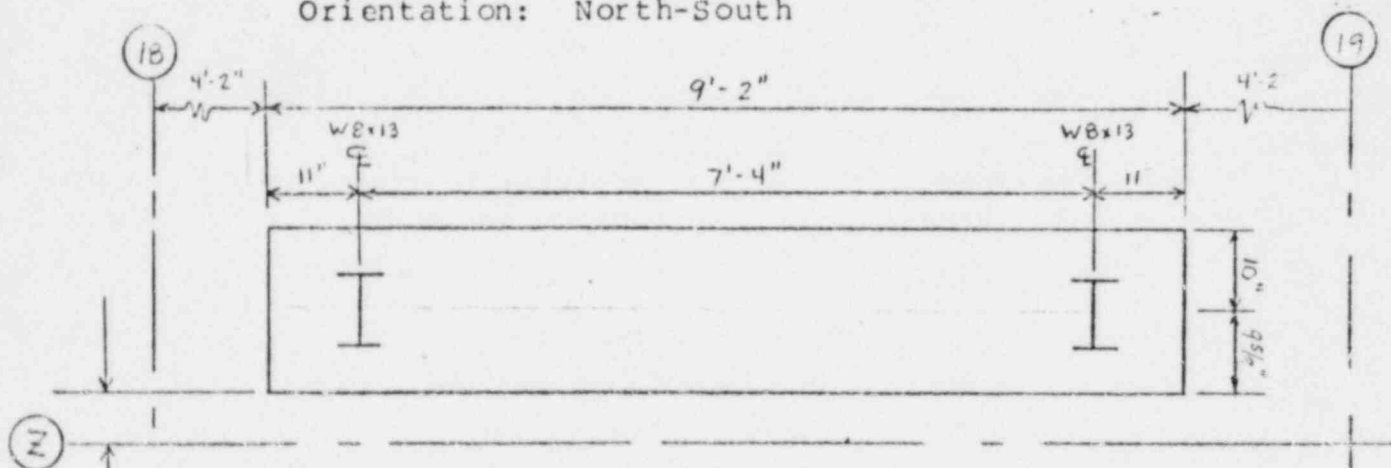
2. SSE $V_{\max} = \frac{346.7 \text{ lbs}}{1/2(36 \text{ in}^2)} = 19.3 \text{ psi} < 57 \text{ psi (OK)}$

Check Weld Capacity 4-1 1/2" Long 1/4" Welds Per Plate

1. OBE $P = 8' \times 0.2288 \text{ k/1} = 1.83 \text{ k} < 0.707 \times 21 \times 1/4 \times 2 \times 4 \times 1.5 = 44.5 \text{ k (OK)}$

2. SSE $P = 8' \times 0.3467 \text{ k/1} = 2.77 \text{ k} < 1.6 \times 44.5 \text{ k} = 71.2 \text{ k (OK)}$

Wall No. 5A-134
 Floor El. 401'-0"
 Drawing No. A-312, S-812
 Orientation: North-South



Wall Properties

$t_w = 1'-7 \frac{5}{8}"$ (Solid)	$A = 19.625 \times 12 = 235.5 \text{ in}^2/\text{ft}$
$L_w = 9'-2"$	$S = \frac{(12)(19.625)^2}{6} = 770.3 \text{ in}^3/\text{ft}$
$h_w = 8'-6"$	$I = \frac{(12)(19.625)^3}{12} = 7558.4 \text{ in}^4/\text{ft}$
Density = 145 pcf	$W_w = (145 \text{ pcf})(19.625 \text{ in})/12 \text{ in/ft} = 237.1 \text{ psf}$
$f'_c = 1800 \text{ psi}$	
$f'_m = 1350 \text{ psi}$	
	$E_m = 1000 f'_m = 1.35 \times 10^6 \text{ psi}$

Consider a 1'-0" Thick Horizontal Beam Strip from the 7'-4" Simply Supported Span and the 11" Cantilever Span

$$f = \frac{56}{2\pi} \sqrt{\frac{EmI}{144W_w L_w^4}} \quad \text{Simply Supported Wall}$$

$$f_{ss} = \frac{56}{2\pi} \sqrt{\frac{(1.35 \times 10^6)(7558.4)}{144(237.1)(7.33)^4}} = 90.7 \text{ cps} \quad T = 0.011 \text{ Seconds}$$

$$f = \frac{20}{2\pi} \sqrt{\frac{EmI}{144W_w L_w^4}} \quad \text{Cantilevered Wall}$$

$$f_{cantilever} = \frac{20}{2\pi} \sqrt{\frac{(1.35 \times 10^6)(7558.4)}{144(237.1)(0.9167)^4}} = 2070.8 \text{ cps}, T = 0.00048 \text{ sec}$$

Both f_{ss} and $f_{cant.}$ are greater than 33 cps therefore both are in the rigid zone. "gH" values are determined from the response spectra of the base of the wall, EL. 401'-0". Peak values are used for attachment loads.

