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Ltr:WRC:01-88-30

OAK RIDGE NATIONAL LABORATORY

OPERATED BY MARTIN MARIETTA ENERGY SYSTEMS INC

POST OFFICE BOX Y
OAK RIDGE, TENNESSEE 37831

February 11, 1988

Mr. M. E. Mayfield
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
5650 Nicholson Lane
Rockville, MD 20852

Dear Mr. Mayfield:

Continued Investigation of the Integrity of LWR Vessel Supports

As a followup to our letter to C. Z. Serpan of September 2, 1987, regarding the integrity of LWR vessel support structures, and in response to your request, we are describing our plans for continuing the investigation of vessel supports.

It is our intent to categorize supports by types, select a design that is believed to be the most susceptible to failure as a result of radiation degradation, and determine, by detailed fracture-mechanics analysis, if there is at least one LWR plant for which excessive radiation degradation (embrittlement) of the vessel supports is likely before the end of design life. With your concurrence we have already initiated our effort and are at the point of selecting a specific PWR plant for detailed evaluation of its vessel supports. We believe the study described can be completed by the end of FY 1988 and look forward to discussing appropriate budgetary arrangements and other technical priorities with you.

Please let me know if we can provide additional information.

Sincerely,

W. R. Corwin

W. R. Corwin, Manager
Heavy-Section Steel Technology
Program

WRC:mt

Attachment

cc/enc: R. D. Cheverton
A. P. Malinauskas
R. K. Nanstad
C. E. Pugh
G. C. Robinson
H. E. Trammell

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FOIA 88-118 C/1

FOIA-88-118

CONTINUATION OF INVESTIGATION OF THE INTEGRITY OF LWR VESSEL SUPPORTS*

R. D. Cheverton   G. C. Robinson
W. R. Corwin

Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

1.0 INTRODUCTION

This document identifies the research activities we perceive to be needed as a continuation of our preliminary study[1] performed on the integrity of LWR vessel support structures. The intent is to determine if there are any LWR plants for which excessive radiation degradation of the vessel supports is likely before the end of design life. This study will also provide input for addressing the three goals cited in Ref. 2.

2.0 RECOMMENDED APPROACH

After our earlier preliminary study[1] and discussions with the nuclear industry concerning the integrity of LWR vessel support structures, it was apparent to us that a meaningful continuation of the study would require specific-plant evaluations because of the complex and diversified nature of support designs. As indicated in the summary (Table 1), a thorough search of the literature for types and details of specific support designs should be the first step in our continued effort. This effort has in fact been under way for some time, and summaries of the acquired information are given in Tables 2 and 3 and Figs. 1 through 10.

The next step is the identification of one or two support designs that appear to be the most susceptible to failure as a result of radiation degradation. With the aid of the nuclear industry, detailed design data will be obtained for these supports and an analysis performed to determine the effect of radiation degradation on their permissible lifetimes. The potential for failure of the supports will be evaluated using fracture-mechanics methods of analysis and, to the extent possible, will consider (1) the radiation-damage rate effect deduced from the HFIR surveillance program, (2) secondary (residual, thermal, etc.) stresses as well as primary stresses, and (3) all credible loading conditions. Radiation damage will be assessed on the basis of DPA to account for the softer neutron spectrum for LWR supports relative to those in the HFIR vessel, LWR surveillance specimens, and test specimens in MTR's.

As indicated in Table 1, it is anticipated that the investigation described herein can be completed by the end of FY 1988 at a cost of ~$320 K.

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*Attachment to letter from W. R. Corwin, ORNL, to M. E. Mayfield, NRC/RES, "Proposal for Investigation of the Integrity of LWR Vessel Supports," February 8, 1988.

Criteria pertaining to the evaluation of the integrity of the vessel supports will have to be established as a part of the study. In this regard there are two points of particular interest: (1) should reactor vessel supports be classified as Class-1 structures, and (2) are large-break LOCAs, involving severance of a main coolant line close to the vessel, considered credible? According to the ASME Code [Sect. III, Subsect. NF-3131 (e)], "For Class 1 piping and component supports, protection against nonductile fracture shall be provided." Depending upon the interpretation, this might imply that the operating temperatures should be well above NDTT. With regard to the large-break LOCA, the resulting horizontal loads on the vessel would probably be substantially greater than any others considered credible.

As indicated in Table 1, 9 man-months of effort is estimated for analysis of the one or two structures selected for detailed analysis. This assumes that ORNL will have to perform essentially all of the analysis. Hopefully, the designers of the supports (AE's and/or NSS suppliers) will be able and willing to participate, in which case the ORNL effort could be significantly less. Preliminary contacts with industry indicate a desire to participate, but the question will remain open until the specific supports for evaluation have been selected.

Selection of one or two specific plants for vessel-support evaluation will not necessarily imply that supports at other plants having the same or a different category of supports should not eventually be reevaluated. As indicated in Ref. 1, there are general concerns about all catgories, particularly with regard to secondary stresses that might not have been considered in the original design analysis. It is important to note that a detailed evaluation of all LWR vessel supports is not within the scope of this investigation.

3.0 STATUS OF CURRENT WORK

As mentioned above, our first step is to establish what basic types of vessel supports exist, and this effort is essentially complete. Tables 2 and 3 summarize the "type" data and indicate the following:

1. All but one of the BWR vessels (Big Rock Point is the exception) **and 10% of the PWR vessels (all B&W vessels but two) are supported on skirts, as illustrated in Figs. 1 and 2.**

2. 10% of PWR vessels are supported on long columns (Fig. 3).

3. 10% of PWR vessels are supported on shield tanks (Fig. 4).

4. 70% of PWR vessels are supported by short "columns" that extend down from the nozzles to the concrete biological shield or to a steel cantilever beam extending out from the biological shield at an elevation above the core midheight and in some cases below the top of the core (Figs. 5 through 8).

5. The Big Rock Point vessel is suspended by long rods (in tension) that extend down to the elevation of the top of the core (Fig. 9).

TABLE 1

SUMMARY OF COST AND SCHEDULE DATA FOR PWR VESSEL SUPPORT EVALUATION PROGRAM

Task	Completion Date	[redacted]
1. Search literature for types of supports; categorize according to basic design concepts	2-15-88	
2. Select specific plants for detailed analysis of supports	3-15-88	
3. Consult with industry (EPRI, NSS, Utility, and AE) to establish background information on the supports of Item 2; i.e., (a) drawings, (b) fabrication history, (c) materials, (d) preservice and in-service inspection, (e) design and analysis, and (f) radiation and temperature environment	5-15-88	
4. Establish cooperative relationship with industry to assist in analysis of Item 2 supports	6-1-88	
5. Complete analyses	9-1-88	
6. Review analyses and prepare recommendations to NRC	10-1-88	
	Subtotal	
	Travel	
	Computer	
	Report	
	Contingency	
	Total	

Ex. 5

It is tentatively concluded that skirt supports (40% of all LWR supports) are too far removed from the core to experience significant radiation degradation. Shield-tank and long-column supports extend the full length of the core and thus are exposed to relatively high neutron fluxes. Supports in categories 4 and 5 are exposed to intermediate levels of the flux, but in some cases the fluxes may be nearly the maximum available because of the rather flat distribution of the flux in the axial direction (Fig. 10).

Recent studies conducted by industry indicate that the integrity of the shield-tank and long-column supports is not excessively challenged by radiation degradation,[3,4] and a similar indication was expressed[5] at a recent DOE workshop held at Sandia National Laboratory on the topic of embrittlement of reactor support structures. We are not aware of recent studies similar to those on shield tanks and column supports for categories 4 and 5. Based on data accumulated thus far, it appears that some of these latter supports include weldments that may introduce significant residual tensile stress; there are tie-down bolts, which, of course, are in tension; and the rod-suspension (Fig. 9) and cantilever (Figs. 7 and 8) designs introduce bending and thus tensile stresses. It may be that shield-tank and/or long-column supports will prove to be no less susceptible to failure as a result of radiation degradation. However, we have tentatively concluded that categories 4 and 5 should be considered first for selection of specific plants for a detailed evaluation of the vessel supports. These two categories include ~47% of all LWR's, but within this category there is considerable variation in support detail design and proximity of support to the core.

As mentioned above, selection of one or two categories of supports for specific-plant analysis does not imply that others should not be evaluated in light of the new embrittlement data from HFIR.

4.0 REFERENCES

1. R. D. Cheverton, ORNL letter to C. Z. Serpan, NRC, "LWR Vessel Supports," September 2, 1987.

2. Victor Stello, Jr., NRC letter to William Kerr, Chairman, ACRS, "ACRS Comments on the Embrittlement of Structural Steel in Reactor Support Structures," January 28, 1988.

3. Personal communication between R. D. Cheverton, ORNL, and T. J. Griesbach, EPRI.

4. Personal communication between R. D. Cheverton, ORNL, and D. J. Ayres, CE.

5. C. F. Bergeron, Stone and Webster, "Neutron Shield Tanks," presented at the DOE Plant Lifetime Improvement Program, Albuquerque, Sept. 15-16, 1987.

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TABLE I

SUMMARY OF COST AND SCHEDULE DATA FOR PWR VESSEL SUPPORT EVALUATION PROGRAM

Task	Completion Date	[redacted]
1. Search literature for types of supports; categorize according to basic design concepts	2-15-88	
2. Select specific plants for detailed analysis of supports	3-15-88	
3. Consult with industry (EPRI, NSS, Utility, and AE) to establish background information on the supports of Item 2; i.e., (a) drawings, (b) fabrication history, (c) materials, (d) preservice and in-service inspection, (e) design and analysis, and (f) radiation and temperature environment	5-15-88	
4. Establish cooperative relationship with industry to assist in analysis of Item 2 supports	6-1-88	
5. Complete analyses	9-1-88	
6. Review analyses and prepare recommendations to NRC	10-1-88	
	Subtotal	
	Travel	
	Computer	
	Report	
	Contingency	
	Total	

Ex. 5

Table 2. Basic Types of Support for LWR Vessels

No.	Power Station Name	Type of Plant	Types of Support: Skirt 1	Long Column 2	Shield Tank 3	Short Column 4	Suspension 5
1	Arkansas Nuclear Unit 1	PWR	1				
2	Arkansas Nuclear Unit 2	PWR		2			
3	Beaver Valley Unit 1	PWR			3		
4	Big Rock Point Unit 1	BWR					5
5	Browns Ferry Unit 1	BWR	1				
6	Browns Ferry Unit 2	BWR	1				
7	Browns Ferry Unit 3	BWR	1				
8	Brunswick Unit 1	BWR	1				
9	Brunswick Unit 2	BWR	1				
10	Byron Unit 1	PWR				4	
11	Callaway Unit 1	PWR				4	
12	Calvert Cliffs Unit 1	PWR				4	
13	Calvert Cliffs Unit 2	PWR				4	
14	Catawba Unit 1	PWR				4	
15	Cook Unit 1	PWR				4	
16	Cook Unit 2	PWR				4	
17	Cooper Station	BWR	1				
18	Crystal River Unit 3	PWR	1				
19	Davis-Besse Unit 1	PWR				4	
20	Diablo Canyon Unit 1	PWR				4	
21	Diablo Canyon Unit 2	PWR				4	
22	Dresden Unit 2	BWR	1				
23	Dresden Unit 3	BWR	1				
24	Duane Arnold	BWR	1				
25	Farley Unit 1	PWR				4	
26	Farley Unit 2	PWR				4	
27	Fitzpatrick	BWR	1				
28	Fort Calhoun Unit 1	PWR				4	
29	Fort St. Vrain	GCR		Not Applicable			
30	Ginna	PWR				4	
31	**Grand Gulf Unit 1**	**BWR**	**1**				
32	Haddam Neck	PWR				4	
33	Hatch Unit 1	BWR	1				
34	Hatch Unit 2	BWR	1				
35	Indian Point Unit 2	PWR				4	
36	Indian Point Unit 3	PWR				4	
37	Kewaunee	PWR				4	
38	La Crosse	BWR	1				
39	La Salle Unit 1	BWR	1				
40	La Salle Unit 2	BWR	1				
41	Limerick Unit 1	BWR	1				
42	Maine Yankee	PWR			3		

Table 2. (Continued)

No.	Power Station Name	Type of Plant	Types of Support				
			Skirt	Long Column	Shield Tank	Short Column	Suspension
			1	2	3	4	5
43	McGuire Unit 1	PWR				4	
44	McGuire Unit 2	PWR				4	
45	Millstone Unit 1	BWR	1				
46	Millstone Unit 2	PWR				4	
47	Monticello	BWR	1				
48	Catawba Unit 2	PWR				4	
49	Hope Creek Unit 1	BWR	1				
50	Millstone Unit 3	PWR			3		
51	Nine Mile Point Unit 1	BWR	1				
52	North Anna Unit 1	PWR			3		
53	North Anna Unit 2	PWR			3		
54	Oconee Unit 1	PWR	1				
55	Oconee Unit 2	PWR	1				
56	Oconee Unit 3	PWR	1				
57	Oyster Creek Unit 1	BWR	1				
58	Palisades	PWR				4	
59	Palo Verde Unit 1	PWR		2			
60	Palo Verde Unit 2	PWR		2			
61	Peach Bottom Unit 2	BWR	1				
62	Peach Bottom Unit 3	BWR	1				
63	Pilgrim Unit 1	BWR	1				
64	Point Beach Unit 1	PWR				4	
65	Point Beach Unit 2	PWR				4	
66	Prairie Island Unit 1	PWR		2			
67	Prairie Island Unit 2	PWR		2			
68	Quad Cities Unit 1	BWR	1				
69	Quad Cities Unit 2	BWR	1				
70	Rancho Seco Unit 1	PWR	1				
71	River Bend Unit 1	BWR	1				
72	Robinson Unit 2	PWR				4	
73	Salem Unit 1	PWR				4	
74	Salem Unit 2	PWR				4	
75	San Onofre Unit 1	PWR		2?			
76	San Onofre Unit 2	PWR		2			
77	San Onofre Unit 3	PWR		2			
78	Sequoyah Unit 1	PWR				4	
79	Sequoyah Unit 2	PWR				4	
80	St. Lucie Unit 1	PWR		2			
81	St. Lucie Unit 2	PWR		2			
82	Summer Unit 1	PWR				4	
83	Surry Unit 1	PWR			3		
84	Surry Unit 2	PWR			3		

Table 2. (Continued)

No.	Power Station Name	Type of Plant	Types of Support				
			Skirt	Long Column	Shield Tank	Short Column	Suspension
			1	2	3	4	5
85	Susquehanna Unit 1	BWR	1				
86	Susquehanna Unit 1	BWR	1				
87	Three Mile Island Unit 1	PWR	1				
88	Trojan	PWR				4	
89	Turkey Point Unit 3	PWR				4	
90	Turkey Point Unit 4	PWR				4	
91	Vermont Yankee Unit 1	BWR	1				
92	Washington Nuclear Unit 2	PWR	1				
93	Waterford Unit 3	PWR				4	
94	Wolfe Creek Unit 1	PWR				4	
95	Yankee-Rowe Unit 1	PWR			3		
96	Zion Unit 1	PWR				4	
97	Zion Unit 2	PWR				4	
98	Vogtle Unit 1	PWR				4	
99	Vogtle Unit 2	PWR				4	
100	Beaver Valley Unit 2	PWR			3		
101	Bellefonte Unit 1	PWR				4	
102	Bellefonte Unit 2	PWR				4	
103	Braidwood Unit 1	PWR				4	
104	Braidwood Unit 2	PWR				4	
105	Byron Unit 2	PWR				4	
106	Clinton Unit 1	PWR	1				
107	Comanche Peak Unit 1	PWR				4	
108	Comanche Peak Unit 2	PWR				4	
109	Fermi Unit 2	PWR	1				
110	Grand Gulf Unit 2	PWR	1				
111	Limerick Unit 2	PWR	1				
112	Nine Mile Point Unit 2	PWR	1				
113	Palo Verde Unit 3	PWR		2			
114	Perry Unit 1	PWR	1				
115	Perry Unit 2	PWR	1				
116	Seabrook Unit 1	PWR				4	
117	Seabrook Unit 2	PWR				4	
118	Shearon Harris Unit 1	PWR				4	
119	Shearon Harris Unit 2	PWR				4	
120	Shoreham	PWR	1				
121	South Texas Unit 1	PWR				4	
122	South Texas Unit 2	PWR				4	
123	Watts Bar Unit 1	PWR				4	
124	Watts Bar Unit 2	PWR				4	
125	Washington Nuclear Unit 1	BWR				4	
126	Washington Nuclear Unit 3	PWR		2			

Table 3. Points and Means of Vessel-Support Load Application to Vessel, Shield and Floor

No.	Power Station Name	Point of load application to vessel			Type of Support Feature Transmitting Vertical Loads to Shield or Floor				Type of Support Feature Transmitting Horizontal Loads to Shield or Floor				
		Bottom Head	Inlet and/or Outlet Nozzle	Wall-Mounted Brackets	Skirt to Floor	Weldment to Shield	Columns to Floor	Unusual Structural Features	Skirt to Floor	Weldment/Key Anchored to Floor	Weldment/key with Segmented or Ring Girder to Wall	Weldment/key with Embedded Structural Steel to Wall	Key to Shield
		A	B	C	A	B	C	D	A	B	C	D	E
1	Arkansas Nuclear Unit 1	A			A				A				
2	Arkansas Nuclear Unit 2		B				C			B			
3	Beaver Valley Unit 1		B					D					E
4	Big Rock Point Unit 1			C				D					
5	Browns Ferry Unit 1	A			A				A				
6	Browns Ferry Unit 2	A			A				A				
7	Browns Ferry Unit 3	A			A				A				
8	Brunswick Unit 1	A			A				A				
9	Brunswick Unit 2	A			A				A				
10	Byron Unit 1		B			B							
11	Callaway Unit 1		B			B				B	C?	D?	
12	Calvert Cliffs Unit 1		B			B				B			
13	Calvert Cliffs Unit 2		B			B				B			
14	Catawba Unit 1		B			B				B			
15	Cook Unit 1		B										
16	Cook Unit 2		B										
17	Cooper Station	A			A				A				
18	Crystal River Unit 3	A			A				A				
19	Davis-Besse Unit 1		B					D				D	
20	Diablo Canyon Unit 1		B			B							
21	Diablo Canyon Unit 2		B			B							
22	Dresden Unit 2	A			A				A				
23	Dresden Unit 3	A			A				A				
24	Duane Arnold	A			A				A				
25	Farley Unit 1		B			B				B			
26	Farley Unit 2		B			B				B			
27	Fitzpatrick	A			A				A				
28	Fort Calhoun Unit 1		B										
29	Fort St. Vrain						Not Applicable						
30	Ginna		B	C?									
31	Grand Gulf Unit 1	A			A				A				
32	Haddam Neck		B			B				B			
33	Hatch Unit 1	A			A				A				
34	Hatch Unit 2	A			A				A				
35	Indian Point Unit 2		B			B					C		
36	Indian Point Unit 3		B			B					C		
37	Kewaunee		B			B		D				D	
38	La Crosse			C?				D	A?				
39	La Salle Unit 1	A			A				A				
40	La Salle Unit 2	A			A				A				
41	Limerick Unit 1	A			A				A				
42	Maine Yankee		B					B					E
43	McGuire Unit 1		B			B				B			
44	McGuire Unit 2		B			B				B			
45	Millstone Unit 1	A			A				A				
46	Millstone Unit 2		B			B				B			
47	Monticello	A			A				A				
48	Catawba Unit 2		B			B				B			
49	Hope Creek Unit 1	A			A				A				
50	Millstone Unit 3		B			B							E
51	Nine Mile Point Unit 1	A			A				A				
52	North Anna Unit 1		B					D					E
53	North Anna Unit 2		B					D					E
54	Oconee Unit 1	A			A				A				
55	Oconee Unit 2	A			A				A				
56	Oconee Unit 3	A			A				A				
57	Oyster Creek Unit 1	A			A				A				
58	Palisades		B			B		D		B			
59	Palo Verde Unit 1		B				C				C		
60	Palo Verde Unit 2		B				C				C		
61	Peach Bottom Unit 2	A			A				A				
62	Peach Bottom Unit 3	A			A				A				
63	Pilgrim Unit 1	A			A				A				
64	Point Beach Unit 1		B			B					C?		
65	Point Beach Unit 2		B			B					C?		

Table 3. (Continued)

No.	Power Station Name	Point of load application to vessel			Type of Support Feature Transmitting Vertical Loads to Shield or Floor				Type of Support Feature Transmitting Horizontal Loads to Shield or Floor				
		Bottom Head	Inlet and/or Outlet Nozzle	Wall-Mounted Brackets	Skirt to Floor	Weldment to Shield	Columns to Floor	Unusual Structural Features	Skirt to Floor	Weldment/Key Anchored to Floor	Weldment/key with Segmented or Ring Girder to Wall	Weldment/key with Embedded Structural Steel to Wall	Key to Shield [illegible]
		A	B	C	A	B	C	D	A	B	C	D	E
66	Prairie Island Unit 1		B				C						
67	Prairie Island Unit 2		B				C						
68	Quad Cities Unit 1	A			A								
69	Quad Cities Unit 2	A			A								
70	Rancho Seco Unit 1	A			A				A				
71	River Bend Unit 1	A			A				A				
72	Robinson Unit 2		B			B				B?	C?		
73	Salem Unit 1		B			B					C		
74	Salem Unit 2		B			B					C		
75	San Onofre Unit 1		B										
76	San Onofre Unit 2		B				C				C		
77	San Onofre Unit 3		B				C				C		
78	Sequoyah Unit 1		B			B				[illegible]			
79	Sequoyah Unit 2		B			B				[illegible]			
80	St. Lucie Unit 1		B				C				C		
81	St. Lucie Unit 2		B				C				C		
82	Summer Unit 1		B			B				B			
83	Surry Unit 1		B					D					E
84	Surry Unit 2		B					D					E
85	Susquehanna Unit 1	A			A				A				
86	Susquehanna Unit 1	A			A				A				
87	Three Mile Island Unit 1	A			A				A				
88	Trojan		B			B		D				D	
89	Turkey Point Unit 3		B			B		D				D	
90	Turkey Point Unit 4		B			B		D				D	
91	Vermont Yankee Unit 1	A			A				A				
92	Washington Nuclear Unit 2	A			A				A				
93	Waterford Unit 3		B			B		D		B			
94	Wolf Creek Unit 1		B			B				B		D?	
95	Yankee-Rowe Unit 1			C				D					E?
96	Zion Unit 1		B			B					C		
97	Zion Unit 2		B			B					C		
98	Vogtle Unit 1		B			B				B		D?	
99	Vogtle Unit 2		B			B				B		D?	
100	Beaver Valley Unit 2		B					D					E
101	Bellefonte Unit 1		B			B		D					
102	Bellefonte Unit 2		B			B		D					
103	Braidwood Unit 1		B			B							
104	Braidwood Unit 2		B			B							
105	Byron Unit 2		B			B							
106	Clinton Unit 1	A			A				A				
107	Comanche Peak Unit 1		B			B					C?	D?	
108	Comanche Peak Unit 2		B			B					C?	D?	
109	Fermi Unit 2	A			A				A				
110	Grand Gulf Unit 2	A			A				A				
111	Limerick Unit 2	A			A				A				
112	Nine Mile Point Unit 2	A			A				A				
113	Palo Verde Unit 3		B				C				C		
114	Perry Unit 1	A			A				A				
115	Perry Unit 2	A			A				A				
116	Seabrook Unit 1		B			B					C		
117	Seabrook Unit 2		B			B					C		
118	Shearon Harris Unit 1		B			B					C?		
119	Shearon Harris Unit 2		B			B					C?		
120	Shoreham	A			A				A				
121	South Texas Unit 1		B			B				B	C?		
122	South Texas Unit 2		B			B				B	C?		
123	Watts Bar Unit 1		B			B				B		D?	
124	Watts Bar Unit 2		B			B				B		D?	
125	Washington Nuclear Unit 1		B										
126	Washington Nuclear Unit 3		B				C				C		

[a]Bottom head mounted.

[b]Load applied by vessel-mounted brackets through skirt to floor.

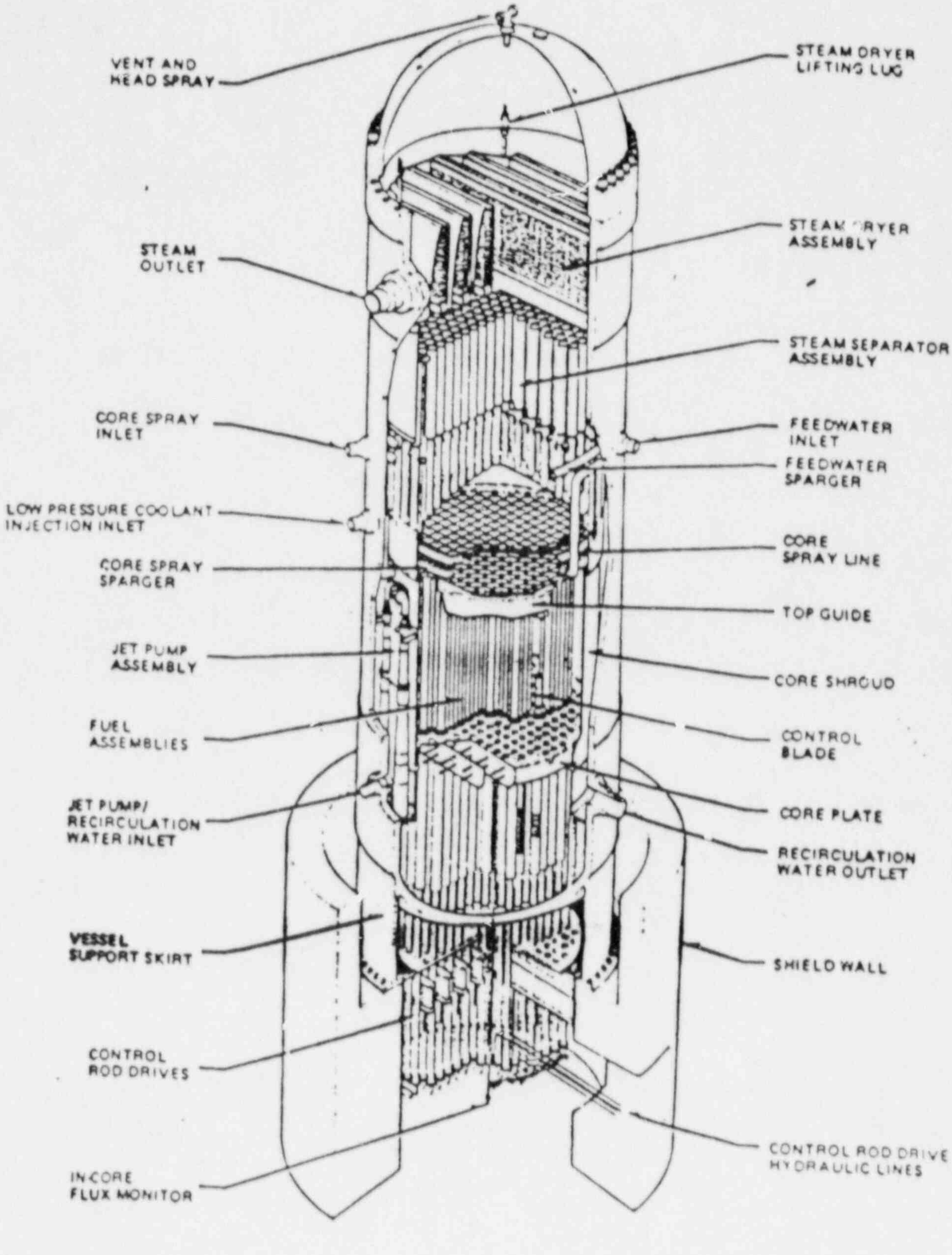


Fig. 1. Reactor vessel with skirt support (GE BWR).

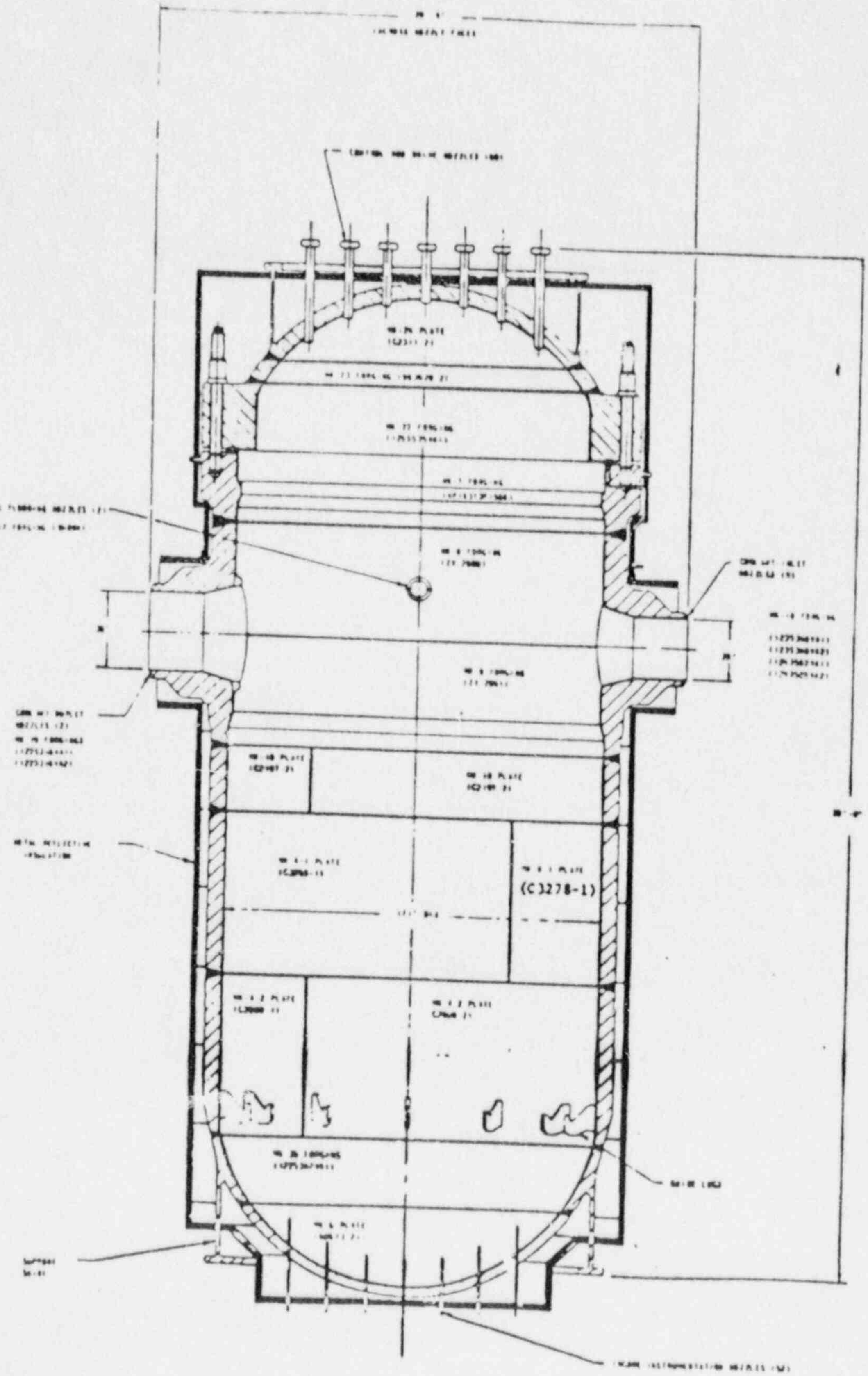


Fig. 2. Reactor vessel with skirt support (B&W PWR).

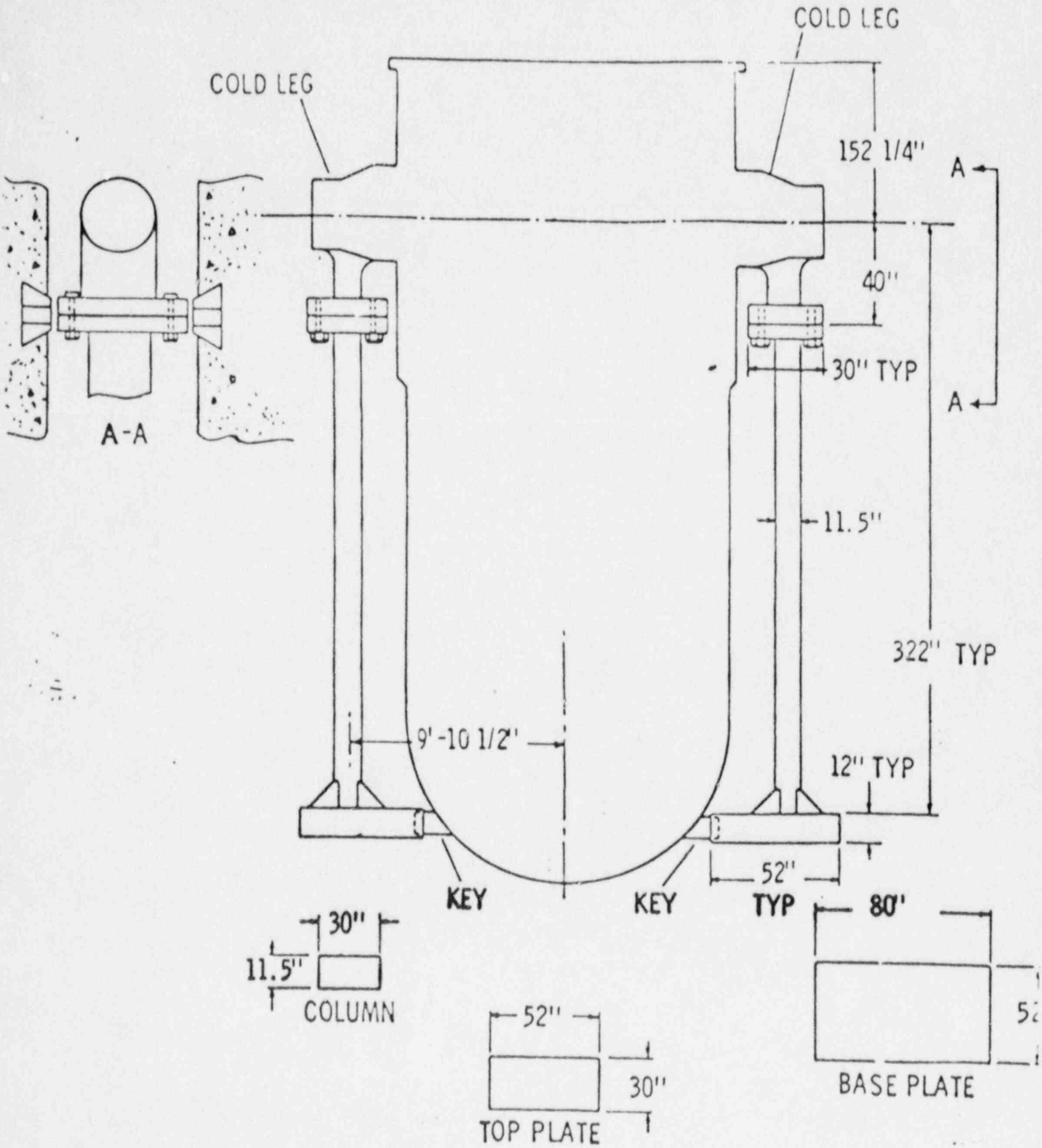
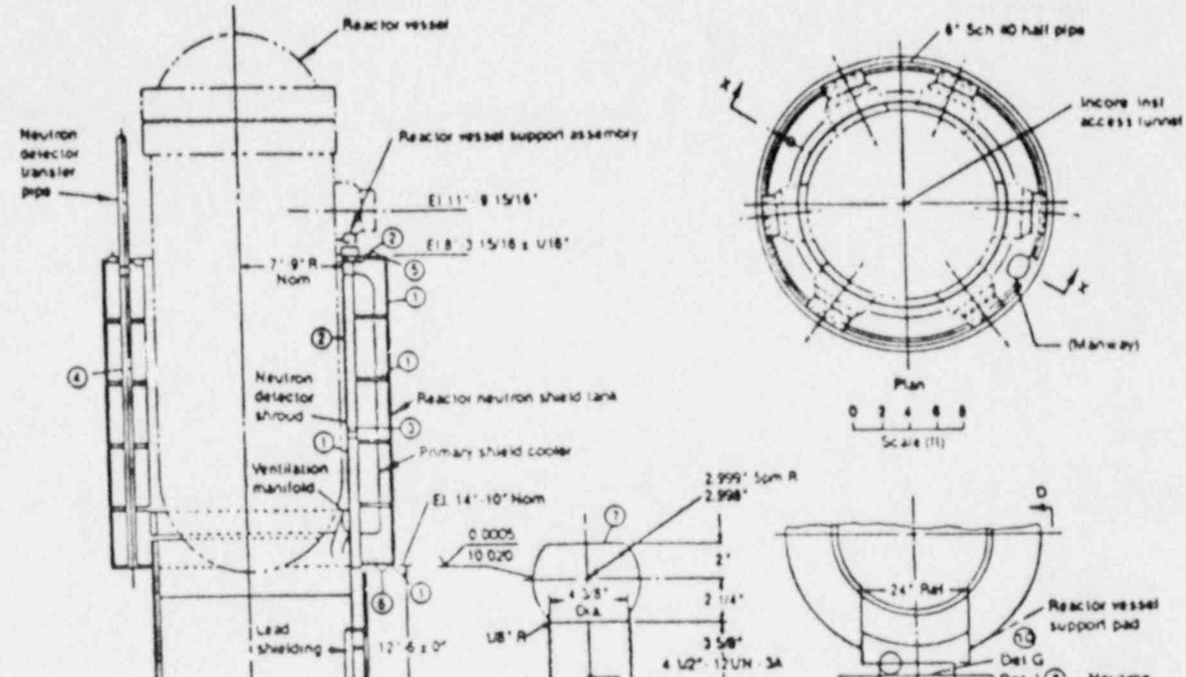


Fig. 3. PWR vessel with long-column supports.



Neutron shield tank RPV support.

Fig. 4. PWR vessel with shield-tank support.

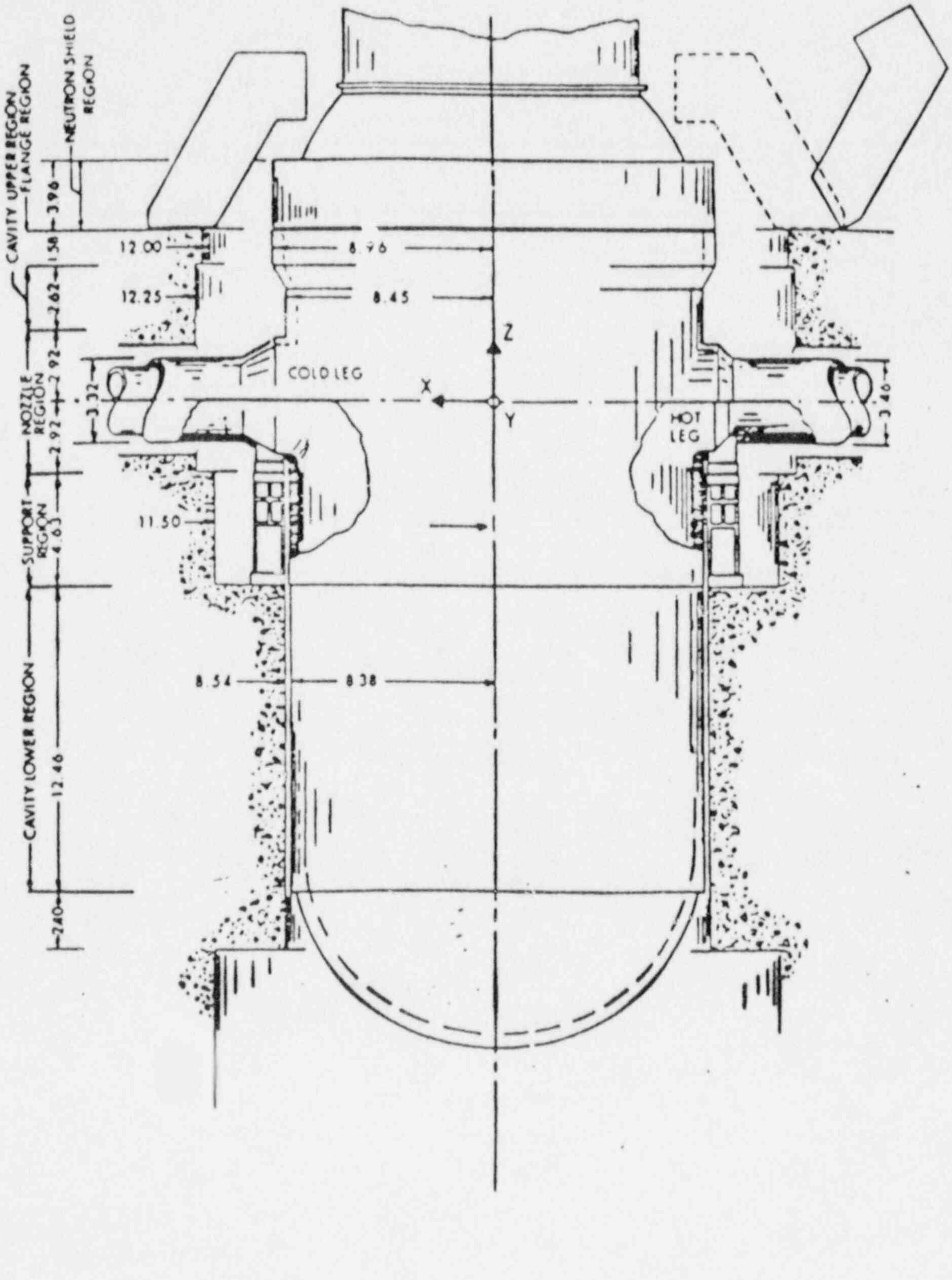


Fig. 5. PWR vessel with short-column support resting on biological shield.

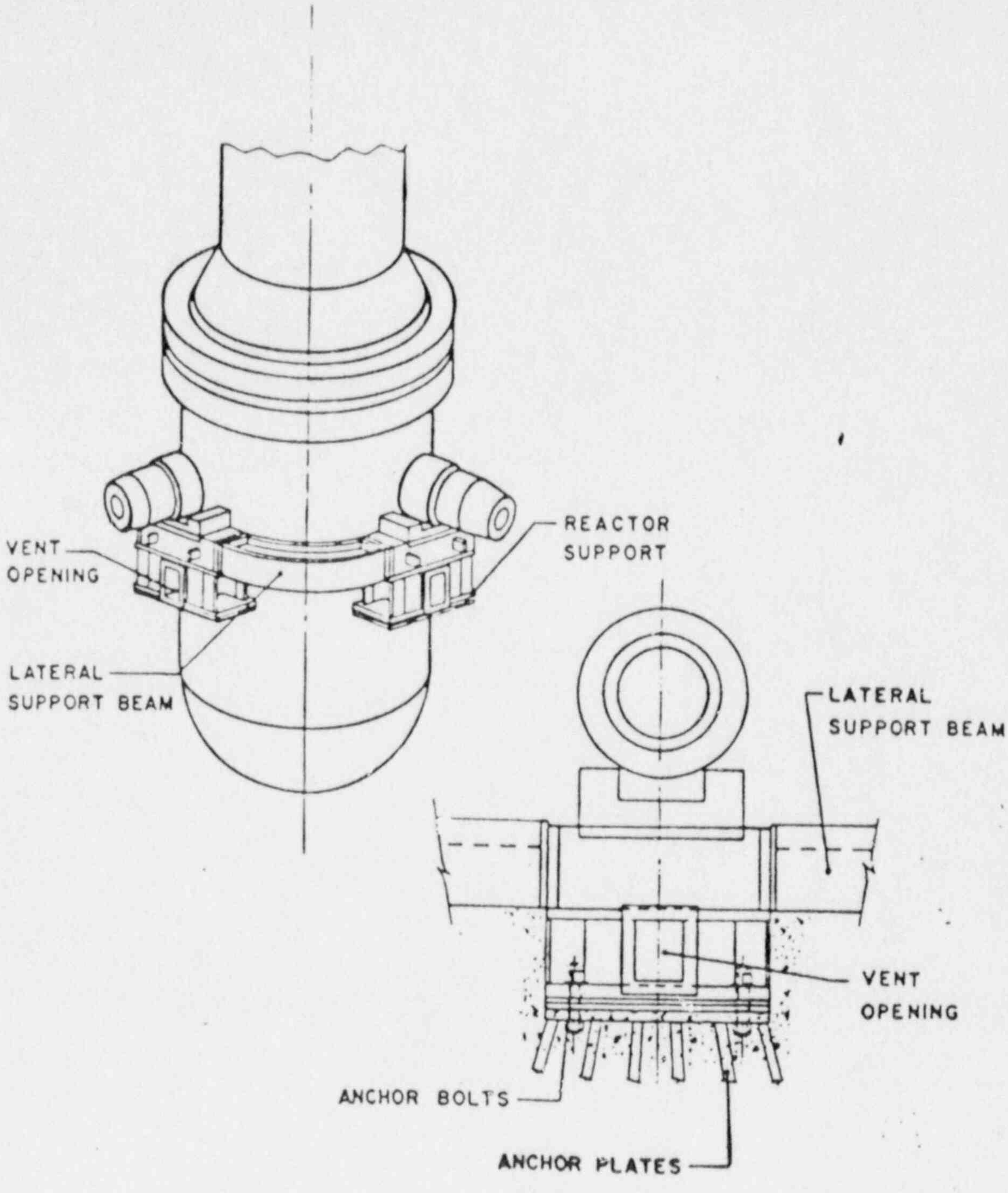


Fig. 6. PWR vessel with short-column support resting on biological shield: details.