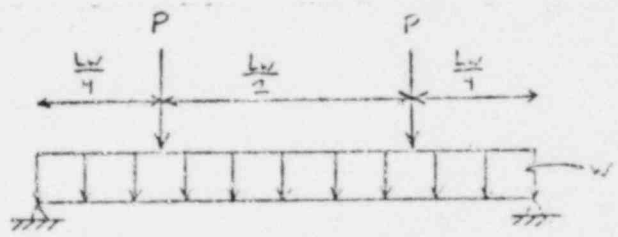
- 4% Damping (OBE)
- 7% Damping (SSE)

		Rigid	Peak	Allowable Stresses F_{bu}	F_v
1	Spectra No: 106-OB-EW	0.28	1.85	78 psi	34 psi
2	Spectra No: 106-SS-EW	0.24	1.2	130 psi	57 psi

Since "gH" values for OBE are larger than gH SSE and OBE has smaller allowable stresses, only OBE need be considered.

Simply Supported Span

$P = 180 \frac{\text{lbs}}{\text{ft. ht.}}$ For Solid Masonry Walls
 $W = 237.1 \text{ psf}$
 $L_w = 7.33 \text{ ft.}$



$$M_{max} = 0.28 \frac{(237.1)(7.33)^2}{8} + \frac{(180)(7.33)(1.85)}{4} = 1056.1 \text{ ft-lbs.}$$

$$f_{bu} = \frac{M}{S} = \frac{(1056.1)(12)}{770.3} = 16.5 \text{ psi} < 78 \text{ psi (OK)}$$

$$V_{max} = 0.28 \frac{(237.1)(7.33)}{2} + (1.85)(180) = 576.3 \text{ lbs}$$

$$f_v = \frac{576.3}{235.5} = 2.45 \text{ psi} < 34 \text{ psi (OK)}$$

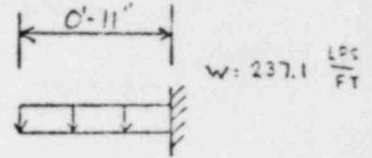
Cantilevered Span

$$M_{\max} = (0.28) (237.1) (1/2) (0.9167)^2 = 27.9 \text{ fts. lbs.}$$

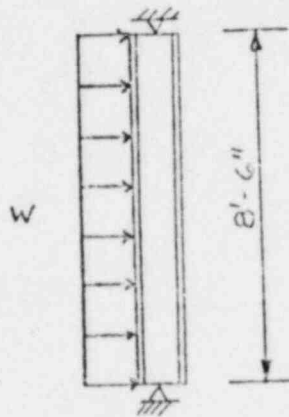
$$V_{\max} = (237.1) (0.9167) (0.28) = 60.9 \text{ lbs.}$$

$$f_{b''} = \frac{27.9 \times 12}{770.3} = 0.43 \text{ psi} < 78 \text{ psi (OK)}$$

$$f_v = \frac{60.9}{235.5} = 0.3 \text{ psi} < 34 \text{ psi (OK)}$$



Check Masonry Steel Columns

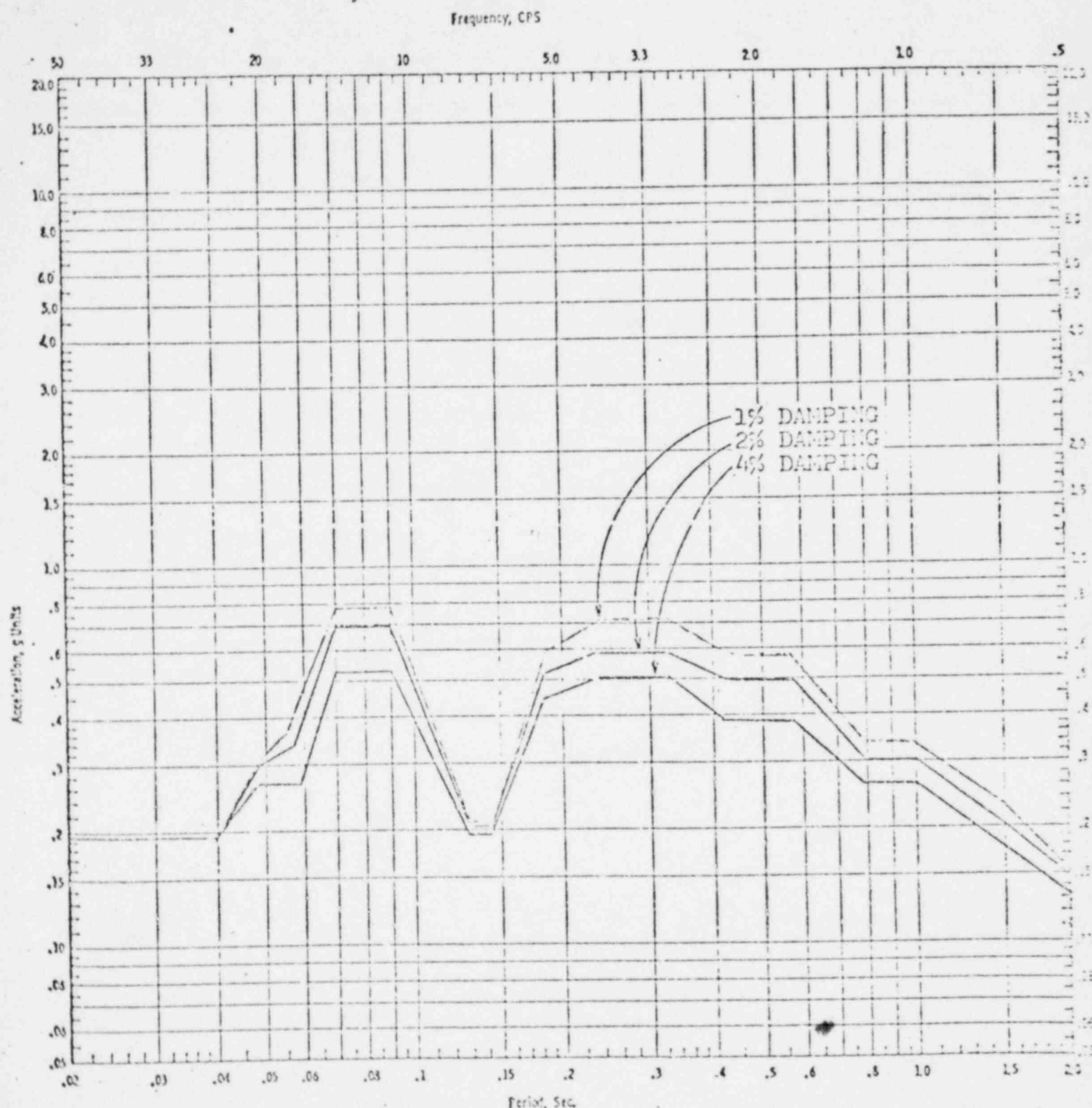
W8x13 $S_x = 9.91 \text{ in}^3$ 
 $F_b \text{ (OBE)} = 24 \text{ ksi}$ Continuous Lateral Support

$$W = 576.3 + 60.9 = 637.2 \frac{\text{lbs.}}{\text{ft}}$$

$$M_{\max} = (637.2) (8.5)^2 (1/8) = 5754.7 \text{ ft-lbs}$$

$$f_b = \frac{(5754.7) (12)}{9.91} = 7 \text{ ksi} < 24 \text{ ksi (OK)}$$

REV. NO.	1	2	7		
DATE	2-27-54	3-14-54	11-1-54		
INITIALS	W.C.	K.D.	J.H.S.		