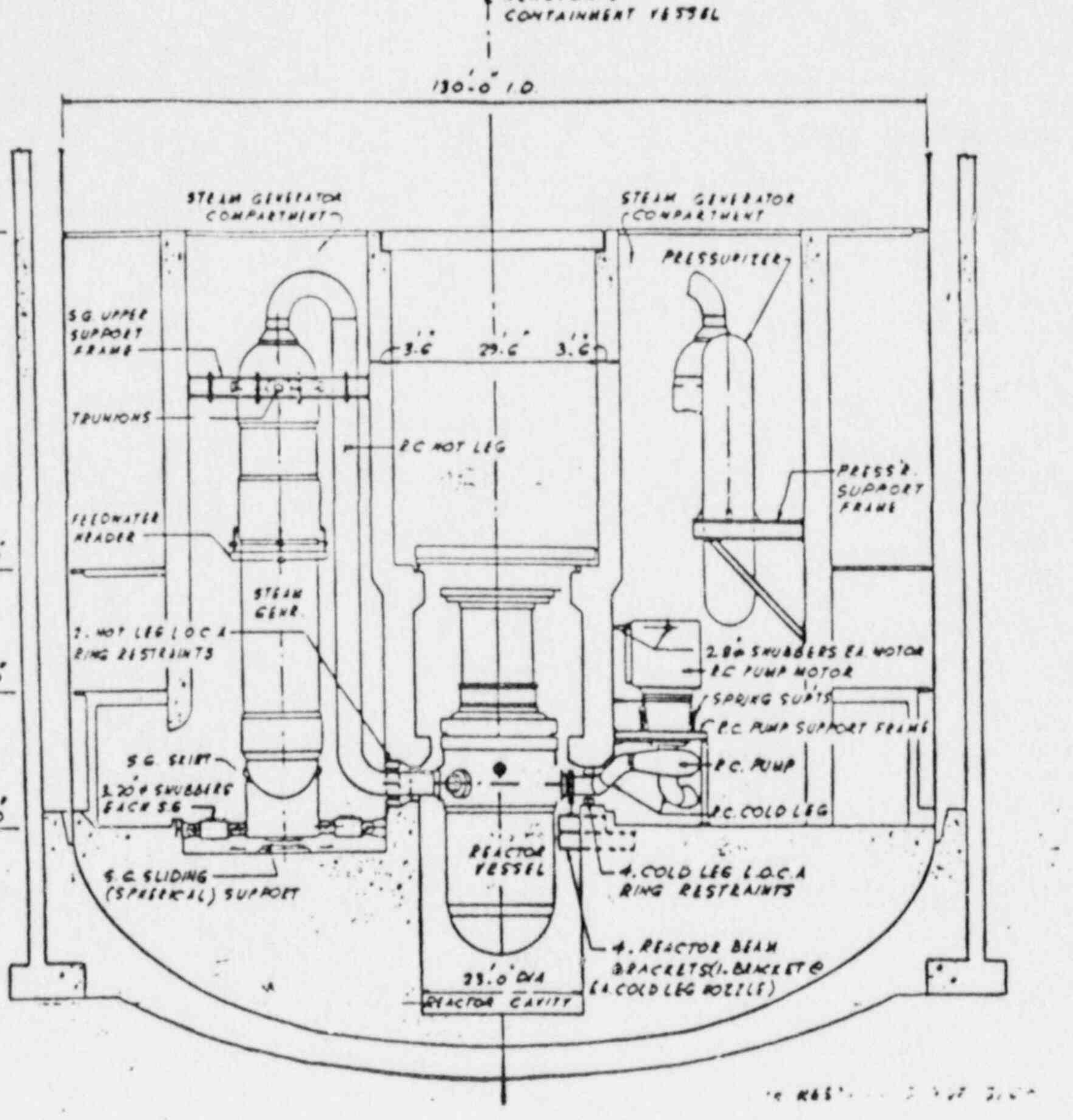


Fig. 7. PWR vessel with short-column support resting on cantilever beam.

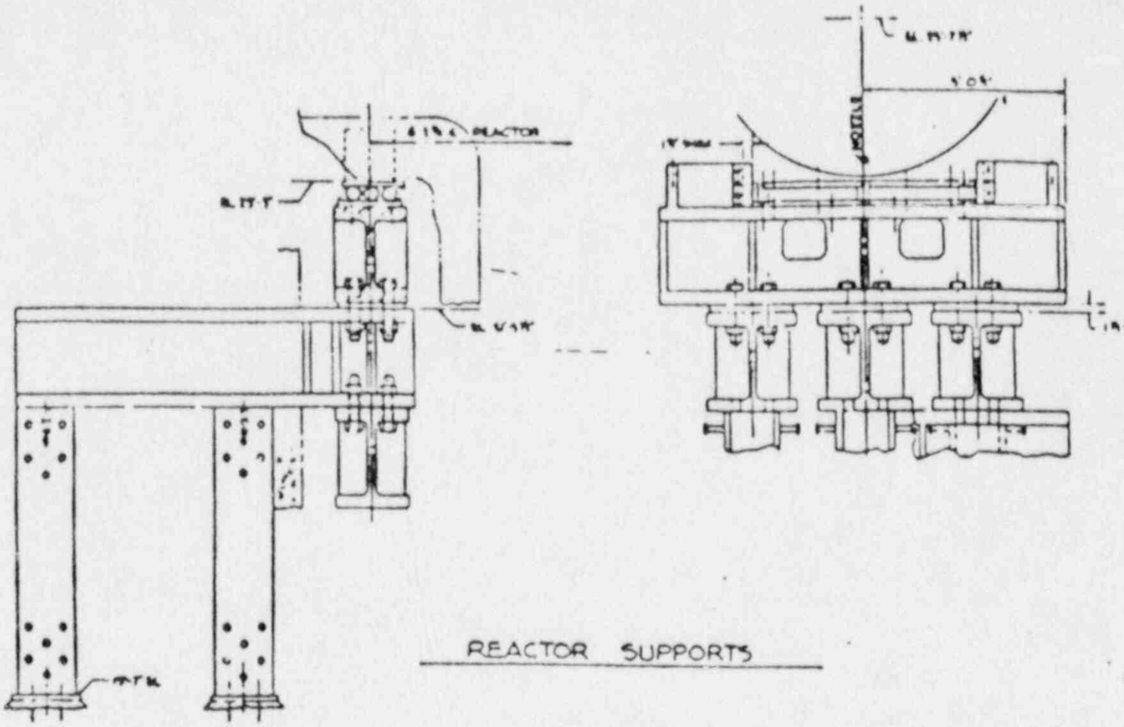


Fig. 8. Details of a PWR-vessel short-column support resting on cantilever beam.

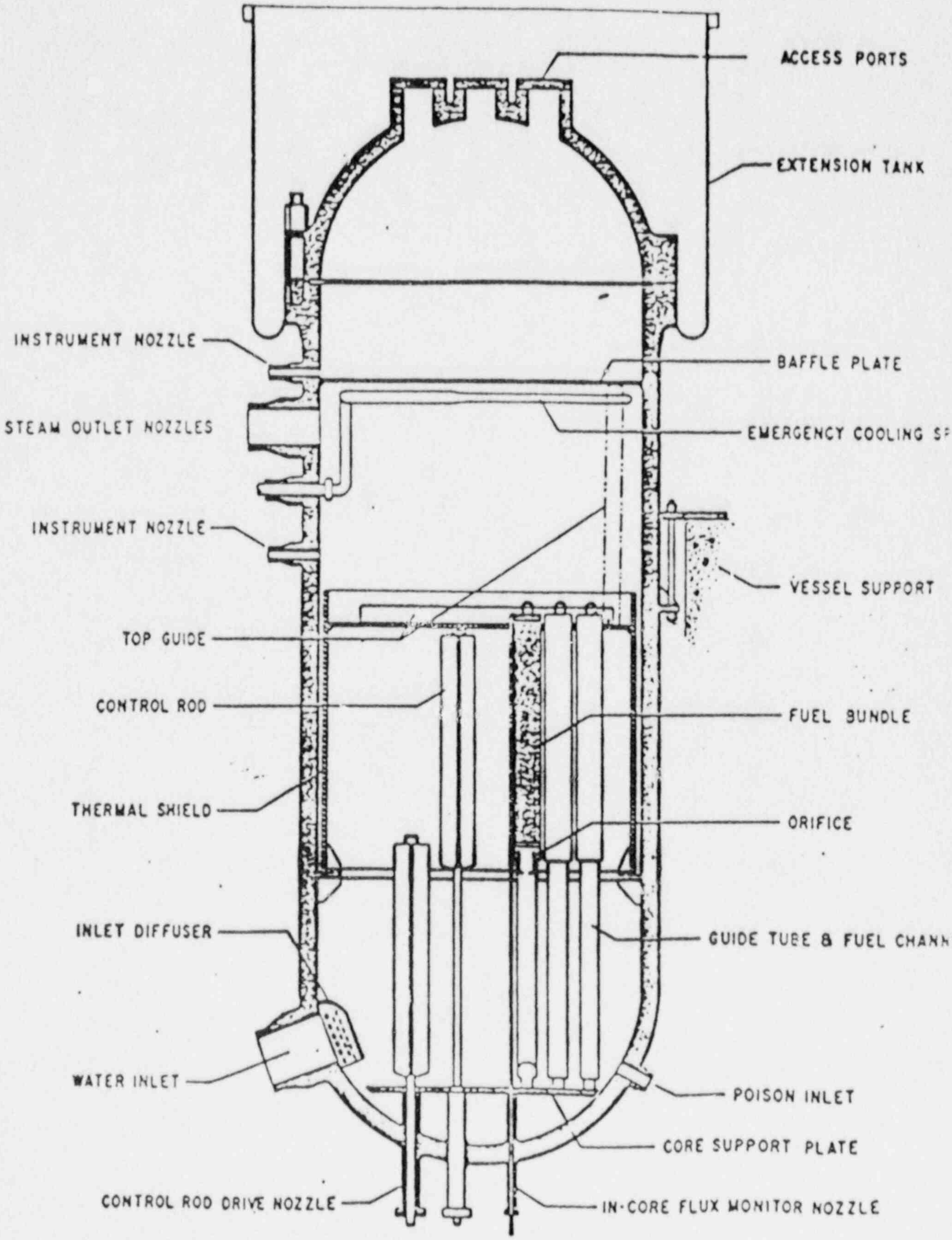


Fig. 9. BWR (Big Rock Point) vessel with suspension supports.

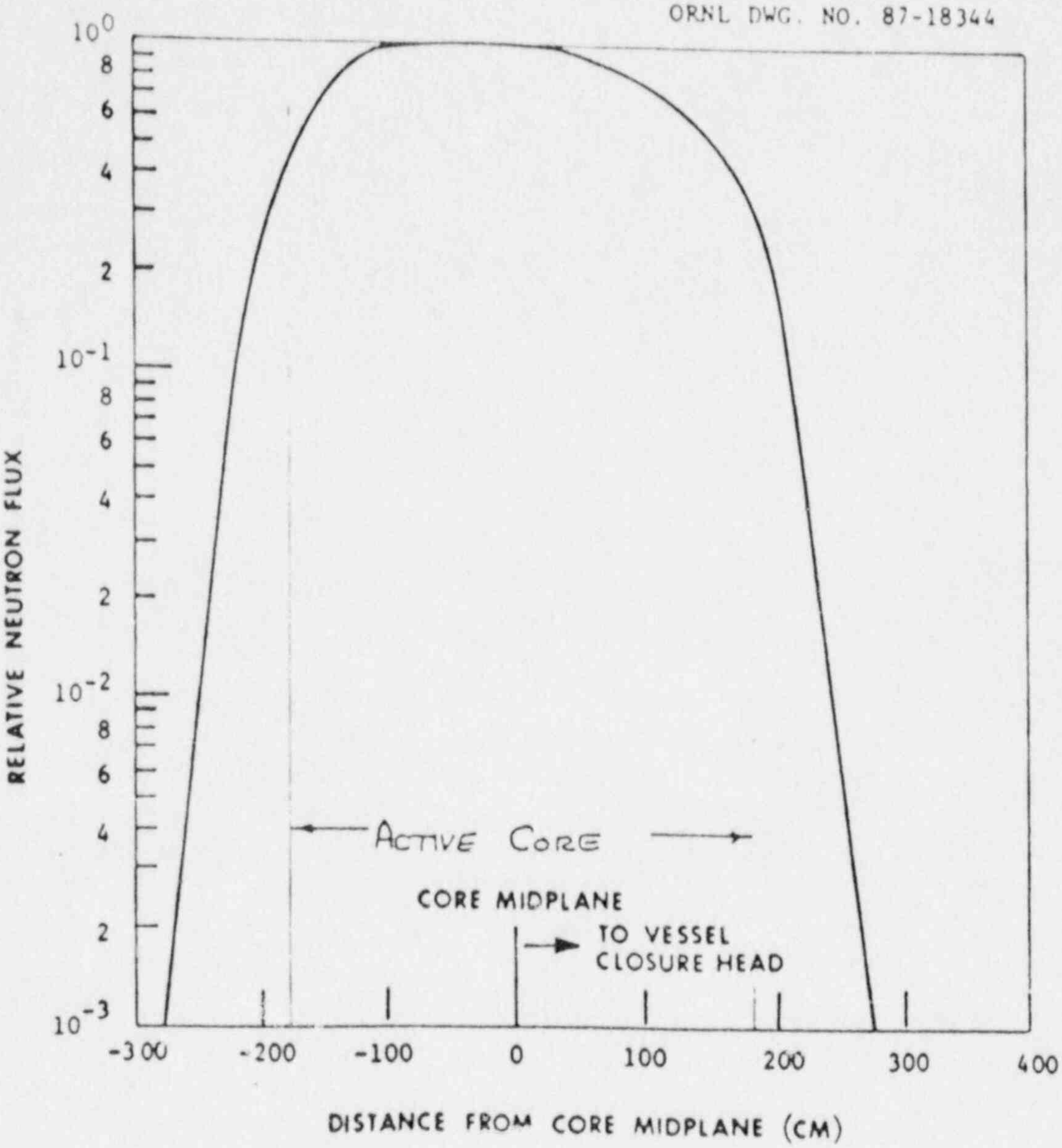


Fig. 10. Relative axial variation of fast neutron flux (E > 1.0 MeV) within the pressure vessel.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 15, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON THE EMBRITTLEMENT OF STRUCTURAL STEEL

Surveillance samples of steel used in the pressure vessel of the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory recently have shown that the nil-ductility transition (NDT) temperature of steel irradiated slowly at 120°F can rise much more rapidly with exposure to fast neutrons than would be expected from the available experimental work obtained in test reactors. This appears to be due to two causes:

- a flux rate effect (A lower fast neutron flux embrittles more than the same fluence accumulated at a much higher flux in test reactors.)

- the difference in temperature (550°F for commercial reactor pressure vessels vs. 120°F for the HFIR)

This has led to a significant shift in the NDT of the steel at a fast neutron fluence lower by roughly a factor of 20 than that predicted by the correlations used in the past for low temperature irradiations. This acceleration is independent of the copper content of the material. This suggests that steel structures outside the pressure vessel in commercial nuclear power plants may have embrittled where such behavior was not expected. We believe it would be prudent for the NRC to do the following:

1. Determine if the brittle failure of any structural steel component near the outside of the primary pressure boundary would have safety significance.

2. Determine, using the low temperature irradiation data now available from test reactors, whether an increase in the fast neutron fluence by a factor of 10-100 would be predicted to give brittle behavior in these components.

3. Implement a research program which would assemble better information on the rate of shift of the NDT of structural steels in

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commercial nuclear power plants at these lower rates and temperatures.

4. Include consideration of the accelerated shift in NDT as part of the evaluation of structures in the program on plant aging.

Sincerely,

W Kerr

William Kerr
Chairman

I DC 0C5005

AUG 04 1987

MEMORANDUM FOR: Dr. William Kerr, Chairman
Advisory Committee on Reactor Safeguards

FROM: Victor Stello, Jr.
Executive Director for Operations

SUBJECT: ACRS COMMENTS ON THE EMBRITTLEMENT OF STRUCTURAL STEEL

Your letter of July 15 on low temperature low-dose-rate irradiation of reactor structural steels correctly focused on the important issue emanating from the HFIR study that applies to operating commercial LWRs. That issue is whether the appearance of a dose rate effect for low temperature irradiation translates to a possible unexpected increase in the embrittlement of reactor vessel support structures (RVSS) that could pose a problem for safe continued operation of LWRs. We have therefore initiated actions to assemble information and make such analyses as necessary to make this assessment. The essential elements of the effort are shown in the enclosure. We have assigned the Heavy Section Steel Technology Program staff at Oak Ridge overall responsibility for the work, with important contributions to be provided by Argonne National Laboratory in the determination of mechanical properties of the Shippingport reactor neutron shield tank. Within NRC, the Office of Nuclear Regulatory Research will manage the effort. Oak Ridge believes that they can provide us with an initial response on the analysis portion of the work by September 4, with a more complete response by the end of September. We are proceeding to obtain material from the Shippingport reactor shield tank for validation as rapidly as the operations schedules at the reactor site will allow. Presently, we believe that it will require several additional months to gain access to the reactor and retrieve the material for machining and testing. Thus, the validation part of the effort could be delayed until the end of the year. In summary all the issues raised in your letter will be considered in the conduct of this safety assessment. I will be in direct communication with you concerning the results of the work.

Your letter helps to focus some assessments currently being made by the staff on embrittlement characteristics of those materials for aging and for extended service life. While we have been aware for many years of the higher embrittlement for steels irradiated at lower temperatures as was the HFIR vessel, it is only more recently that we have also begun to observe higher rates of embrittlement for vessel steels irradiated for longer times at lower dose rates. For example, only power reactor surveillance data are now being proposed as the basis for trend curves in Regulatory Guide 1.99, Revision 2 "Radiation Embrittlement of Reactor Vessel Materials" because they are a distinct body of data separate from test reactor data, and because they show more embrittlement than the test reactor data. Also, data obtained to date from the cooperative

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program with the MPA-Stuttgart in the Federal Republic of Germany on the Gundremmingen reactor vessel material also show somewhat more embrittlement than would be expected for the fluence received.

As you well know, the staff is actively engaged in preparing for evaluation of applications for significant extension of operating life beyond the original 40 year license period. One of the most important aspects of that evaluation is radiation embrittlement of the reactor vessel and other components subject to neutron flux during service. Because of the recent information suggesting the low-flux-induced dose-rate effect, this will be studied much more intently and incorporated into aging and extended plant life evaluations.

Original signed by
Victor Stello

Victor Stello, Jr.
Executive Director for Operations

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RADIATION EFFECTS ON REACTOR VESSEL SUPPORT STRUCTURES (RVSS)

Objective: To make a rapid assessment if there is a safety problem with RVSS subject to neutron radiation in consideration of a dose rate effect and low temperature irradiation.

Technical Issues:

- Flux and fluence in RVSS areas currently and at EOL
- Materials and mechanical properties of RVSS
- Embrittlement trends of RVSS materials
- Estimated dose rate effect and application to RVSS embrittlement trend
- Stresses and loadings on RVSS in normal operation and accidents
- Consequence to safety and safe shutdown of brittle failure of RVSS

Experimental Validation:

- Measure embrittlement of Shippingport shield tank material
    - Absolute NDT temperature
    - Transition temperature increase (using exterior tank material for reference)
    - Identify material composition and detailed chemistry
    - Determine fluence, flux and exposure history
    - Assure irradiation temperature

Integration and Assessment:

Combine material embrittlement trends with accident loadings to predict possible fracture of RVSS, and determine if such failure has an impact on safety and safe shutdown. Validate calculations and predictions using Shippingport material and data.

Initial Technical Assessment - ORNL	September 4, 1987
Final Technical Assessment	September 30, 1987
Shippingport Validation Data - ANL	December 31, 1987

OAK RIDGE NATIONAL LABORATORY

OPERATED BY MARTIN MARIETTA ENERGY SYSTEMS INC

POST OFFICE BOX Y
OAK RIDGE TENNESSEE 37831

September 2, 1987

Mr. C. Z. Serpan, Jr.
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
5650 Nicholson Lane
Rockville, Maryland 20852

Dear Mr. Serpan:

LWR Vessel Supports

In response to your request of July 30, 1987, via telecon, a task force was assembled to determine whether the higher-than-expected radiation damage rate of the HFIR vessel is a matter of concern with regard to the analysis of LWR vessel supports. A preliminary study has been completed, and the results are summarized herein.

The preliminary study included the following efforts:

1. Solicitation of help from B&W, CE, Westinghouse, EPRI, MEA and Bob Odette.

2. Examination of FSAR's.

3. Search for reports on vessel supports.

4. Acquisition of multigroup fluxes for cavity between vessel and biological shield.

5. Extrapolation of HFIR embrittlement data to LWR vessel supports.

6. Estimation of loads and stresses associated with vessel supports.

7. Application of above information to a preliminary assessment of the impact of the HFIR embrittlement data on the analysis of vessel supports.

At this time we have received information from B&W, EPRI, MEA and Odette.

Multigroup (26-group) fluxes for the cavity were available from Tsoulfanidis[1] (University of Missouri) for ANO-1 (cycle 4, B&W), ANO-2 (cycle 4, CE), and McQuire-1 (cycle 1, W) and from Williams[2] for a GE reactor (Tables 1-8). These fluxes correspond to the mid-height of the core and to, or nearly to, the maximum azimuthal position. The cycle-4 loadings for ANO-1 and ANO-2 are "low-leakage" and thus "low-cavity-flux" loadings.

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A typical axial distribution in the vessel wall for the fast flux ($E > 1$ MeV) is given in Fig. 1.[3]

The somewhat unique features of the HFIR vessel with regard to radiation embrittlement are the low flux levels (10^8-10^9 n/cm$^2 \cdot$s) and low temperature (~120°F). Comparison of the HFIR surveillance data with data for the same material (A212B) irradiated in the ORR at ~130°F, and with low-temperature test-reactor data compiled by Hawthorne, indicates a substantially greater embrittlement rate [ΔNDT/fluence ($E > 1$ MeV)] in HFIR than in the ORR and other test reactors. As shown in Figs. 2 and 3, the rate factors are ~10 and 20, respectively. As discussed in Ref. 4, the fast spectrums for HFIR and the other test reactors are essentially the same, but the HFIR vessel fast flux is about a factor of 10^4 less than for the test reactors involved. Thus, the difference in embrittlement rate is tentatively attributed to a rate effect.

As illustrated in Ref. 5, because of inelastic scattering in the vessel wall the fast spectrum in the cavity of an LWR is much softer than that in the core region of test reactors such as the ORR and at the inside surface of the LWR and HFIR vessel walls. Thus, as indicated by Odette, it was more appropriate to correlate the embrittlement data with dpa than fluence. Using the above multigroup fluxes and a set of dpa cross sections, Odette calculated the reactor cavity dpa reaction rates (Table 2,4,6 and 8). This was done assuming, of course, that a vessel support existed in the cavity but ignoring the flux perturbation in the support.

A typical dpa rate for a HFIR-vessel surveillance specimen is 3×10^{-13} dpa/s (Ref. 4, Appendix B). This compares with LWR vessel-cavity values of 7×10^{-13}, 5×10^{-12}, 5×10^{-12} and 6×10^{-14} for the B&W, CE, W and GE reactors, respectively, being considered. Thus, the rate effects for the CE and W reactors would tend to be less than for HFIR. Of course, for vertical positions away from mid-height of the core, for which these values apply, the rates are less and thus there is some position where the rates are the same. Even so, the total dpa, and thus damage, would be less.

Another comparison that must be made between the HFIR vessel and the LWR vessel supports is the normal operating temperature. As mentioned above, the HFIR vessel (and also the surveillance specimens) operate at ~120°F. Many of the LWR vessel supports also operate at about this temperature. Thus, it appears that at least to some extent the HFIR-vessel surveillance data are applicable to the LWR vessel supports.

To obtain an estimate of ΔNDT for the LWR vessel supports, Odette compiled a set of ΔNDT vs dpa data for materials judged to be similar in radiation embrittlement behavior to the support materials and which were irradiated in test reactors at temperatures <200°F. He defined mean and upper-bound curves, and then, using the dpa rates in Tables 2, 4, 6 and 8, calculated 32 EFPY values of ΔNDT for no rate effect and a rate effect obtained by multiplying the abscissa of his ΔNDT vs dpa plot by 10^{-1}. For those vertical

locations corresponding to dpa > 3 x 10^{-13}, the ΔNDT values tend to be an overestimate because, presumably, the rate effect would be less.

Results of Odette's estimates are presented in Table 9. As indicated, the shift in NDT for the BWR is very small but for the other is quite large (175-220°F for B&W; 380-450°F for CE and W) when the rate effect is included.

Assuming that the ASME K_{Ic} vs T - RT_{NDT} curve is appropriate for the vessel-support materials, it is apparent that with the indicated rate effect included, the supports, if located within the active height of the core will be on the lower shelf of the fracture-toughness curve. Thus, the maximum permissible applied K_I value is ~30 ksi $\sqrt{in.}$

As indicated in Appendix A, t.ere are numerous vessel support concepts and designs. The long-column and sh.eld-tank supports extend the length of the core and thus are exposed to the maximum fluxes, while skirt designs (GE and B&W) and some of those for which the nozzles effectively rest on the concrete biological shield preclude large doses in the supports. It appears that the NDT shift for these latter supports is negligible, but for the former it is not.

A stress analysis of the supports receiving significant dose is beyond the scope of this preliminary study. However, ORNL performed a simplified analysis for a typical shield-tank-type support. The results indicated that the "total" stresses associated with a "static" 0.2 g seismic event would still be compressive. However, tensile stresses corresponding to a large-break LOCA could be substantial. It is our understanding, though, that a large-break LOCA is not consiaered credible.