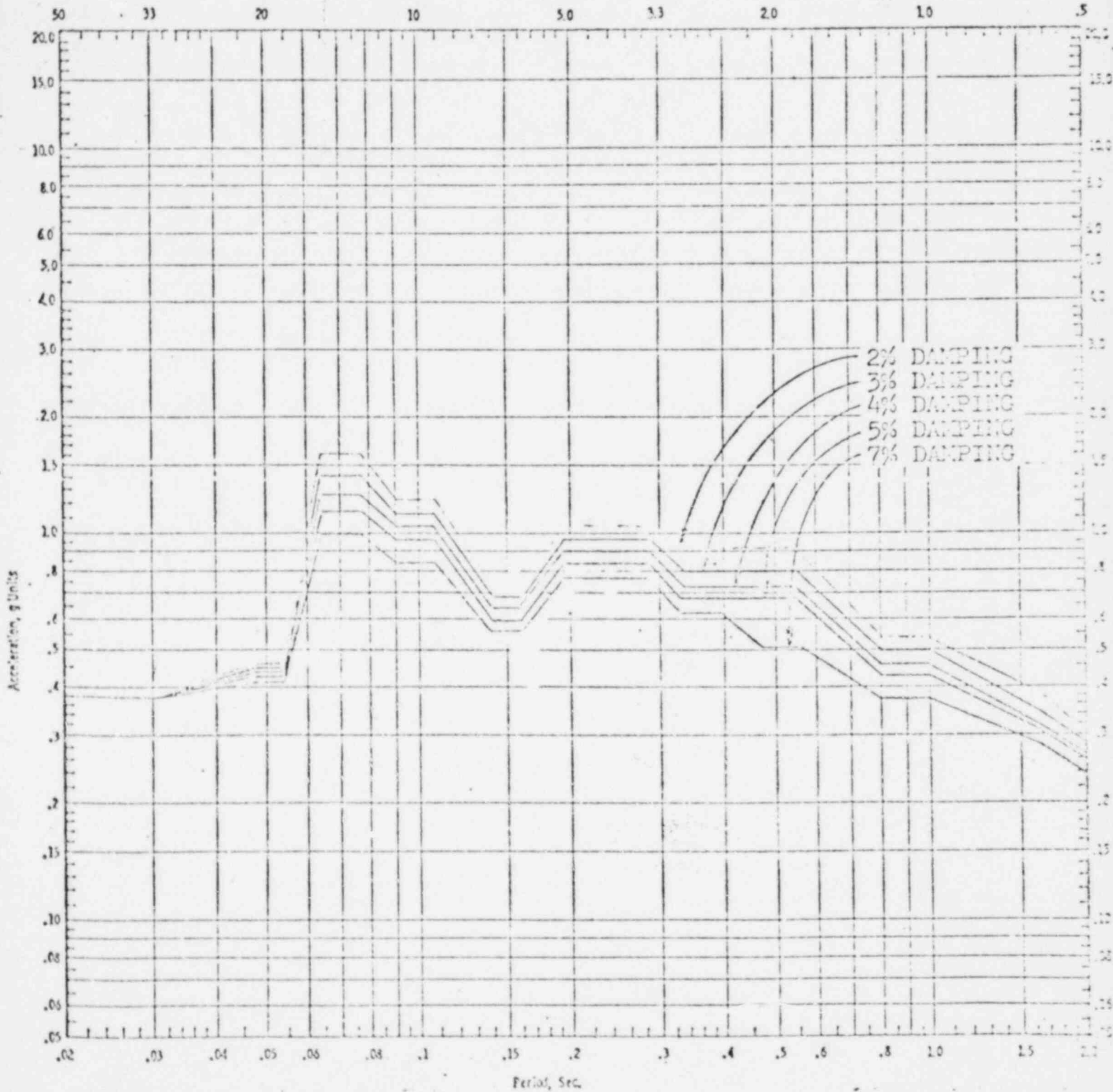


Horizontal Floor Response Spectra
 East-West Component (C.B.E.)
 El. 401'-0"
 Auxiliary-Heater Bay-Turbine

Spectra No.
 107-OB-EW

REV. NO.	2	7		
DATE	6-20-72	11/2/72		
INITIALS	SPS	TGS		

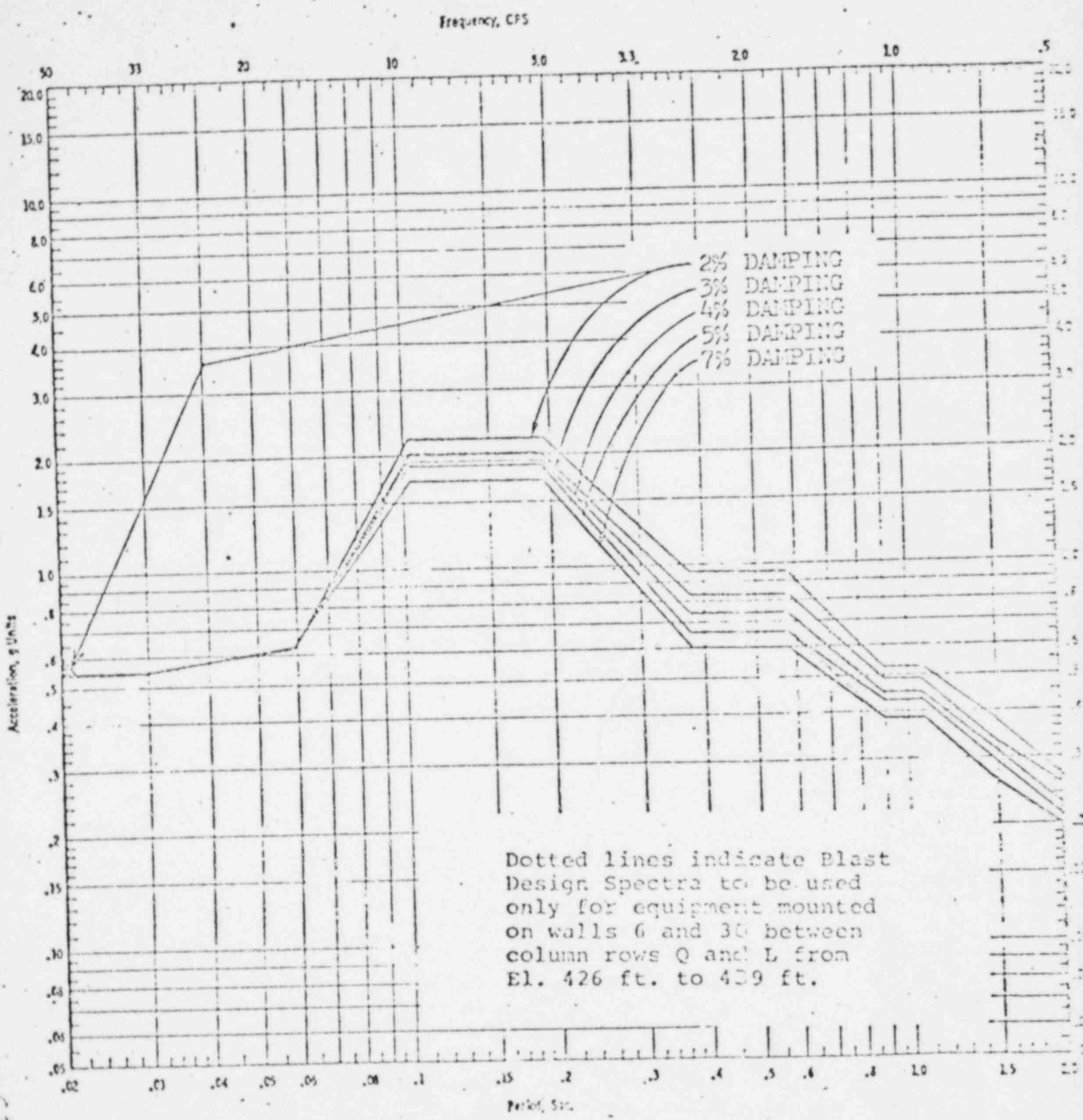
Frequency, CPS



Horizontal Floor Response Spectra
 East-West Component (S.S.E./Blast)
 El. 401'-0"
 Auxiliary-Heater Bay-Turbine

Spectra No.
 107-SS-BW

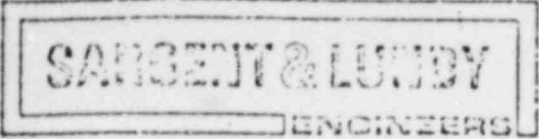
REV. NO.	1	2	3			
DATE	7-7-76	3-30-77	5-28-77			
INITIALS	HS	R.S.	J.H.S.			



Dotted lines indicate Blast Design Spectra to be used only for equipment mounted on walls 6 and 30 between column rows Q and L from El. 426 ft. to 439 ft.