The ORNL calculations did not include thermal streses due to possible steady-state temperature gradients in the supports (550-100°F) or residual stresses in welds that might not have been stress relieved. These stresses can be substantial and must be considered.

In summary, it is clear that plant-specific data are required for an accurate evaluation of the potential for LWR vessel support failure. If supports initially have high tensile stresses due to bending (cantilever-type supports), thermal gradients and/or welding, there could be a problem if the estimated large shifts in NDT apply to these particular plants.

Sincerely,

Dick

R. D. Cheverton

RDC:spf
Attachments (2)
cc: See page 4

cc: R. H. Bryan
W. R. Corwin
F. K. Kam
A. P. Malinauskas
M. E. Mayfield, NRC
J. G. Merkle
F. R. Mynatt
R. K. Nanstad
C. E. Pugh
G. C. Robinson
H. E. Trammell
M. Vagins, NRC

References:

1. Nicolas Tsoulfanidis et al., "Neutron Energy Spectrum Calculations in Three PWR," Proceedings of the Fifth ASTM-EURATOM Symposium on Reactor Dosimetry, pp. 693-701 (1985).

2. M. L. Williams (private communication).

3. Project Topical Report for Surry Unit No. 1 - Life Extension Evaluation of the Reactor Vessel, Westinghouse Electric Corporation.

4. R. D. Cheverton, J. G. Merkle and R. K. Nanstad, Evaluation of HFIR Pressure-Vessel Integrity Considering Radiation Embrittlement, ORNL/TM-10444, draft of May 1, 1987.

5. J. R. Hawthorne and J. A. Sprague, Report by Task C of Interagency Agreement No. NRC-03-79-148, "Radiation Effects to Reactor Vessel Support Structures," Encl. (1) to NRL Memo 6392-291M:JRH:jw, 22 October 1979.

Table 1.

Calculated neutron energy spectra for B&W (ANO-1) Reactor[1]

3-D DOT FLUX ($n/cm^2 \cdot s$)

Group	E upper (MeV)	In Front of PV	At T/4 of PV	In Cavity
1	1.733+01	2.699+07	1.258+07	5.175+05
2	1.221+01	6.543+07	2.940+07	1.061+06
3	1.000+01	2.743+08	1.205+08	3.694+06
4	7.408+00	7.879+08	3.242+08	7.908+06
5	4.966+00	1.133+09	4.882+08	1.348+07
6	3.012+00	7.621+08	3.822+08	1.244+07
7	2.466+00	3.448+08	1.794+08	5.967+06
8	2.307+00	1.376+09	9.010+08	3.893+07
9	1.653+00	2.000+09	1.688+09	1.176+08
10	1.003+00	1.108+09	9.956+08	1.036+08
11	7.427-01	1.863+09	2.314+09	3.413+08
12	4.979-01	2.052+09	3.036+09	5.289+08
13	2.972-01	1.171+09	1.214+09	2.908+08
14	1.832-01	1.235+09	1.643+09	3.327+08
15	1.111-01	8.378+08	1.303+09	2.556+08
16	6.738-02	8.851+08	6.444+08	1.953+08
17	3.183-02	1.820+08	7.730+07	2.832+07
18	2.606-02	3.043+08	4.703+08	1.094+08
19	2.418-02	6.040+08	6.469+08	2.004+08
20	1.503-02	6.938+08	3.110+08	1.652+08
21	7.102-03	2.697+09	1.528+09	4.918+08
22	4.540-04	1.489+09	8.032+08	2.364+08
23	1.013-04	3.968+09	1.807+09	4.823+08
24	1.855-06	1.272+09	3.777+08	1.309+08
25	4.140-07	9.277+09	2.232+08	5.184+08
26	1.000-11			
Total		3.641+10	2.152+10	4.613+09

Table 2. 4-Group Fluxes* for B&W (ANO-1) PWR

Group	Energy Range	OT	1/4T	Cavity
1	1.0 MeV - 17 MeV	.186	.192	.044
2	0.111 MeV - 1.0 MeV	.204	.428	.346
3	0.4 eV - 0.111 MeV	.355	.370	.498
4	10^{-5} eV - 0.4 eV	.255	.010	.112
	dpa/s	1.041E-11	7.232E-12	6.629E-13

*normalized to unity

Table 3.

Calculated neutron energy spectra for Combustion Engineering (ANO-2) Reactor

3-D DOT FLUX ($n/cm^2 \cdot s$)

Group	E upper (MeV)	In Front of PV	At T/4 of PV	In Cavity
1	1.733+01	7.373+07	3.189+07	1.622+06
2	1.221+01	2.330+08	9.728+07	4.384+06
3	1.000+01	1.265+09	5.148+08	1.990+07
4	7.408+00	4.565+09	1.724+09	5.359+07
5	4.966+00	7.486+09	2.910+09	9.773+07
6	3.012+00	5.313+09	2.383+09	9.108+07
7	2.466+00	2.438+09	1.129+09	4.406+08
8	2.307+00	9.928+09	5.725+09	2.754+08
9	1.653+00	1.492+10	1.104+10	8.080+08
10	1.003+00	8.237+09	6.600+09	7.108+08
11	7.427-01	1.460+10	1.602+10	2.258+09
12	4.979-01	1.567+10	2.041+10	3.328+09
13	2.972-01	8.603+09	8.400+09	1.932+09
14	1.832-01	9.282+09	1.138+10	2.164.09
15	1.111-01	6.773+09	8.073+09	1.585+09
16	6.738-02	6.212+09	4.538+09	1.320+09
17	3.183-02	9.359+08	4.206+08	2.006+08
18	2.606-02	2.570+09	3.218+09	7.005+08
19	2.418-02	4.319+09	4.496+09	1.320+09
20	1.503-02	4.580+09	2.194+09	1.119+09
21	7.102-03	1.829+10	9.711+09	3.355+09
22	4.540-04	9.975+09	4.705+09	1.626+09
23	1.013-04	2.731+10	1.172+10	3.458.09
24	1.855-06	8.374+09	2.297+09	9.823+08
25	4.140-07	5.962+10	1.753+09	4.648+09
26	1.000-11			
27				
Total		2.516+11	1.415+11	3.254+10

Table 4. 4-Group Fluxes for CE (ANO-2) Reactor

Group	Energy Range	OT	1/4T	Cavity
1	1.0 MeV - 17 MeV	.184	.181	.055
2	0.111 MeV - 1.0 MeV	.224	.444	.319
3	0.4 eV - 0.111 MeV	.355	.363	.483
4	10^{-5} eV - 0.4 eV	.237	.012	.143
	dpa/s	7.041E-11	4.549E-11	4.975E-12

*normalized to unity

Table 9

Transition Temperature Shift Estimates for 32 EFPY Operation (Odette)

Vendor	Nom./Max.	dpa	ΔT (C)[1]	ΔT (C)[2]
BW	Nom.	6.9×10^{-4}	0	80
	Max.		9	103
GE	Nom.	6.3×10^{-5}	0	0
	Max.		0	9
W	Nom.	4.7×10^{-3}	65	195
	Max.		90	230
CE	Nom.	5.0×10^{-3}	70	200
	Max		95	235

1) Without flux correction; 2) With flux correction.

Table 8. 4-Group Fluxes* for GE Reactor

Group	Energy Range	Core Boundary	Mid-Downcomer	OT	1/4T	Cavity
1	1.0 MeV - 17 MeV	.096	.149	.158	.234	.080
2	0.111 MeV - 1.0 MeV	.314	.099	.123	.419	.313
3	0.4 eV - 0.111 MeV	.460	.237	.235	.333	.478
4	10^{-5} eV - 0.4 eV	.130	.515	.484	.014	.129
	dpa/s			6.909E-13	4.303E-13	6.335E-14

*normalized to unity

Table 5.

Calculated neutron energy spectra for Westinghouse Reactor[1]
McQuire Unit 1, Cycle 1

[illegible] DOT FLUX

Group	E upper (MeV)	In Front of PV	At T/4 of PV	In Cavity R=323.83 cm
1	1.733+01	5.062+07	2.105+07	3.020+05
2	1.221+01	1.567+08	6.249+07	7.764+05
3	1.000+01	8.316+08	3.236+08	3.293+06
4	7.408+00	2.893+09	1.048+09	8.135+06
5	4.966+00	4.424+09	1.662+09	2.014+07
6	3.012+00	3.001+09	1.331+09	2.193+07
7	2.466+00	1.360+09	6.255+08	1.135+07
8	2.307+00	5.295+09	3.126+09	9.912+07
9	1.653+00	7.603+09	5.901+09	4.448+08
10	1.003+00	4.142+09	3.448+09	4.510+08
11	7.427-01	6.947+09	8.317+09	2.546+09
12	4.979-01	7.404+09	1.087+10	4.565+09
13	2.972-01	4.298+09	4.316+09	2.449+09
14	1.832-01	4.468+09	6.069+09	3.419+09
15	1.111-01	3.380+09	4.375+09	2.604+09
16	6.738-02	3.362+09	2.360+09	2.024+09
17	3.183-02	6.466+08	2.369+08	3.793+08
18	2.606-02	1.153+09	1.697+09	1.490+09
19	2.418-02	2.209+09	2.410+09	2.812+09
20	1.503-02	2.657+09	1.170+09	1.946+09
21	7.102-03	1.016+10	5.142+09	5.942+09
22	4.540-04	5.600+09	2.547+09	2.867+09
23	1.013-04	1.473+10	5.865+09	5.993+09
24	1.855-06	4.670+09	1.151+09	1.680+09
25	4.140-07	3.382+10	7.968+08	7.164+09
26	1.000-11			
27				
Total		1.353+11	7.487+10	4.894+10

Table 6. 4-Group Fluxes* for W Reactor

Group	Energy Range	OT	1/4T	Cavity
1	1.0 MeV - 17 MeV	.184	.188	.012
2	0.111 MeV - 1.0 MeV	.202	.441	.274
3	0.4 eV - 0.111 MeV	.359	.360	.56[illegible]
4	10^{-5} eV - 0.4 eV	.250	.011	.146
	dpa's	[illegible]	[illegible]	[illegible]

*normalized to unity

Table 7.

Calculated neutron energy spectra for GE Reactor[1]

Group	E upper (MeV)	In Front of PV	At 1/4T of PV	In Cavity
1	1.733E+01	2.78532E+05	1.19476E+05	1.26309E+04
2	1.419E+01	1.17743E+06	5.10065E+05	5.31195E+04
3	1.221E+01	4.20585E+06	1.73173E+06	1.64842E+05
4	1.000E+01	7.86872E+06	3.22077E+06	2.92429E+05
5	8.607E+00	1.27925E+07	5.10243E+06	4.31247E+05
6	7.408E+00	2.97338E+07	1.15308E+07	8.93362E+05
7	6.065E+00	3.82946E+07	1.42439E+07	1.06782E+06
8	4.966E+00	5.75978E+07	2.15559E+07	1.67350E+06
9	3.679E+00	3.59718E+07	1.51213E+07	1.29333E+06
10	3.012E+00	2.41009E+07	1.12497E+07	9.89848E+05
11	2.725E+00	2.59448E+07	1.28555E+07	1.19784E+06
12	2.466E+00	1.25811E+07	6.32145E+06	6.21796E+05
13	2.365E+00	3.22105E+06	1.78435E+06	1.76455E+05
14	2.346E+00	1.50974E+07	8.68747E+06	8.79155E+05
15	2.231E+00	3.58901E+07	2.23694E+07	2.14032E+06
16	1.920E+00	3.56560E+07	2.62479E+07	2.89304E+06
17	1.653E+00	4.59384E+07	3.65276E+07	4.35308E+06
18	1.353E+00	6.38427E+07	6.42605E+07	9.28453E+06
19	1.003E+00	4.05210E+07	4.46711E+07	7.60235E+06
20	8.208E+01	2.11290E+07	1.92272E+07	4.04310E+06
21	7.427E+01	4.78865E+07	6.94558E+07	1.34120E+07
22	6.081E+01	4.06699E+07	5.56024E+07	1.30706E+07
23	4.979E+01	4.48645E+07	6.27020E+07	1.33038E+07
24	3.688E+01	4.04880E+07	6.98931E+07	1.48399E+07
25	2.972E+01	6.22632E+07	7.57791E+07	2.24843E+07
26	1.832E+01	5.40647E+07	7.47653E+07	2.27497E+07
27	1.111E-01	4.23644E+07	4.61578E+07	1.65013E+07
28	6.738E-02	3.61008E+07	3.55079E+07	1.25341E+07
29	4.087E-02	1.56710E+07	9.46462E+06	4.06071E+06
30	3.183E-02	1.18315E+07	2.92956E+06	4.08343E+06
31	2.606E-02	7.64521E+06	1.84057E+07	5.40984E+06
32	2.418E-02	7.29383E+06	1.11433E+07	3.93861E+06
33	2.188E-02	2.23029E+07	1.78285E+07	8.46670E+06
34	1.503E-02	4.12070E+07	2.00703E+07	1.19762E+07
35	7.102E-03	4.04597E+07	2.52165E+07	1.15085E+07
36	3.355E-03	3.90943E+07	1.99575E+07	1.03076E+07
37	1.585E-03	6.33659E+07	3.07840E+07	1.57353E+07
38	4.540E-04	3.75323E+07	1.35174E+07	8.53409E+06
39	2.144E-04	3.76390E+07	1.74130E+07	8.29094E+06
40	1.013E-04	5.01136E+07	2.50163E+07	1.05576E+07
41	3.727E-05	6.23387E+07	3.06902E+07	1.23638E+07
42	1.068E-05	3.71304E+07	1.64081E+07	6.92802E+06
43	5.043E-06	4.87669E+07	1.80571E+07	8.48279E+06
44	1.855E-06	3.58256E+07	9.95936E+06	5.74201E+06
45	8.764E-07	3.49526E+07	6.91034E+06	5.16106E+06
46	4.140E-07	1.19921E+08	6.44878E+06	1.10811E+07
47	1.000E-07	1.25993E+09	9.40594E+06	3.51009E+07
48	1.000E-11			
Total		2.8536E+09	1.1263E+09	3.3669E+08

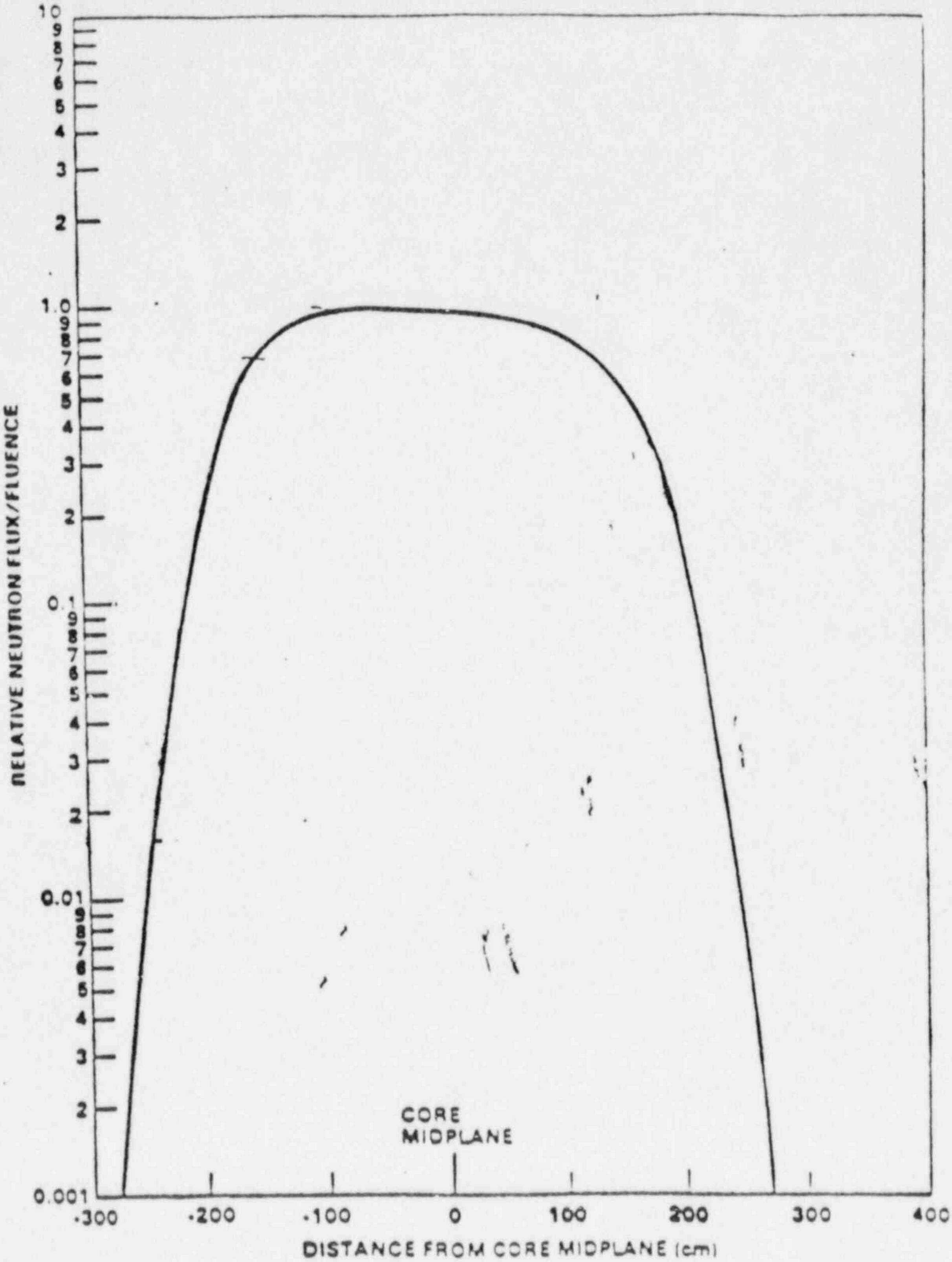


FIGURE 1 Relative Axial Variation of Fast Neutron (E > 1.0 MeV) Flux and Fluence Within the Pressure Vessel Wall

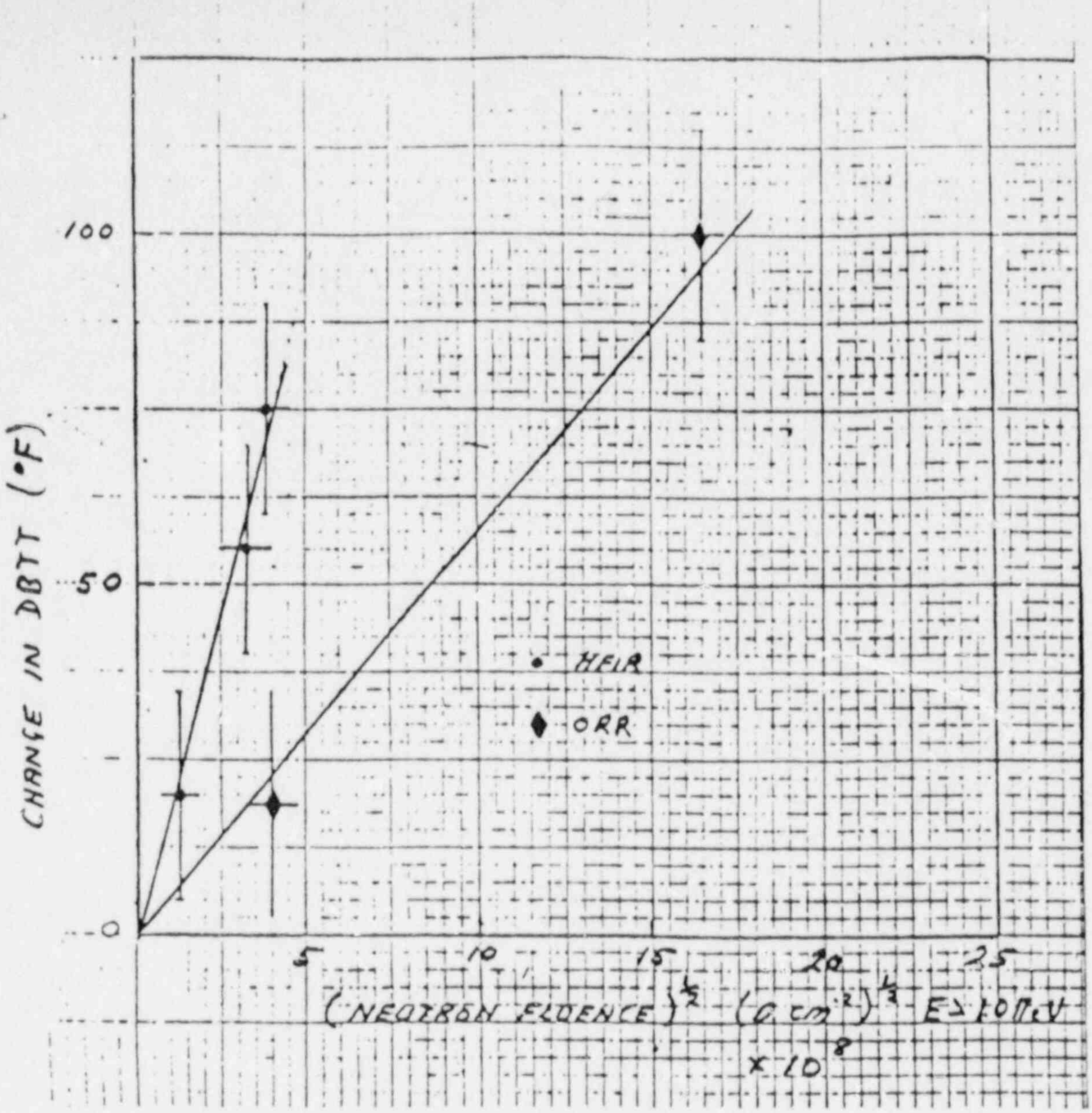


Fig. 2 Change in DBTT versus the square root of fluence for A212B HFIR surveillance specimens and for archive specimens irradiated in the ORR.