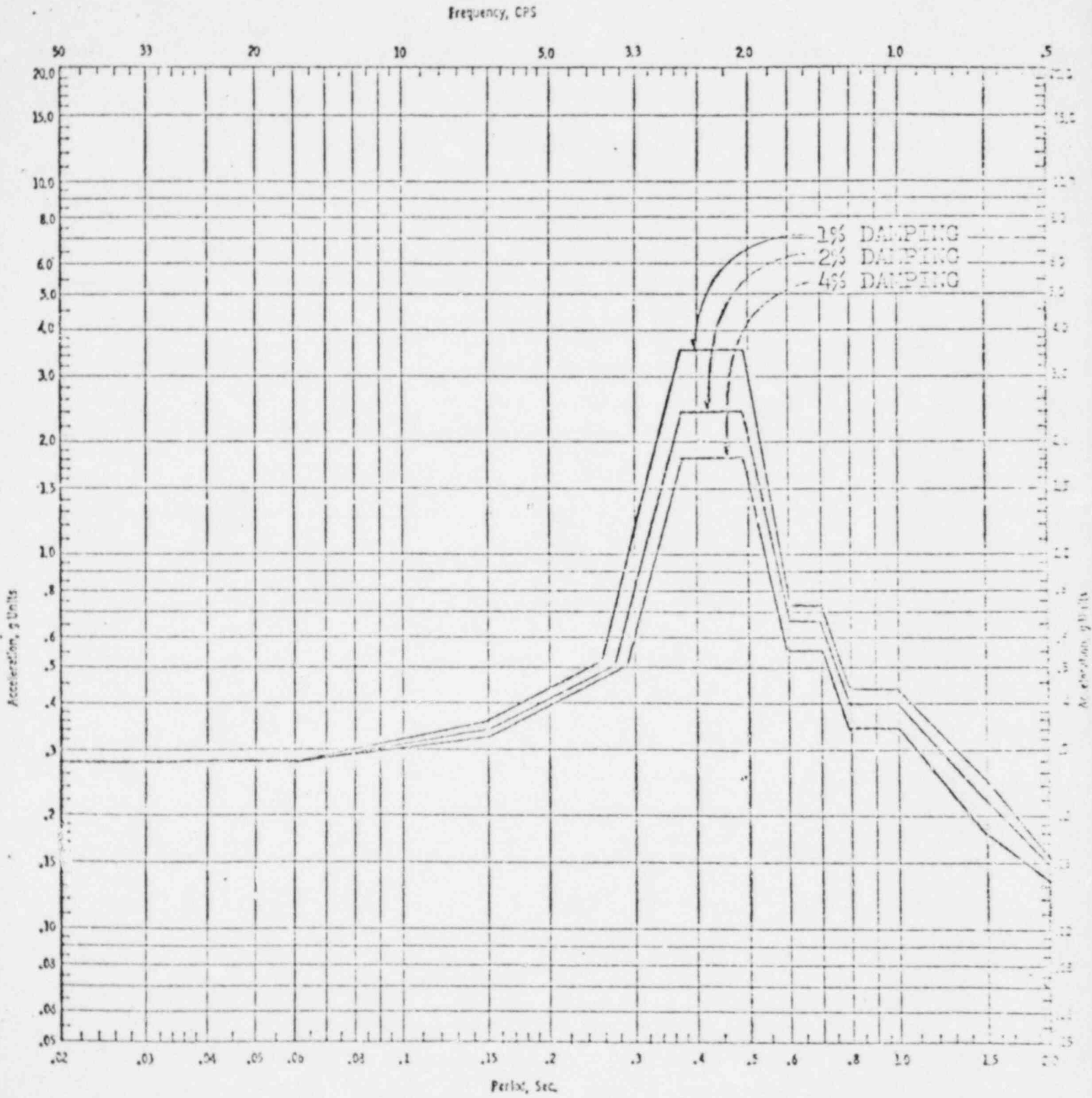
Horizontal Floor Response Spectra
 North-South Component (S.S.E./Blast)
 El. 439'-0"
 Auxiliary Building

Spectra No. 109-SS-NS



CLIENT COMMONWEALTH PETROL CO.
 PROJECT W. 100 ROAD Enclosure 3
 DESIGN BY _____ DATE 3-1-73
 CHECKED BY _____ DATE 3-22-73 SHEET 42 OF 150

REV. NO.	1	2				
DATE	2-27-73	3-24-73				
INITIALS		C.S.				



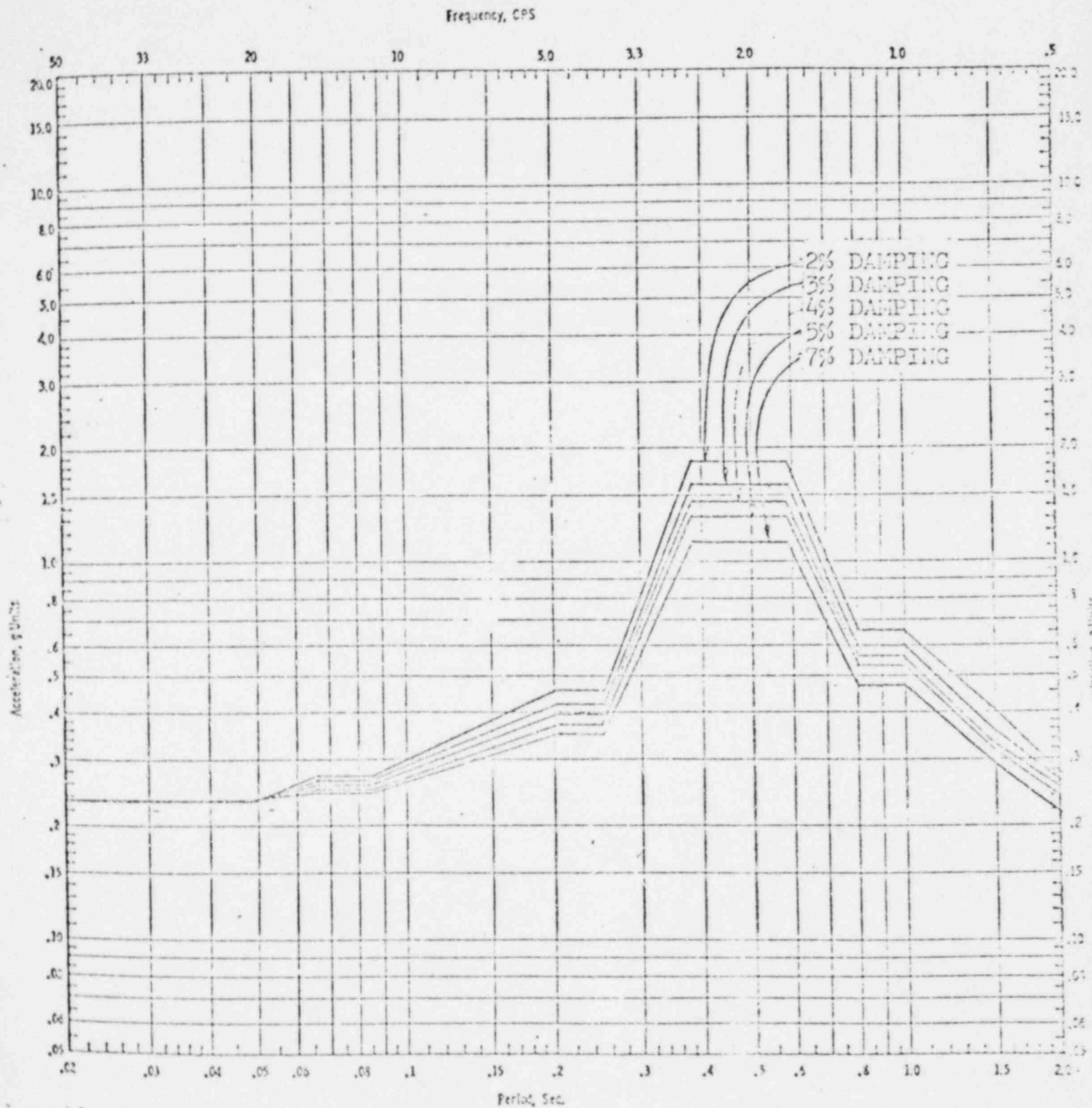
Horizontal Floor Response Spectra
 East-West Component (O.B.E.)
 El. 401'-0"
 Fuel Handling Building

Spectra No.
 106-0B-EW

SARGENT & LUNDY
ENGINEERS

CLIENT GENERAL ELECTRIC COMPANY
 PROJECT TR / MAIN CO. Enclosure 3
 DESIGN BY _____ DATE 2-27-72
 CHECKED BY _____ DATE 3-23-72 SHEET 44 OF 150

REV. NO.	1	2			
DATE	2-7-72	3-23-72			
INITIALS	HS	RS			



Horizontal Floor Response Spectra
 East-West Component (S.S.E./Blast)
 El. 401'-0"
 Fuel Handling Building

Spectra No.
 106-SS-EW

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 27: NRC Question No. 130.53
(Spent Fuel Pool Racks)

The NRC requested comparison of the method used for seismic analysis of the spent fuel storage racks versus the SRP 3.8.4 method using spectral velocity. Enclosed is a letter from NUS Corporation dated December 18, 1981 providing the information requested. The method used by NUS is more conservative than the SRP method.



4 RESEARCH PLACE
ROCKVILLE, MARYLAND 20850
301 948-7010

5106/5107-NUS-208
December 18, 1981

Mr. M. Amin
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

Subject: New and Spent Fuel Storage Racks
Byron/Braidwood Units 1 & 2
P.O. Nos. 205996 and 205997
S&L Spec. F/L-2743

- References:
1. Appendix D to SRP 3.8.4, "Technical Position on Spent Fuel Pool Racks", Rev. 0, July 1981
 2. NUS Internal Correspondence EMD-HJE-187

Dear Mr. Amin:

Per your request, the following information is provided for your use in responding to NRC on their question 130.53. This question basically asked if the racks were designed in accordance with Appendix D to SRP3.8.4 (Reference 1). Our response was that the only exception of note concerns the basis for calculating the maximum velocity of the fuel assembly for use in the fuel-can impact analysis. Our procedure, and the reasons for not following Reference 1, were given in Reference 2.

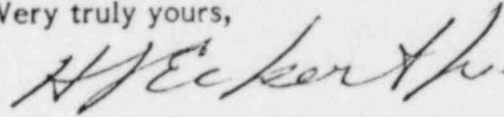
The maximum velocity of the fuel during the SSE as calculated by NUS is 12in/sec. This velocity is equal to the SRSS of the maximum floor velocity and the maximum velocity of the fuel rack with respect to the floor. The NRC position is that "the maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly". The natural frequency of the submerged fuel assembly is a function of the support, or restraint, afforded the assembly. In our design, as well as the typical design in the industry, the fuel assembly is restrained laterally by frictional forces at the bottom and by the walls of the guide tube, or can, along the upper portion of the assembly. During the postulated seismic event, this upper support changes from no restraint during the time the assembly moves from one side of the can to the other, and the length and location of contact of the support changes after initial contact as the assembly deforms. Since the assembly support is a function of time, the natural frequency of the assembly is also a function of time.