ORNL-DWG 87-8154

DIFFERENCE IN HFIR VESSEL AND TEST-REACTOR IN-CORE DAMAGE RATES APPARENTLY DUE TO LOWER FLUENCE RATE IN HFIR

● ϕ_F (TEST REACTOR) $\simeq 10^4 \times \phi_F$ (HFIR)

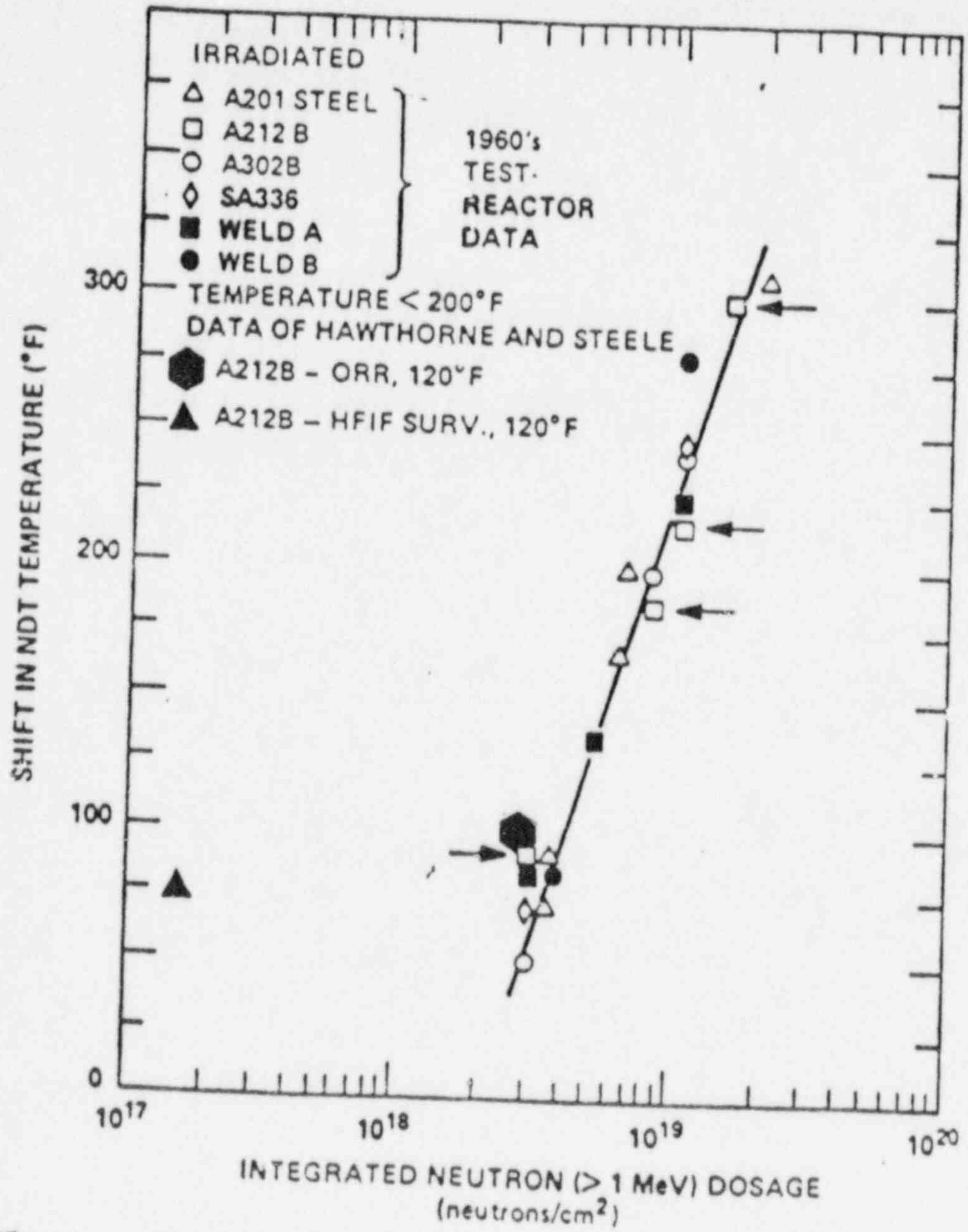


FIG 3

Correlation of transition temperature shifts with total neutron exposure (> 1.0 MeV) of materials irradiated under 200°F.

(J. R. Hawthorne)

APPENDIX A

DESCRIPTION OF REACTOR VESSEL SUPPORTS

The following brief description of reactor vessel supports is premised on the description provided in the Final Safety Analysis Reports.

General Electric BWR Plants

Two of the early General Electric plants suspended the reactor vessel from the shield wall; that is, Big Rock Point and Humboldt Bay, as shown in Fig. A.1. All subsequent General Electric plants that were reviewed used skirt support arrangements typically as shown in Figs. A.2 through A.5.

Babcock and Wilcox PWR Plants

Early B&W reactor vessel designs incorporated skirt supports as shown in Fig. A.6. Later plants incorporated the design features shown for the B&W standard plant; that is, Babcock 241, shown on Figs. A.7 and A.8. In the Babcock 241 arrangement the weight and vertical loads are transmitted from the vessel nozzles to structural weldments resting on the shield walls. The horizontal loads are resisted at two elevations by structural weldments mounted on the shield wall. The structural weldments are designed to permit convection cooling in order that the portion in contact with the shield wall not exceed approximately 150°F. The AE support for Babcock 241 type plants may depart considerably from that shown in Figs. A.7 and A.8. For example, the Davis Besse plant uses cantilevered beam brackets to support the reactor vessel at four nozzles shown in Figs. A.9 and A.10. Similarly, the Bellefonte plant uses a haunch (structural weldment or reinforced concrete) canti-levered out from the shield wall under the reactor vessel nozzles as shown in Fig. A.11.

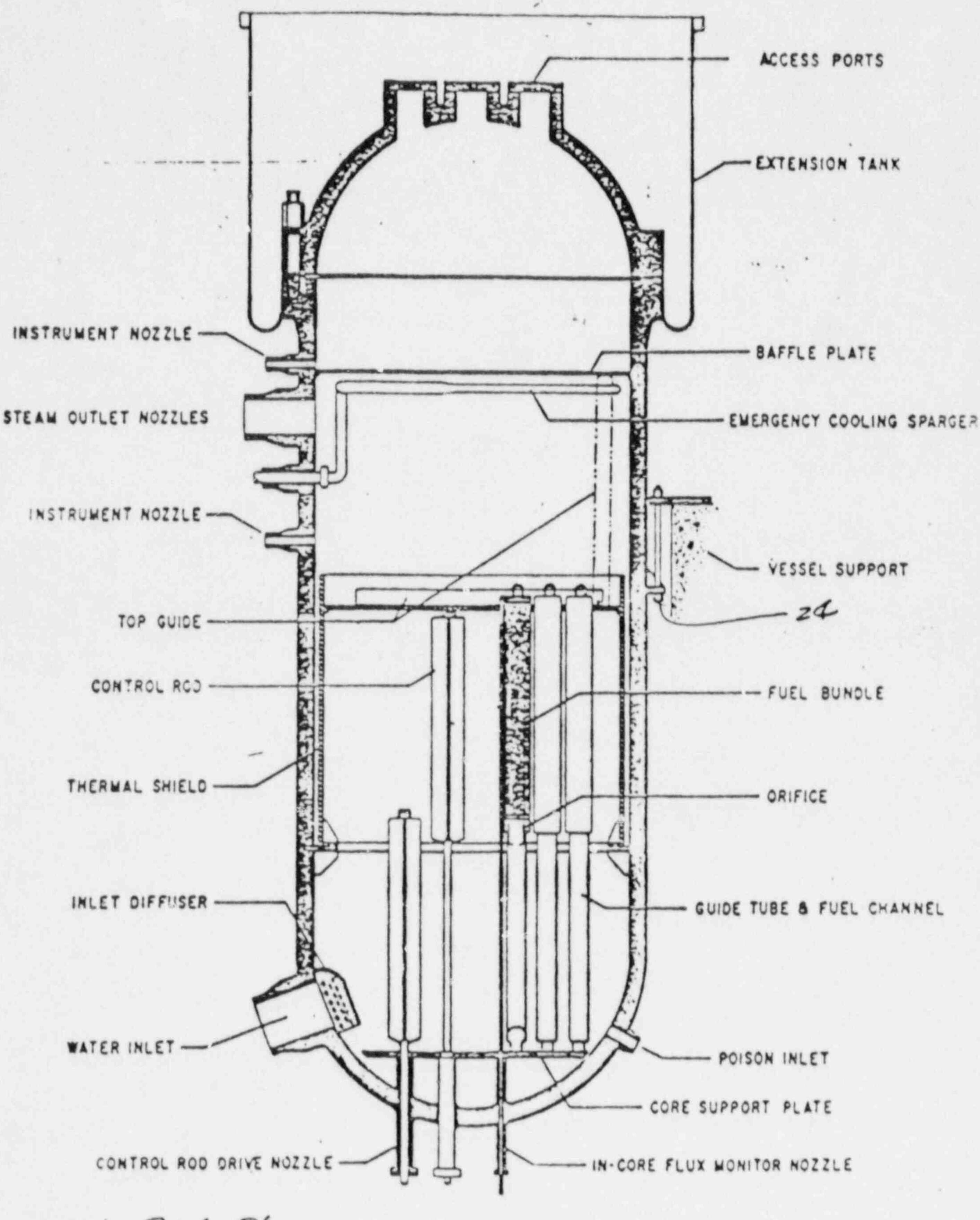
Combustion Engineering PWR Plants

All of the CE plants support the reactor vessel from the nozzles with a welded pad arrangement similar to that shown in Fig. A.12. The receiving support design varies depending on the AE designing the plant. For example, a structural weldment on a haunch cantilevered from the shield wall is shown in Fig. A.13 for Palisades; a neutron shield tank support is shown in Fig. A.14 for Maine Yankee; a structural weldment on the shield wall is shown in

Fig. A.15 for Calvert Cliffs 1 and 2; a structural weldment in a pocket of the shield wall is shown in Fig. A.16 for Millstone Point; and columns underneath the nozzles extending to the shield floor are shown in Fig. A.17 for WPPS 3 and 5 and in Figs. A.18 and A.19 for Arkansas Nuclear 1. The column arrangement is that depicted by the CE System 80 support arrangement shown in Figs. A.20 and A.21.

Westinghouse Electric PWR Plants

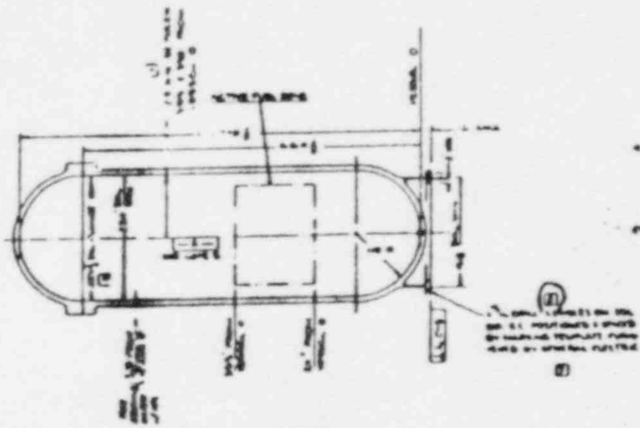
The Yankee Rowe reactor vessel used support lugs welded on the cylindrical shell below the nozzles to transfer vertical loads to the shield wall as shown in Fig. A.22. Subsequent designs used welded pads on nozzles as load points as shown in Fig. A.23. The receiving support designs are widely divergent depending on the AE designer. Maine Yankee and Surry, Figs. A.24 and A.25, use neutron shield tanks for vessel support. Cantilevered beam brackets embedded in the shield wall are used for Turkey Point 3 and 4 as shown in Fig. A.26. A variety of structural weldments that transfer loads from the nozzles to the shield wall are shown in Figs. A.27 through A.32. It is generally implied for these designs that the rebar in the reinforced concrete of the shield walls resists the imposed horizontal loads. However, Seabrook, Figs. A.30 and A.31, used a structural steel ring girder anchored to a step in the shield wall to resist directly the horizontal loading. In all of the designs shown in Figs. A.27 through A.33, the structural elements are openly arranged to permit convection cooling for protection of the reinforced concrete so that a temperature gradient from approximately 550 to 150°F exists in the structure. These designs reflect the features shown in the Westinghouse SNUPPS standard design shown in Fig. A.33.



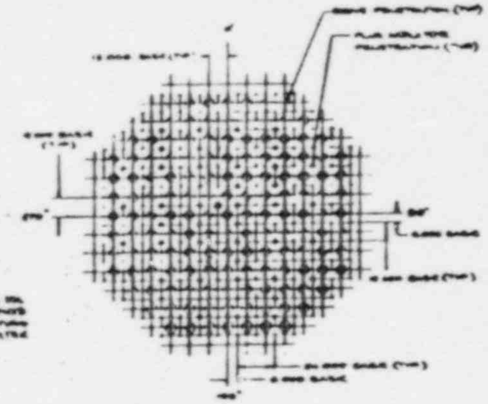
Big Rock Pt.
50-155

REACTOR VESSEL SCHEMATIC
GE

FIGURE A 1



LONGITUDINAL SECTION



BOTTOM HEAD PLAN VIEW

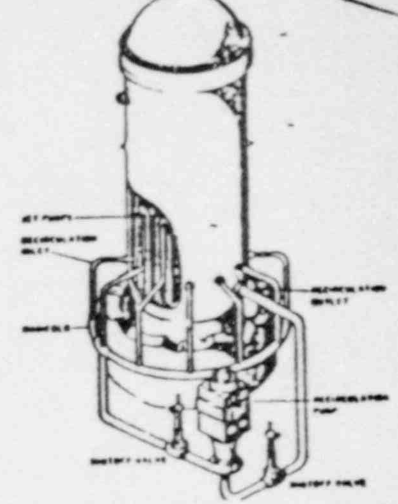


FIGURE IV-2.2. REACTOR VESSEL ISOMETRIC

NOZZLE SCHEDULE

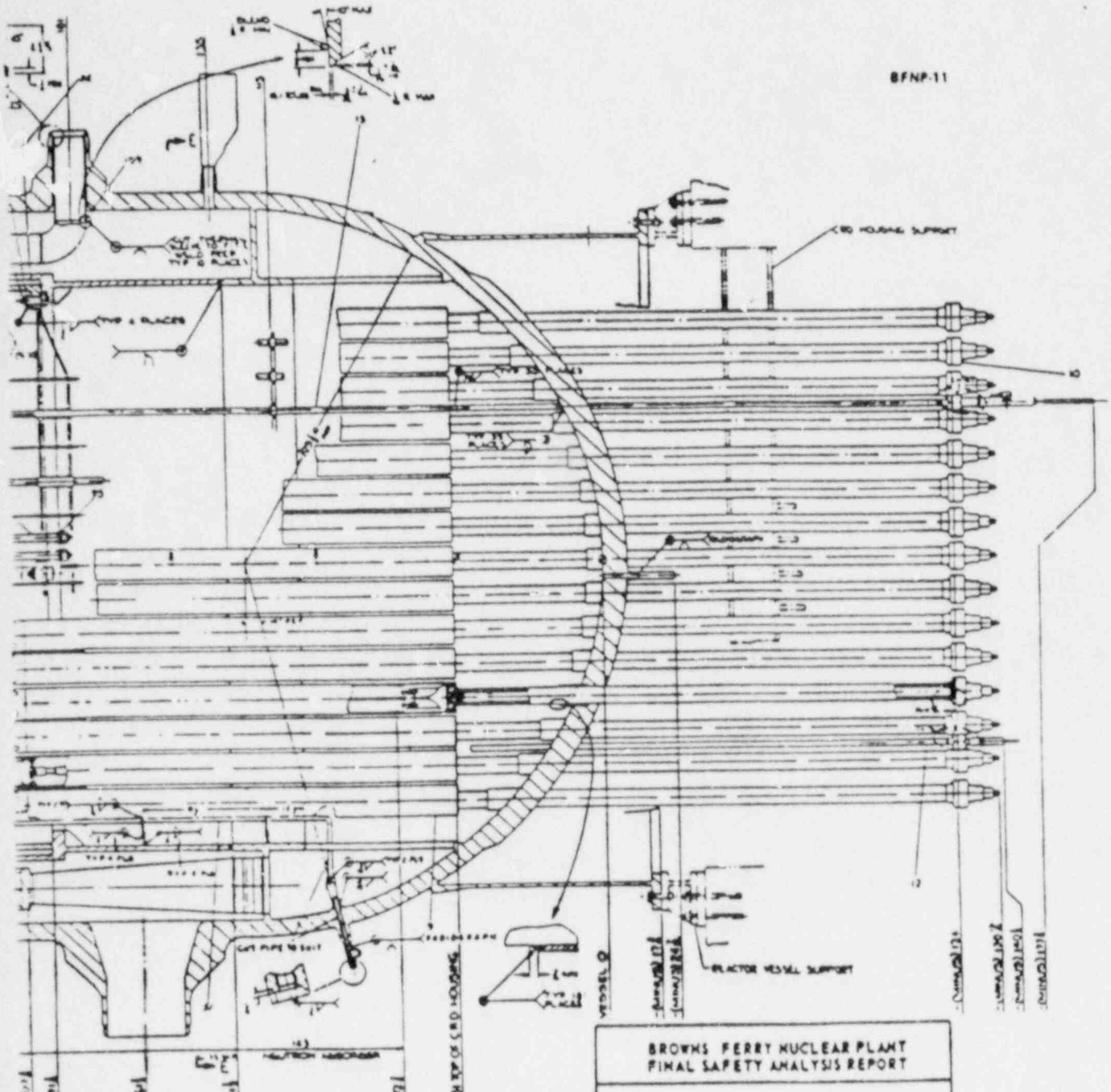
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

[illegible]

Millstone
50-245

FIGURE IV-2.1. REACTOR VESSEL AND NOZZLES

FIGURE A2



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Reactor Vessel
FIGURE 4.2-1
(Revised by Amendment 11)

Docket 259, 260

FIGURE A3

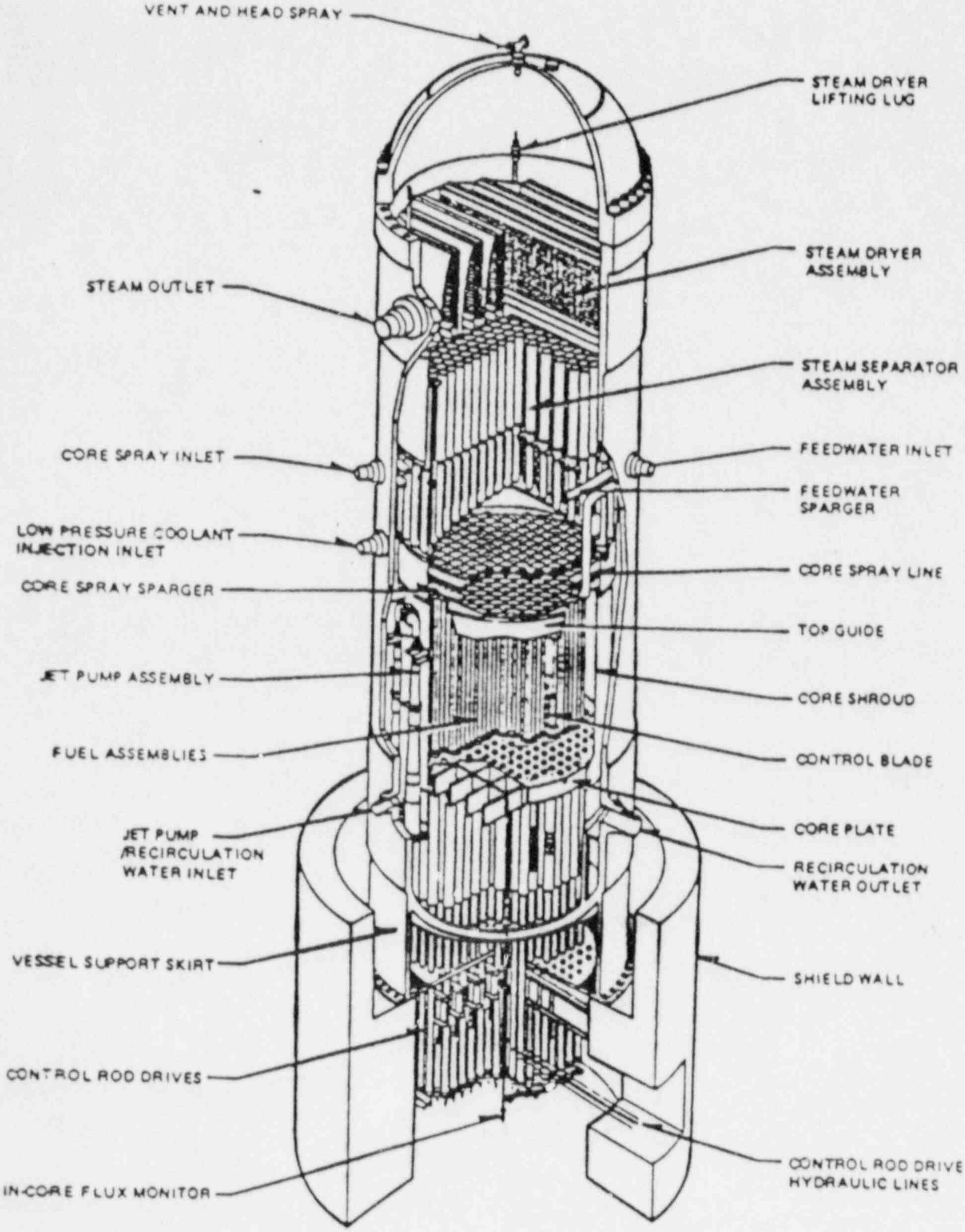
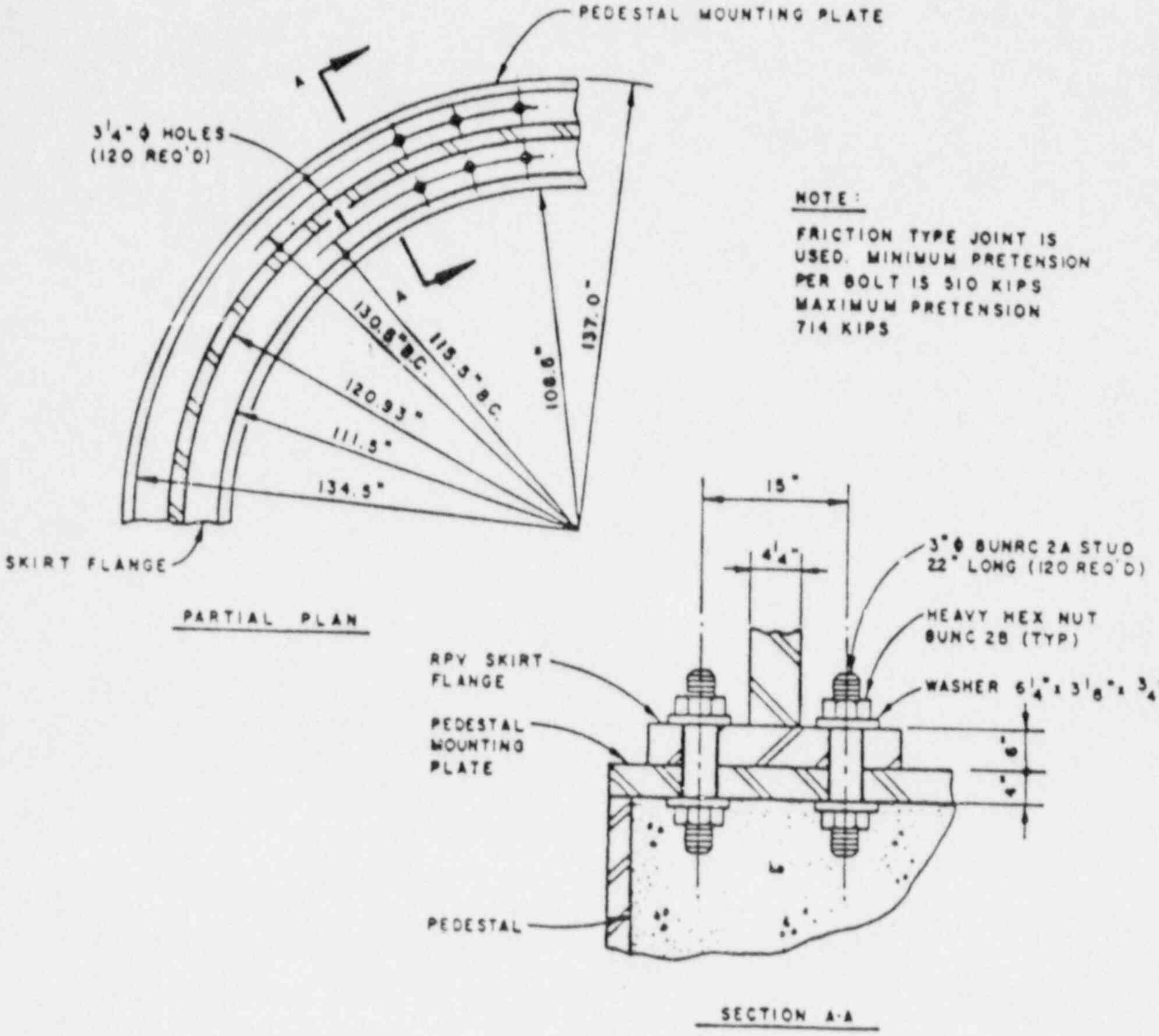


Figure 3.9-8. Reactor Vessel Cutaway

FIGURE A4

DESCRIPTION OF STRESSES RESULTING FROM CONNECTION OF RPV TO PEDESTAL



SUMMARY OF STRESSES IN VESSEL SUPPORT

LOAD CONDITION	P_m STRESS	P_m ALLOWABLE	$P_m + P_b$ STRESS	$P_m + P_b$ ALLOWABLE	$P_m + Q_m$ STRESS	$P_m + Q_m$ ALLOWABLE	$P_m + Q_m + P_m + Q_b$ STRESS	$P_m + Q_m + P_m + Q_b$ ALLOWABLE
LEVEL A & B	31.7	54	ENR	—	78.0	90	93.7	144
LEVEL C	STRESSES ARE SAME AS A & B CONDITIONS, ALLOWABLES ARE HIGHER							
LEVEL D	77.4	94.5	ENR	—	ENR	—	ENR	—