The fundamental frequency of a submerged PWR fuel assembly with simply supported end conditions is on the order of 3 Hz. For the case where the assembly is supported (pinned) only at the bottom, its first nonzero frequency is approximately 5 Hz. For the case of maximum lateral support, i.e., when the upper one-third of the fuel assembly is in contact with the can, the frequency is approximately 9 Hz. The table below gives the spectral velocities corresponding to these frequencies for the appropriate SSE floor response spectrum at 4% damping.

<u>Frequency (Hz)</u>	<u>Velocity (in/sec)</u>
3	9.6
5	8.1
9	6.8

The velocity, as calculated using the above procedure, varies from about 6 to 10 in/sec., depending upon the conditions assumed. The value of 12 in/sec. used by NUS is therefore greater than that which would be calculated by using the procedure of Reference 1.

Very truly yours,



Howard J. Eckert, Jr.
Manager, Engineering
Mechanics Department

/ac

cc: P. D. Arrowsmith
B. J. Reckman
J. L. Renehan
S. B. Gerges
G. Antonucci, Jr.
DCC
File

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 29: Buried Piping

S&L to provide clarification of the calculations showing the effects of a joint displacement at the 90° bend of the Essential Service Water Pipe Concrete Encasement. Calculations should also be provided for the effects of axial load in the concrete pipe encasement.

Refer to the attached sheets; calculation number 2.3.3. BY, Revision 1, pages 45 and 51.

Client *CE Co*
Project *24*
Proj. No. *9341* Equip. No.

Prepared by *Robert...* Date *12-1-51*
Reviewed by *J. H.* Date *12-2-51*
Approved by *J. J. Fobe* Date *2-2-51*

Concrete measurement check (Cont.)

$M_x SSE = 190 K'$	} straight M.M.	} From Comp.
$F_{K SSE} = 8 K$		
$M_y SSE = 132 K'$		
$M_x SSE = 287 K'$	} Rt. Angle M.M.	
$M_y SSE^* = 30 K'$		
$F_{a SSE} = 5.27 K$		
$M_T SSE = 86 K'$		

Rt. Angle M.M. Corrections

$M_{x static} = wL^2 \left(\frac{1}{15} + \frac{3}{20} \right)$ $M_{y static} = \frac{1}{2} (2.5) 25^2$
 $\approx 293 \text{ ft}$ $= 286 K'$
 Paul Anderson's = $5.5 (25^2) (.267 + .15)$
 Statistically
 Determinants = $1432 K'$
 Corrections

$M_{ux} = 1.0 (M_{static}) + 1.0 (M_{SSE})$ $M_{uy} = 280 K'$
 $= 1.0 (1432) + 1.0 (287)$ $M_{uy} = (286 + 30) 1.0$
 $M_{ux} = 1721 K' < 2197 K'$ $M_{ux} = 316 K' < 880$

Check Torsional Shear

Torsional Mom = $86 K' SSE + 5.5 (25^2) \left(\frac{1}{20} + \frac{1}{30} \right)$ Anderson ibid.
 $= 86 K' SSE + 286 K' STATK$
 $= 372 K' \approx 445,400 \text{ ft}^2$

Form GO 3.05.1 Rev. 2

* THE MOMENT M_y RESULTING FROM THE DEFLECTION AT THE BEND IS INCLUDED IN THE $M_{y SSE}$ GIVEN BY END CALCULATIONS. EMO-034142

Client CE. C.
Project (VLT)
Proj. No. 4391 Equip. No.

Prepared by J. L. 71 Date 12/13/91
Reviewed by [Signature] Date 12-15-91
Approved by [Signature] Date 12-20-91

CALCULATE ADDITIONAL STRESS IN CONCRETE REINFORCEMENT DUE TO AXIAL FORCE.

AXIAL FORCE AT 90° BEND = 5270 lbs.

TOTAL LONGITUDINAL REINFORCEMENT = 18#9 = 18 in²
 18#10 = 22.96 in²
 8#9 = 9
 49.96 in²

ADDITIONAL AXIAL STRESS = $\frac{5.27}{49.96} = .11 \text{ KSI}$

NEGLECTIBLE.

BYRON/BRAIDWOOD
STRUCTURAL DESIGN AUDIT (SUPPLEMENTAL INFORMATION)
OCTOBER, 1981

Action Item 30: Missile Protection for Manhole Cover

S&L to provide information for the manhole cover to comply with missile protection.

Ductile iron with 100 ksi tensile strength is being provided for all Category I manhole covers. These manhole covers have been designed in accordance with SRP 3.5.3 for tornado missile effects.