P_m = PRIMARY MEMBRANE STRESS, KSI
P_b = PRIMARY BENDING STRESS, KSI
Q_m = SECONDARY MEMBRANE STRESS, KSI
Q_b = SECONDARY BENDING STRESS, KSI
ENR = EVALUATION NOT REQUIRED

Figure 5.3-6. Details of Vessel Support

FIGURE A5

Rancho Seco
50-312

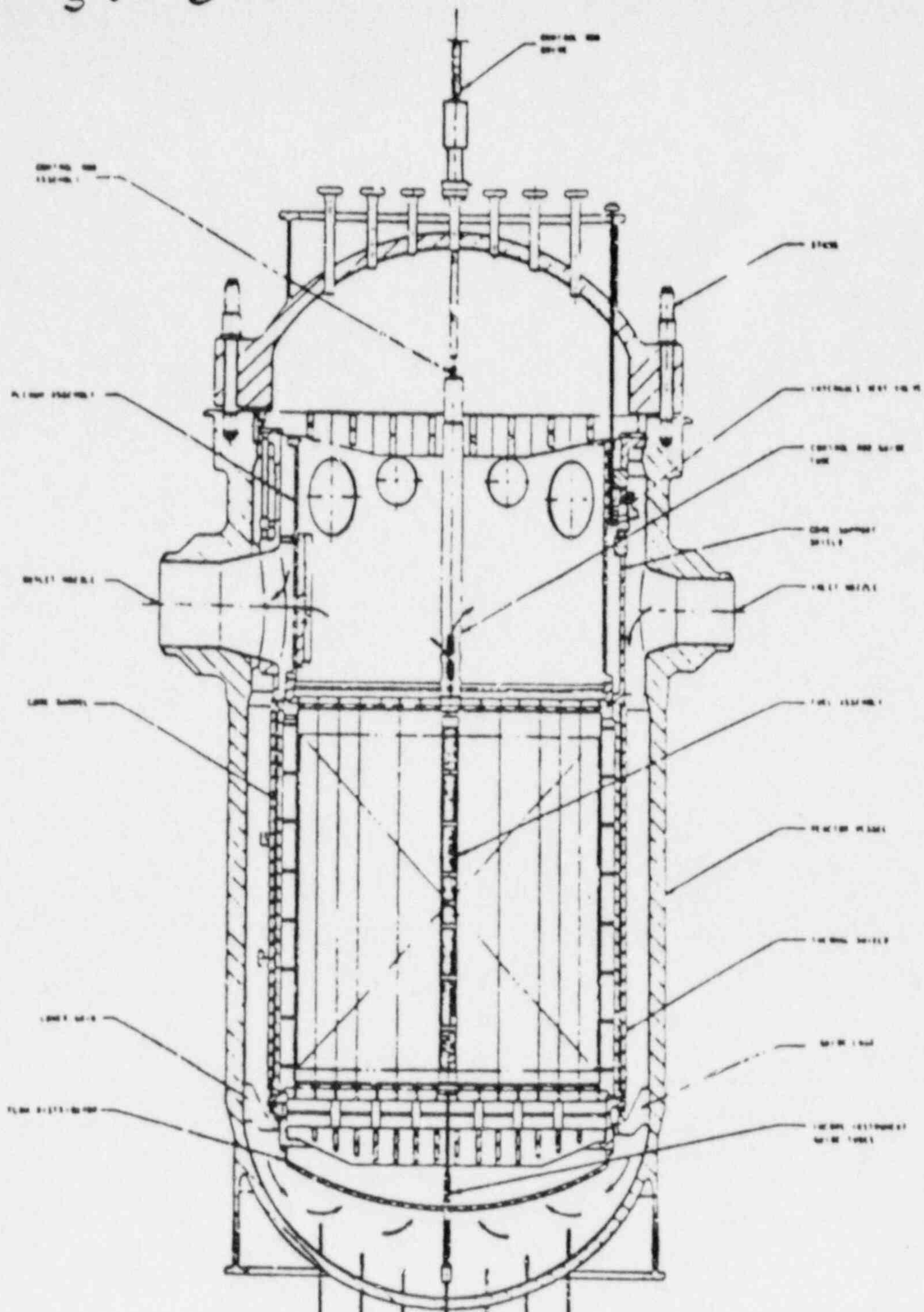
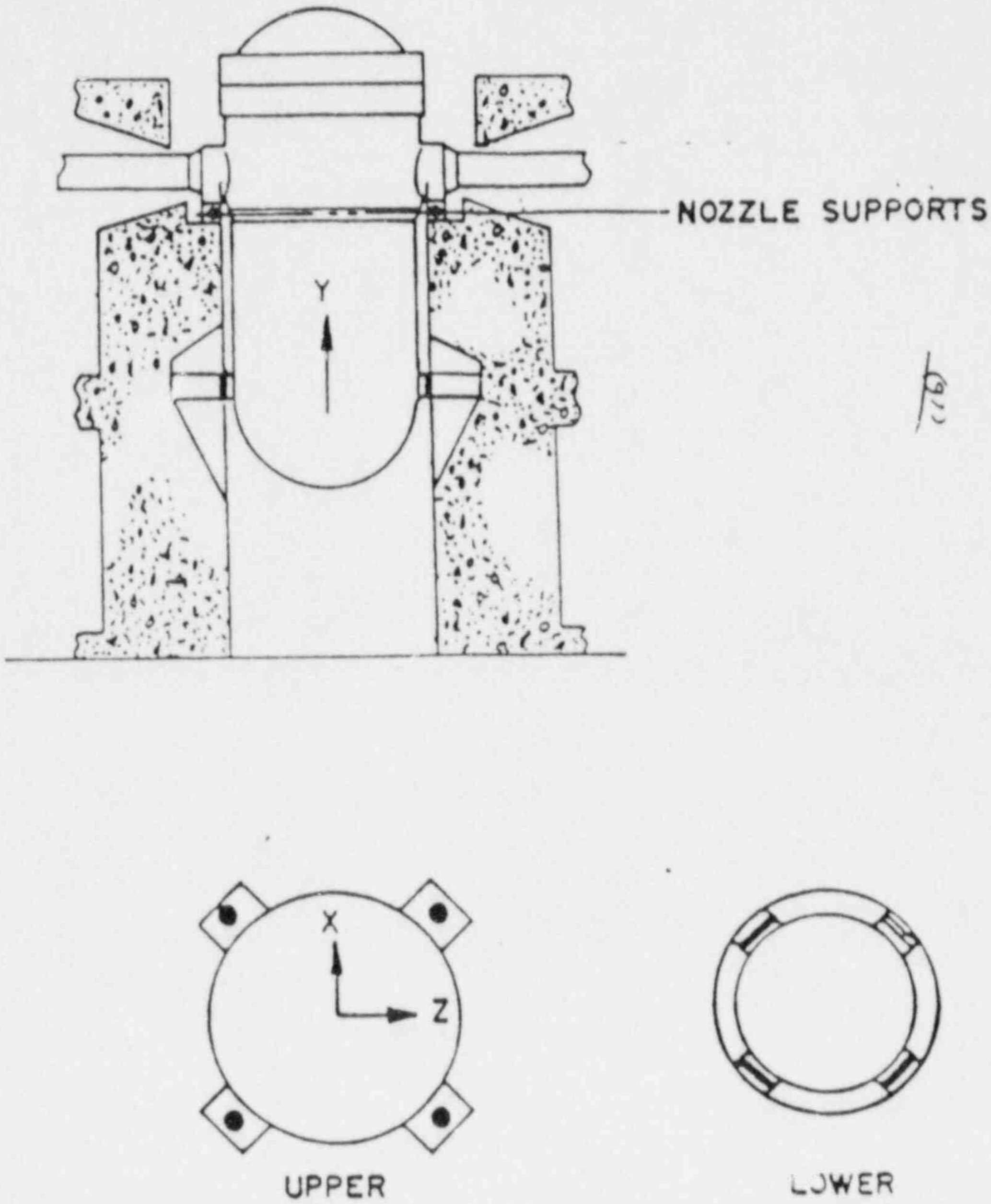


FIGURE 3.2-8
REACTOR VESSEL AND INTERNALS-
GENERAL ARRANGEMENT

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

B & W Bechtel

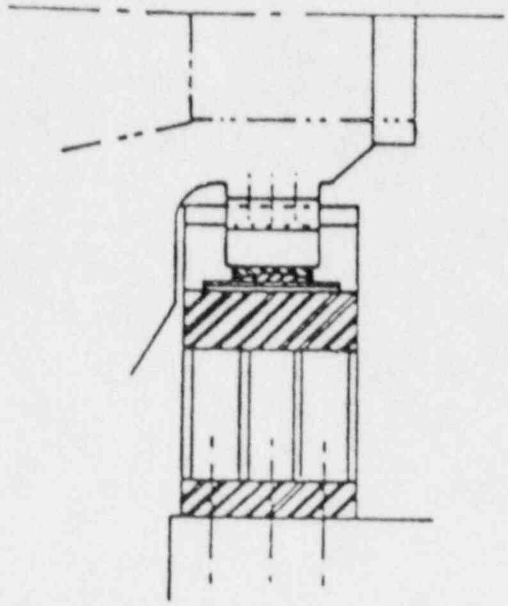
FIGURE A6



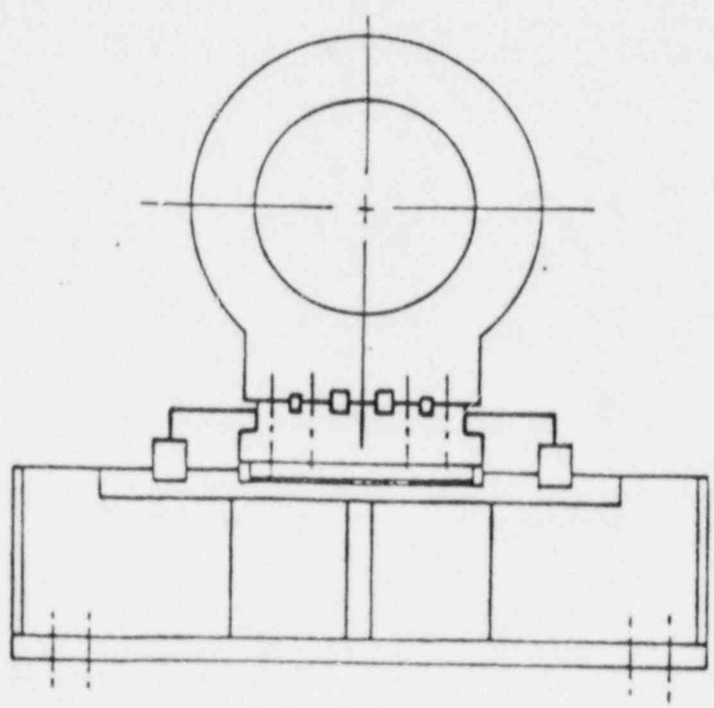
BABCOCK 241
REACTOR VESSEL SUPPORTS

Figure 5.5-9

FIGURE A7



SECTION A-A

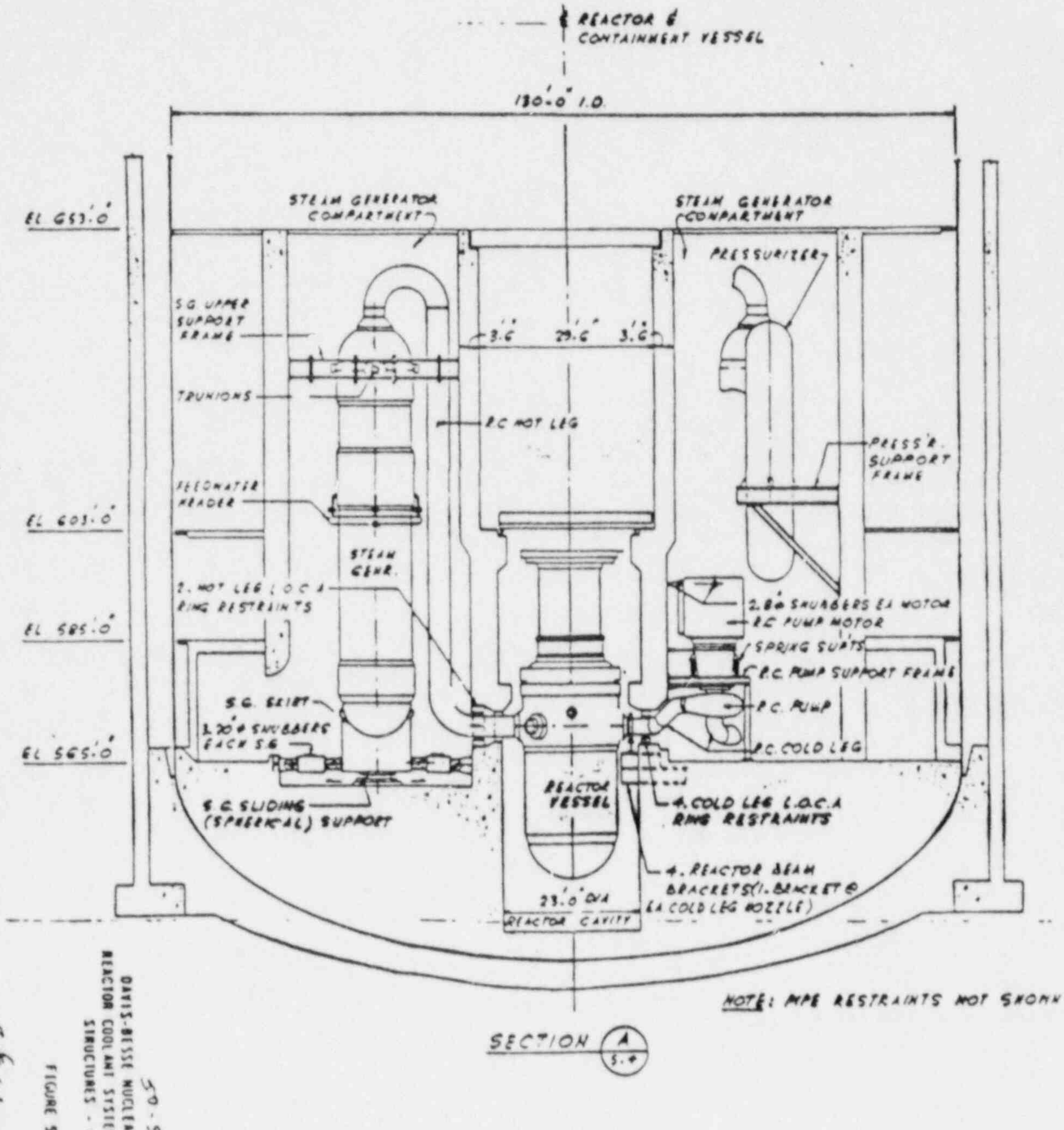


ELEVATION

FIGURE 1-8

BABCOCK 241
REACTOR VESSEL SUPPORT PAD

Figure 5.5-10



NOTE: MPE RESTRAINTS NOT SHOWN

SECTION A/5.4

FIGURE A7

DAVIS-BESSE NUCLEAR POWER STATION
REACTOR COOLANT SYSTEM AND SUPPORTING STRUCTURES - ELEVATION
FIGURE 5.1-4 REVISION 0 JULY 1982

50-346 ... 5.0 ...

B&W Bechtel

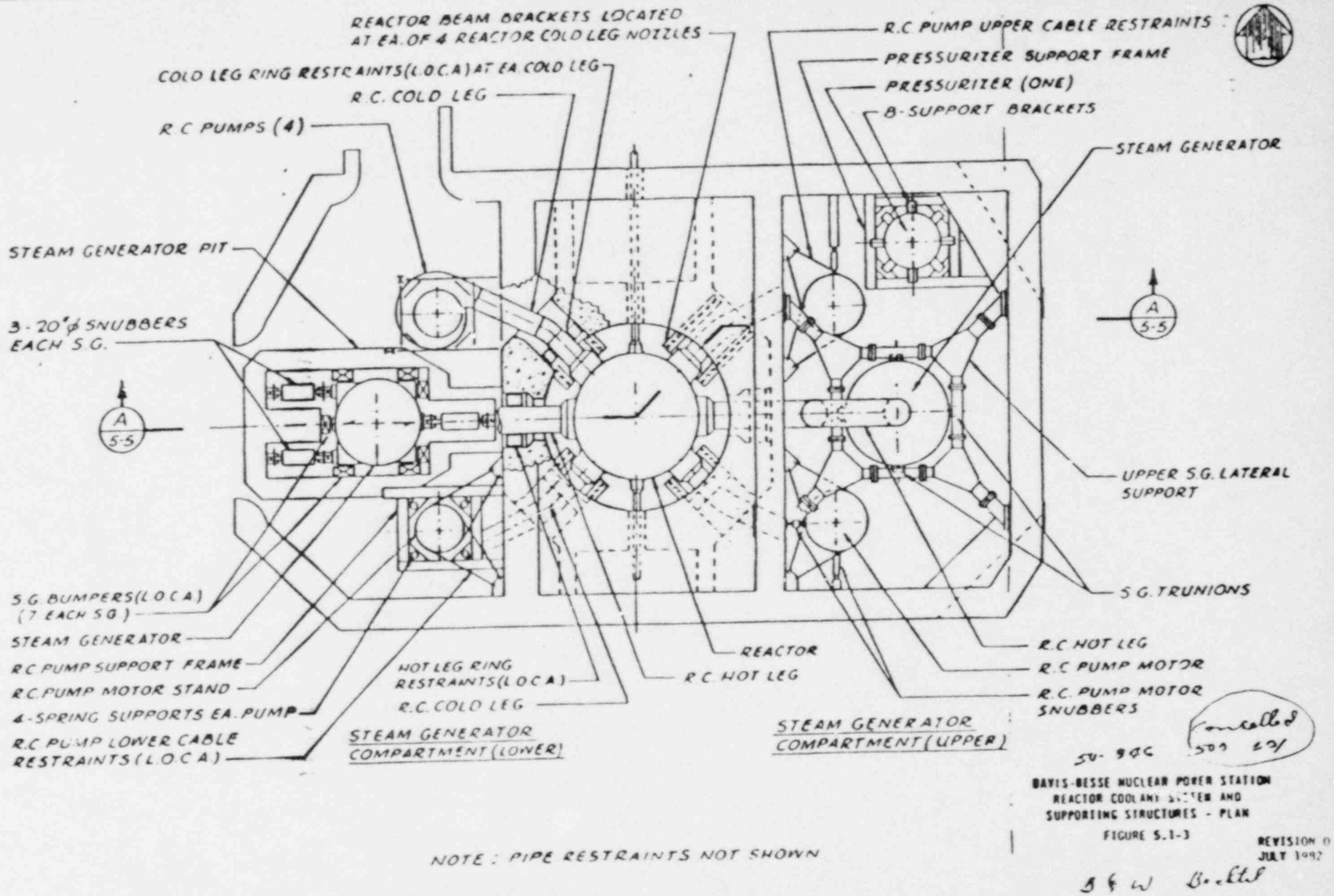
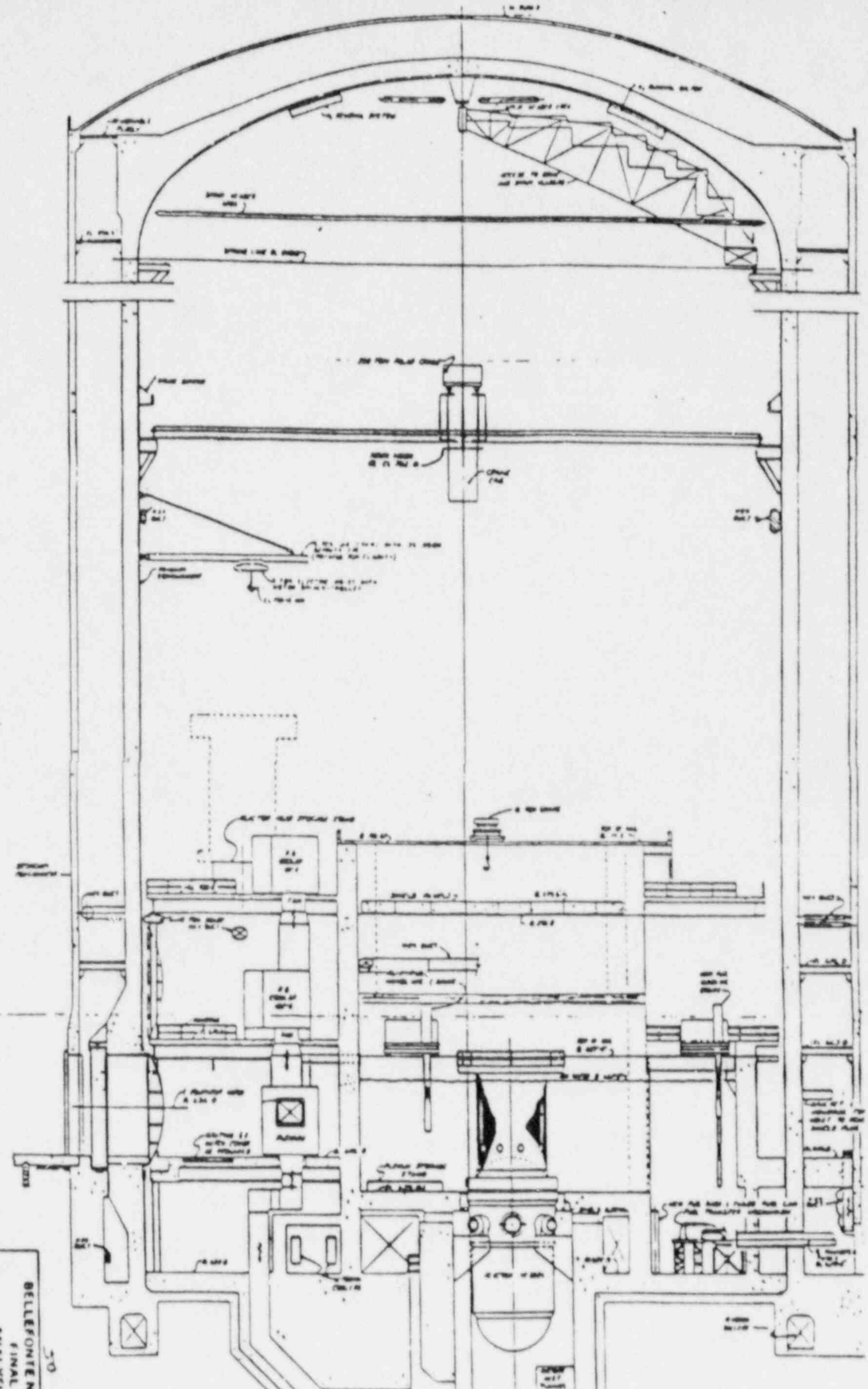

REACTOR BEAM BRACKETS LOCATED AT EA. OF 4 REACTOR COLD LEG NOZZLES
COLD LEG RING RESTRAINTS (L.O.C.A) AT EA. COLD LEG
R.C. COLD LEG
R.C PUMPS (4)
STEAM GENERATOR PIT
3-20"ϕ SNUBBERS EACH S.G.
A 5-5
S.G. BUMPERS (L.O.C.A) (7 EACH S.G.)
STEAM GENERATOR
R.C PUMP SUPPORT FRAME
R.C. PUMP MOTOR STAND
4-SPRING SUPPORTS EA. PUMP
R.C PUMP LOWER CABLE RESTRAINTS (L.O.C.A)
HOT LEG RING RESTRAINTS (L.O.C.A)
R.C. COLD LEG
STEAM GENERATOR COMPARTMENT (LOWER)
REACTOR
R.C. HOT LEG
R.C. PUMP UPPER CABLE RESTRAINTS
PRESSURIZER SUPPORT FRAME
PRESSURIZER (ONE)
8-SUPPORT BRACKETS
STEAM GENERATOR
A 5-5
UPPER S.G. LATERAL SUPPORT
S.G. TRUNIONS
R.C HOT LEG
R.C PUMP MOTOR
R.C. PUMP MOTOR SNUBBERS
STEAM GENERATOR COMPARTMENT (UPPER)
NOTE: PIPE RESTRAINTS NOT SHOWN
DAVIS-BESSE NUCLEAR POWER STATION
REACTOR COOLANT SYSTEM AND
SUPPORTING STRUCTURES - PLAN
FIGURE 5.1-3
REVISION 0
JULY 1982
B&W

FIGURE A10

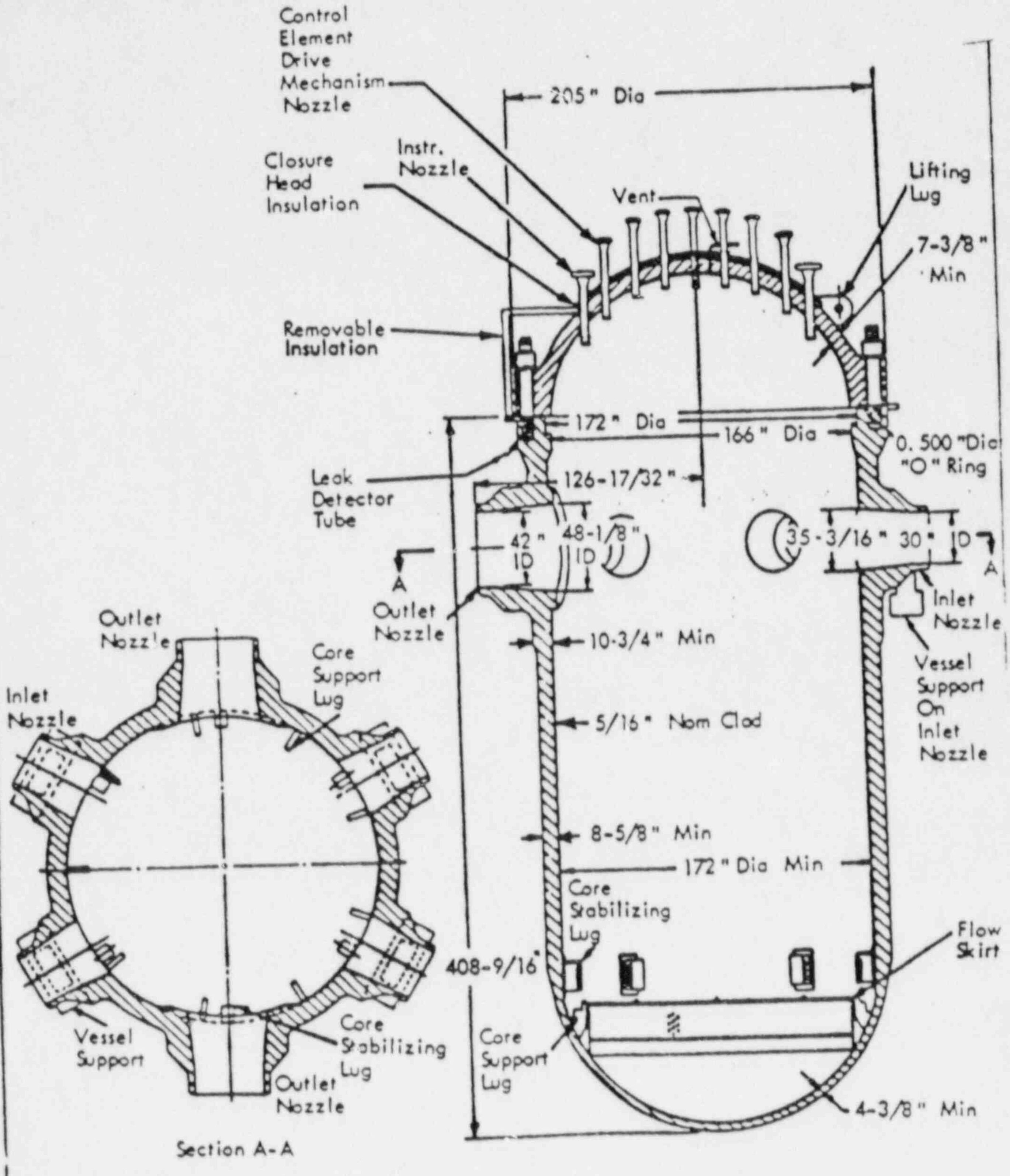


50-438/437

BELLEFONTE NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

EQUIPMENT–REACTOR BUILDING
SECTION B15-B15
TVA DWG. NO 38W0206-06 15R4
FIGURE 1.2-16

FIGURE A11

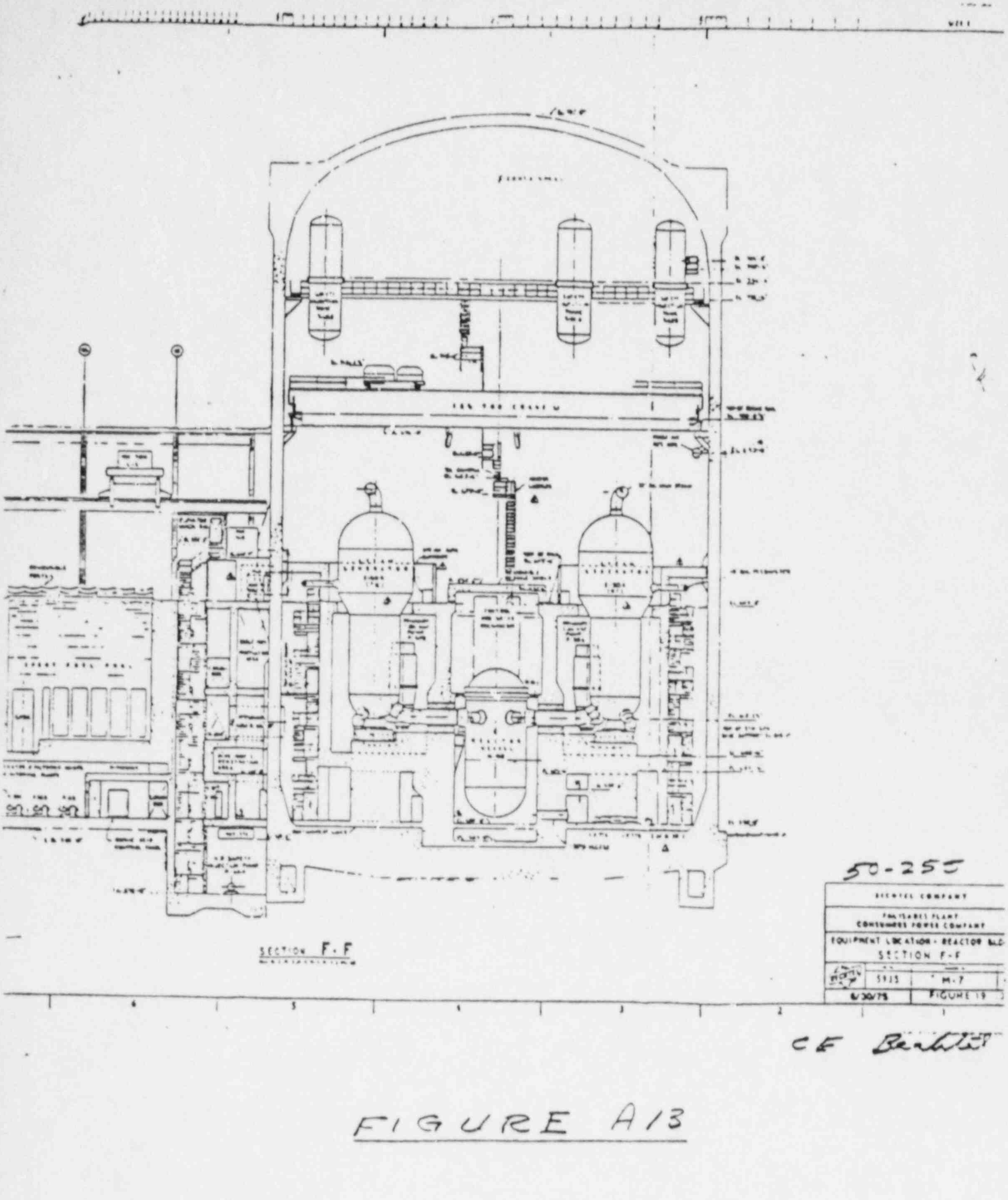


50-317, 318

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	REACTOR VESSEL	Figure 4-2

CE Bechtel

FIGURE A12



CE Bechtel

FIGURE A13

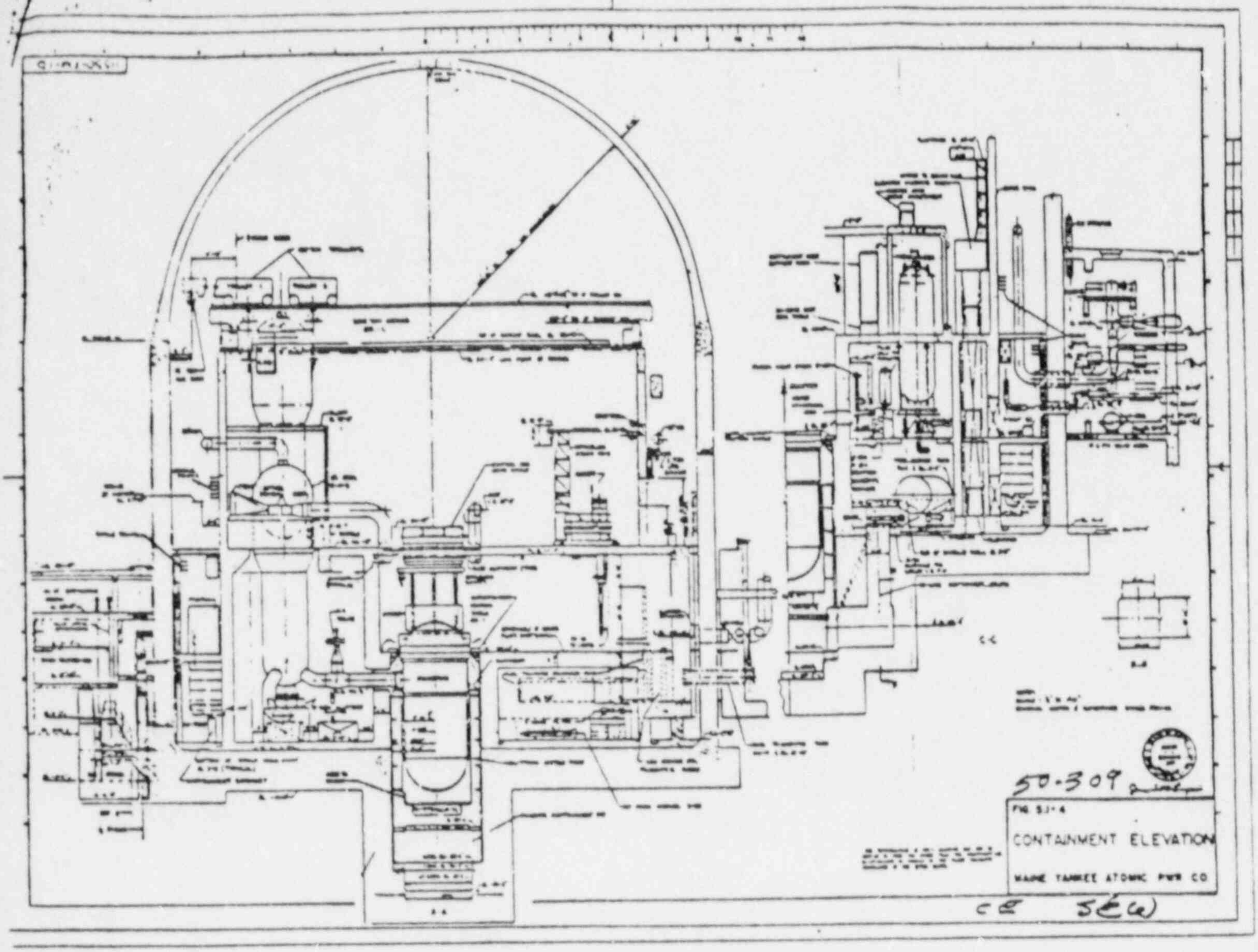
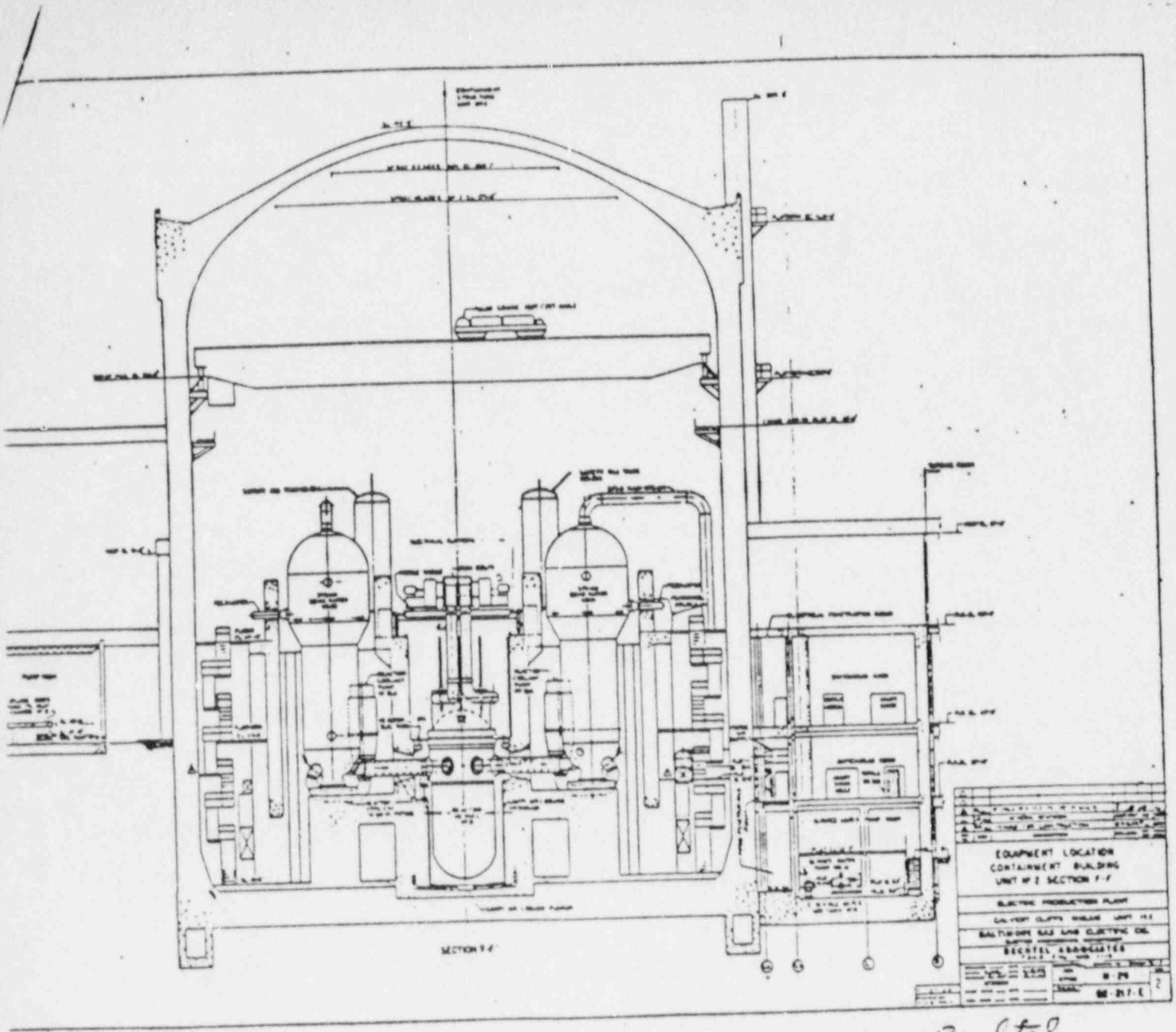

50-309
FIG. 5.1-4
CONTAINMENT ELEVATION
MAINE YANKEE ATOMIC PWR CO.
CE 5E(a)

FIGURE A14

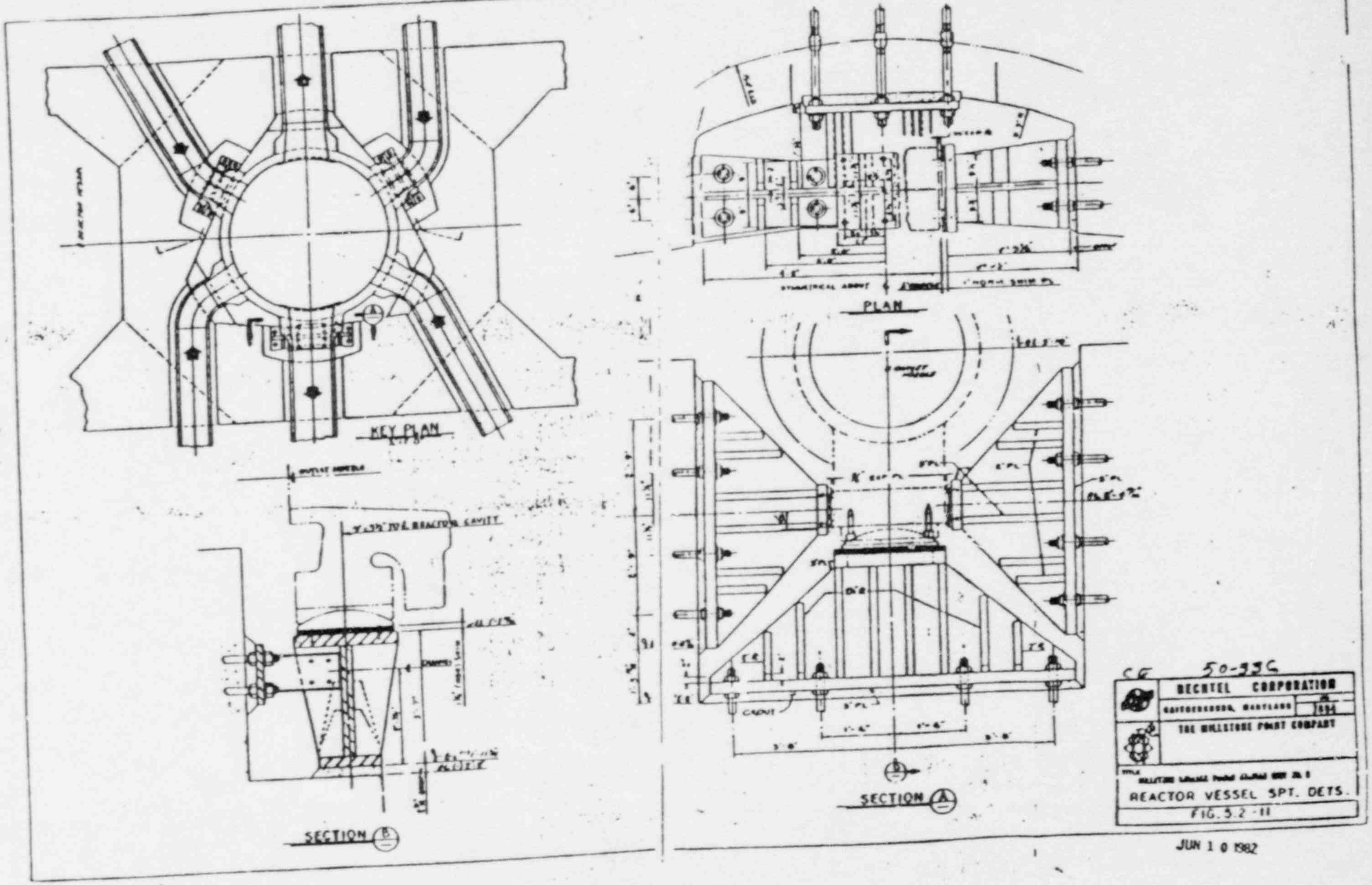


Bechtel

CE
50-317
318

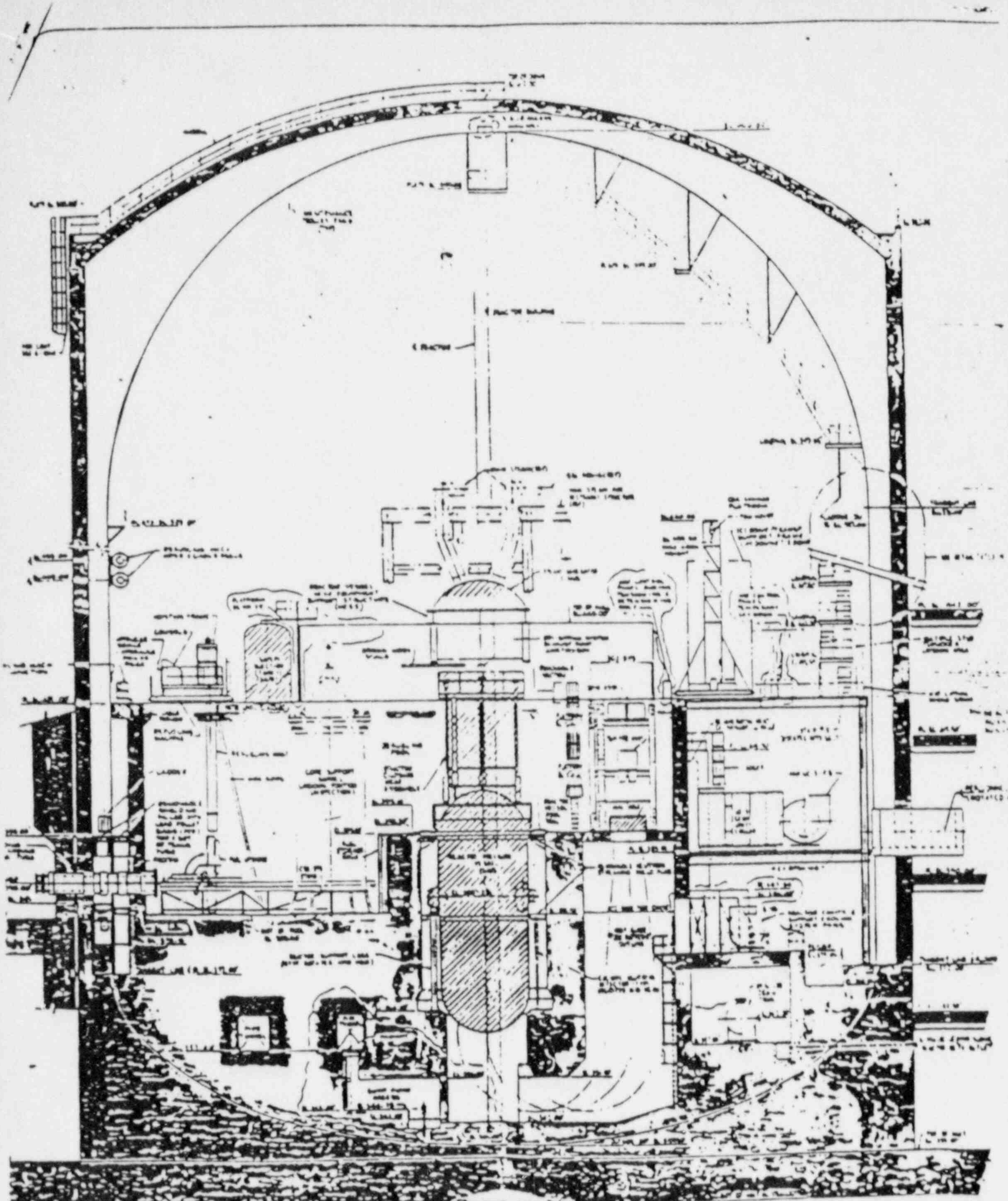
FIGURE A15

Millstone 2 FSAR 50-336



JUN 1 0 1982

FIGURE A16



WPPS 3 & 5

cancel

FIGURE A17 50-508

REF DWG WPPS-3240

CE

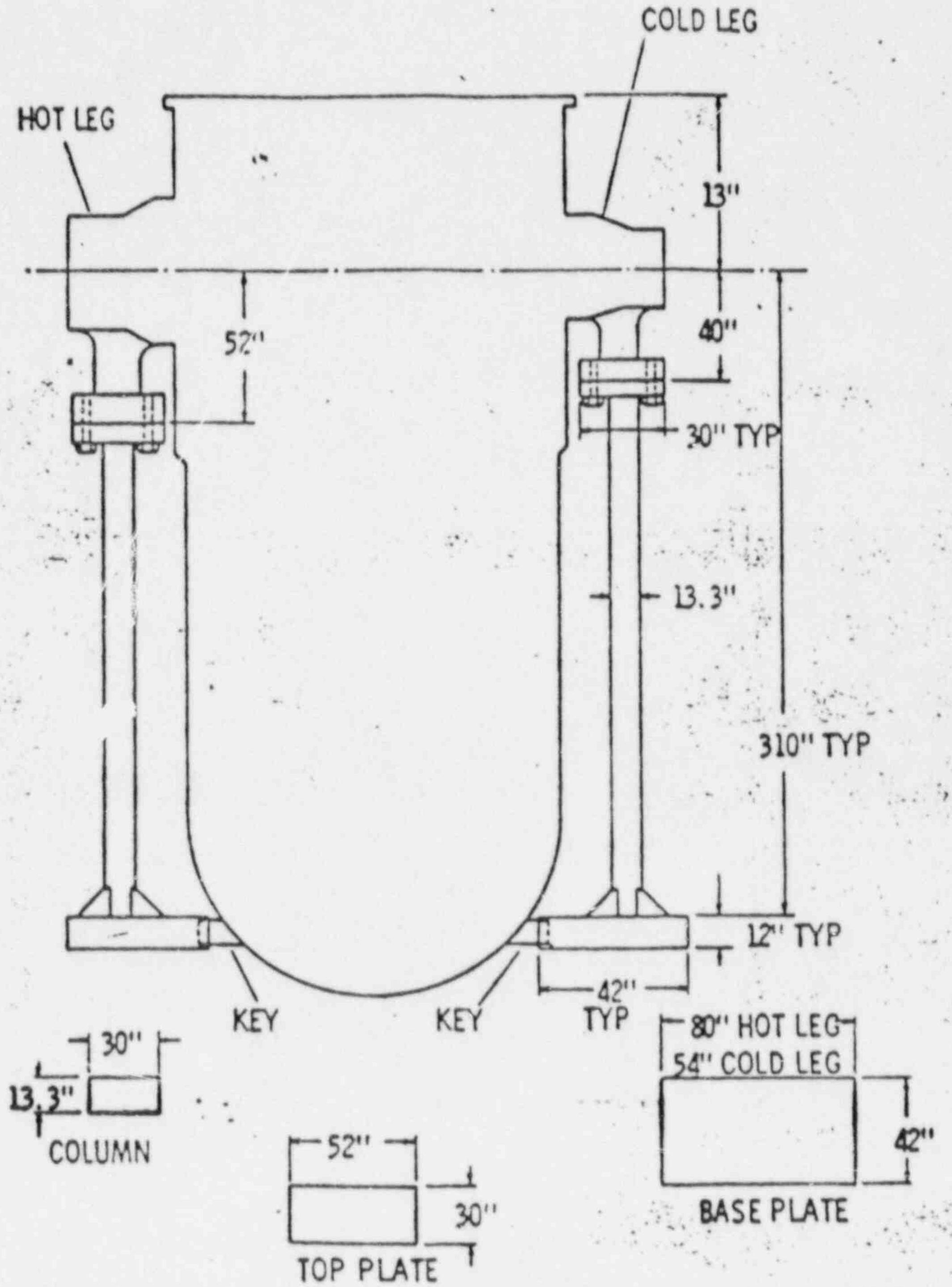
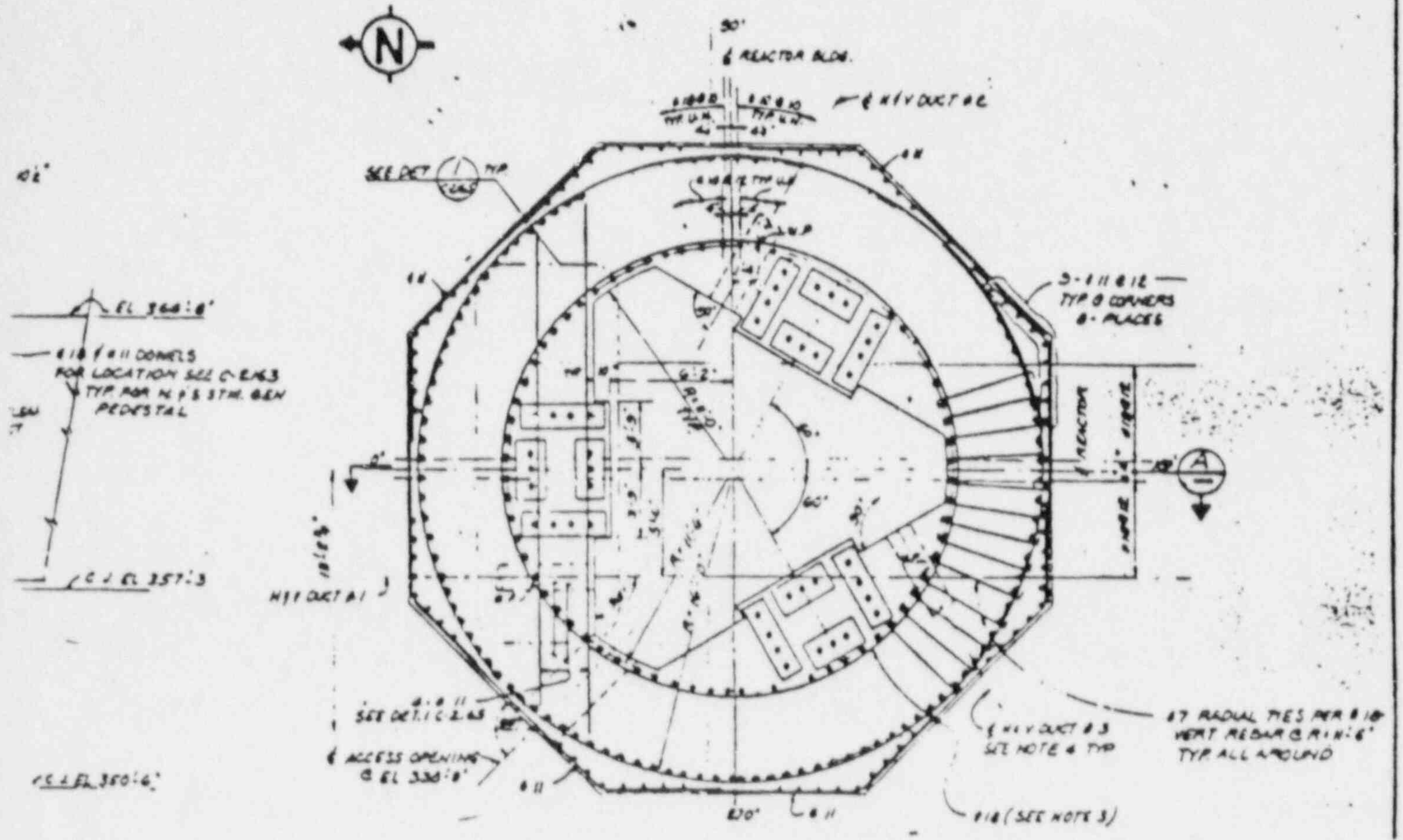


FIGURE A18

ARKANSAS POWER & LIGHT CO. Arkansas Nuclear One - Unit 2	REACTOR VESSEL SUPPORTS 50-313 50-368 CE Bechtel	Figure 5.5-12



PLAN @ EL 336'-6"

PRIMARY SHIELD WALLS

CE Bechtel 50-368

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE UNIT-2	INTERNAL STRUCTURES PLAN & SECTION	FIG. NO. 3.8-15

FIGURE A19

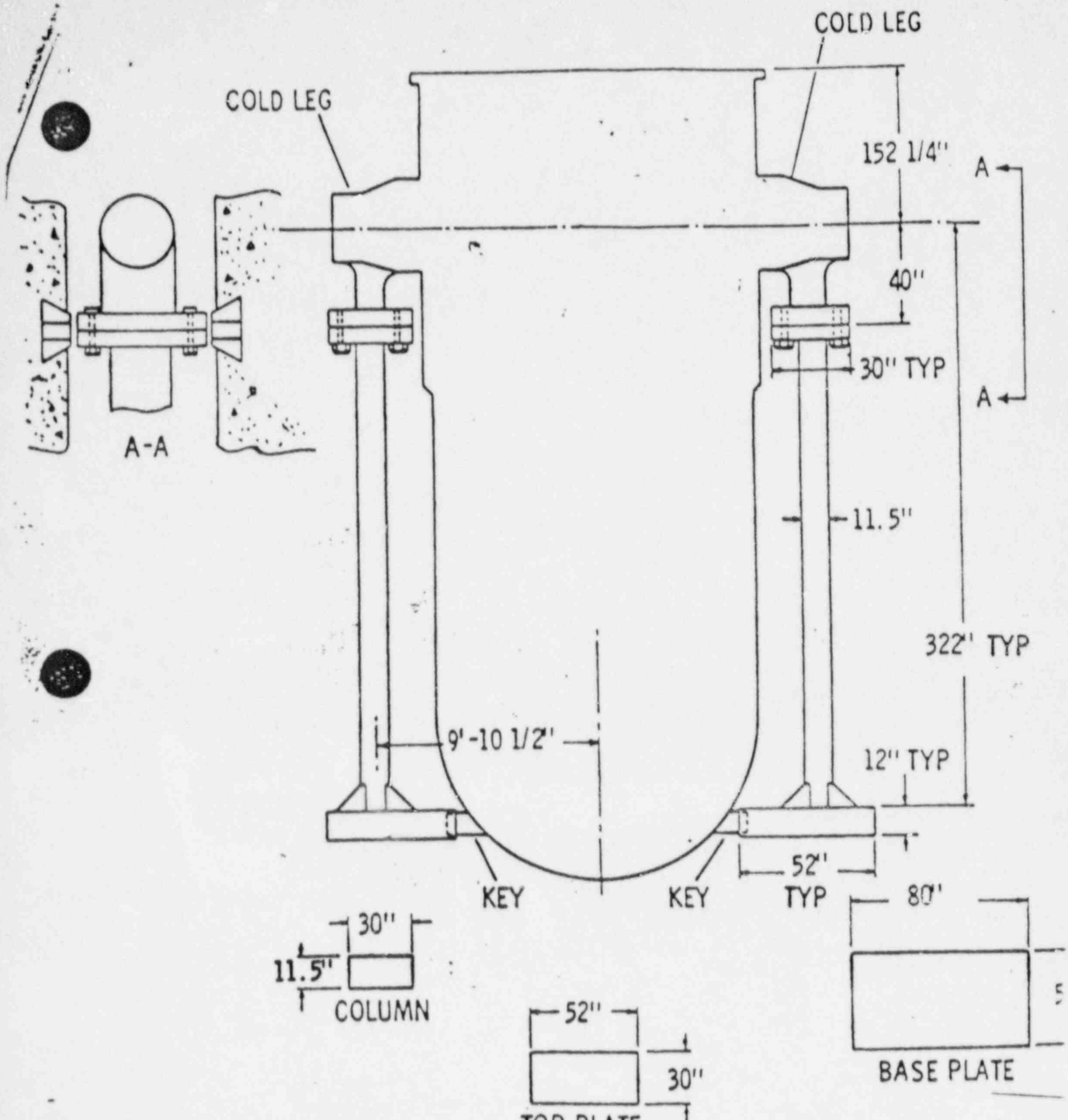


FIGURE A20

C-E SYSTEM 80	REACTOR VESSEL SUPPORTS	Figure 5.4.14-2

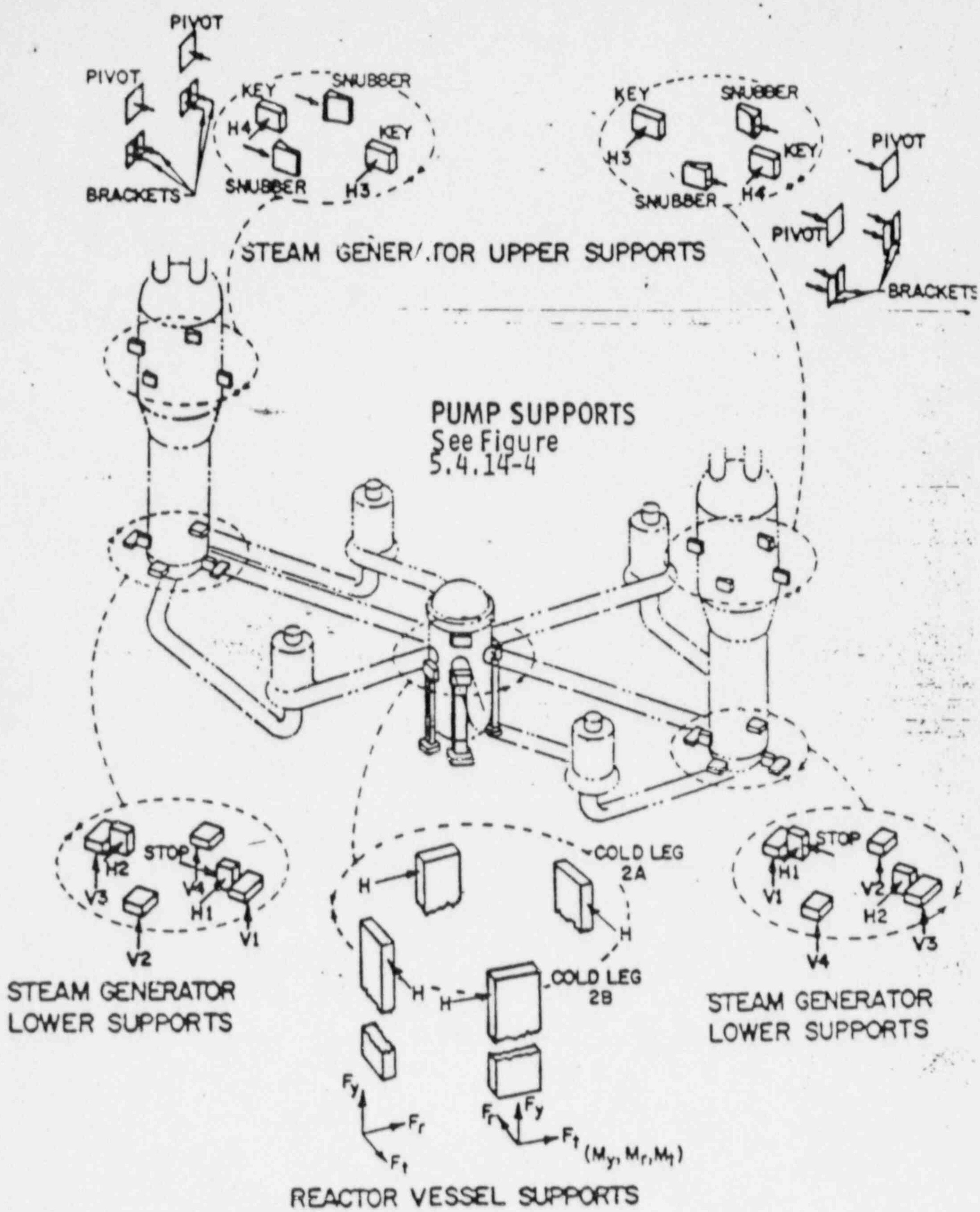


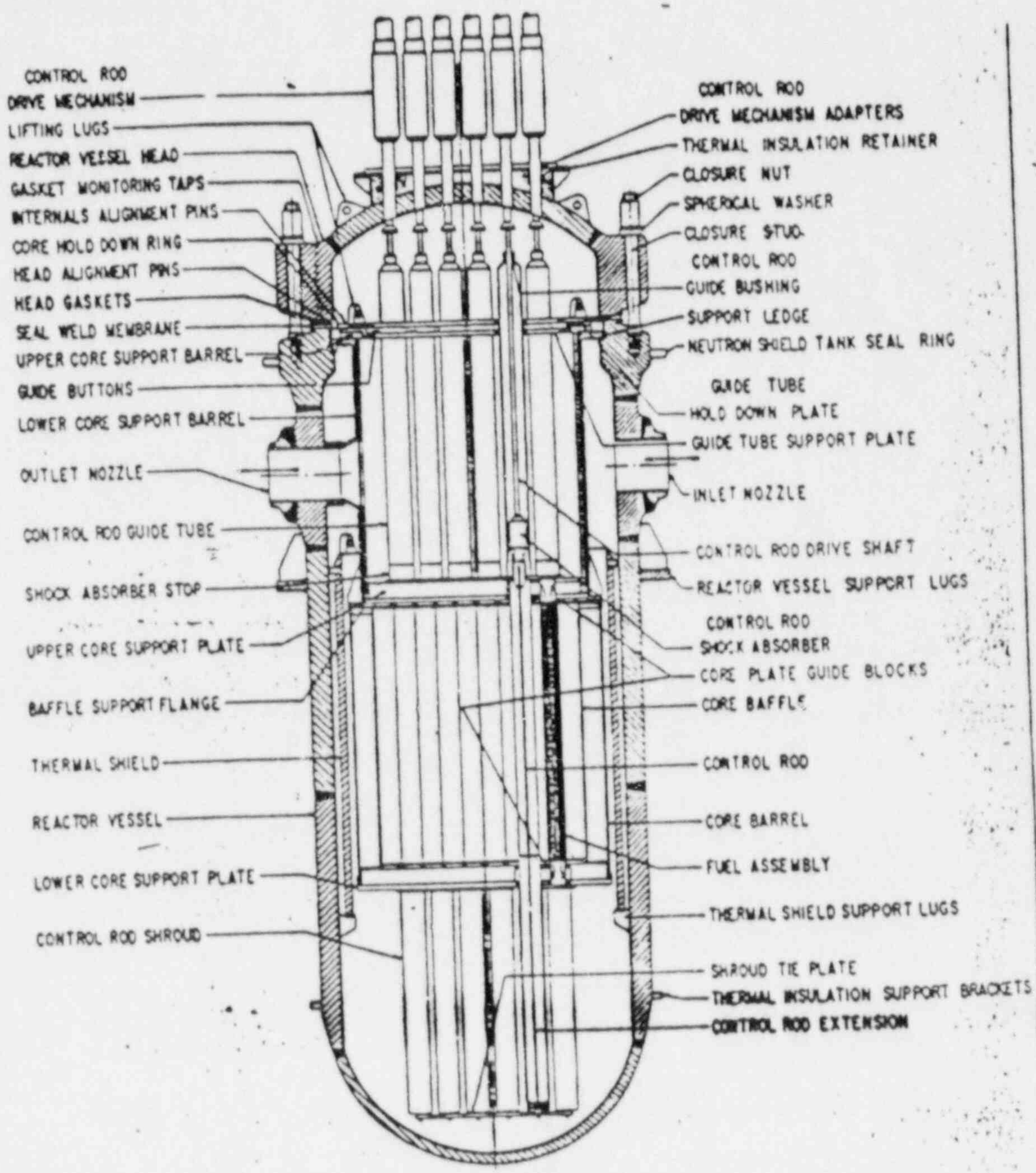
FIGURE A21

C-E SYSTEM 80	REACTOR COOLANT SYSTEM ARRANGEMENT AND SUPPORT POINTS	Figure 5.4.14-1

Yankee Rowe Docket 50-29

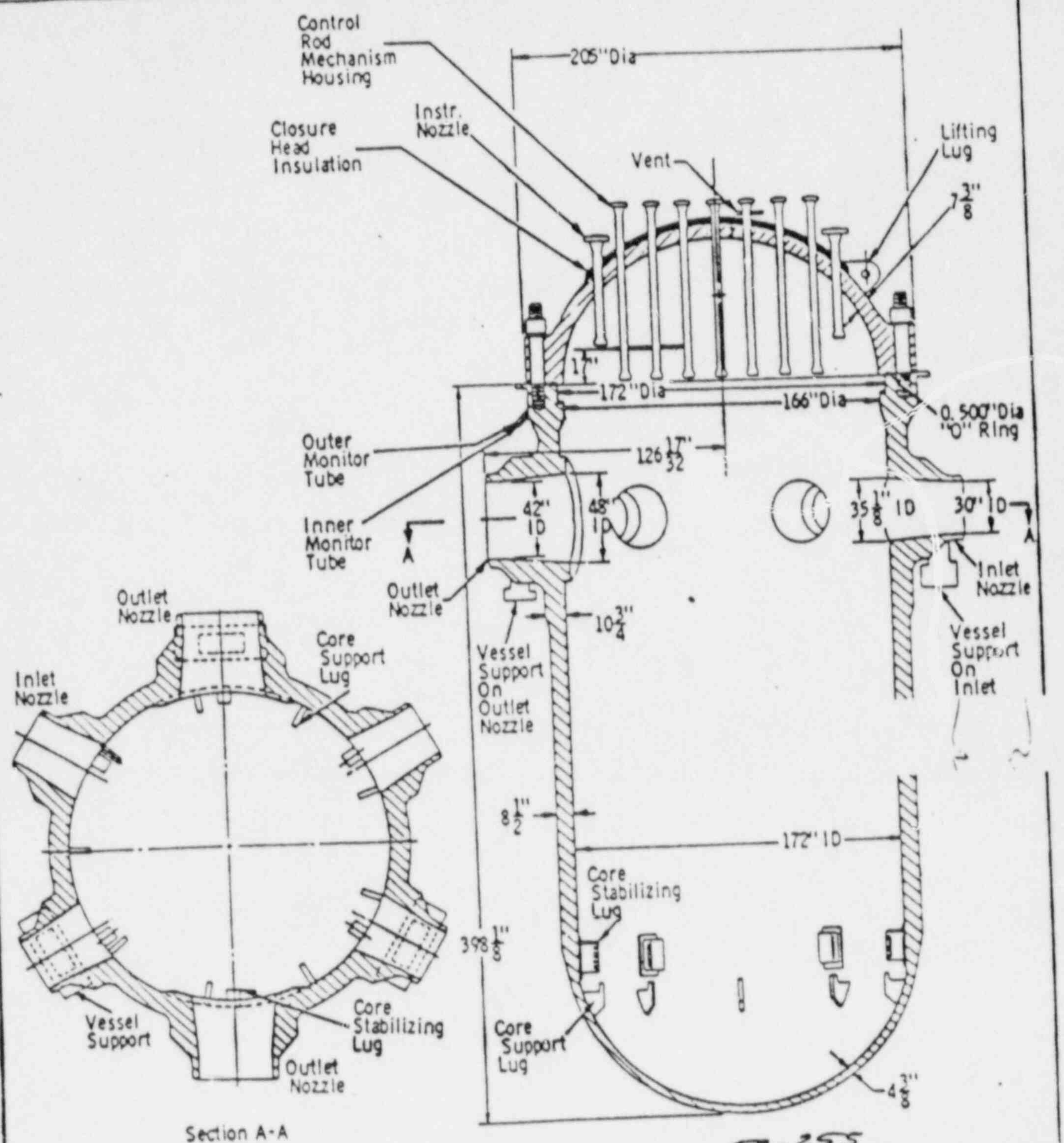
FSAR

.WE EW



REACTOR VESSEL ASSEMBLY

FIGURE A22



50-255

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

REACTOR VESSEL

FIGURE NO 4-2

REVISION NO 0

FIGURE A23

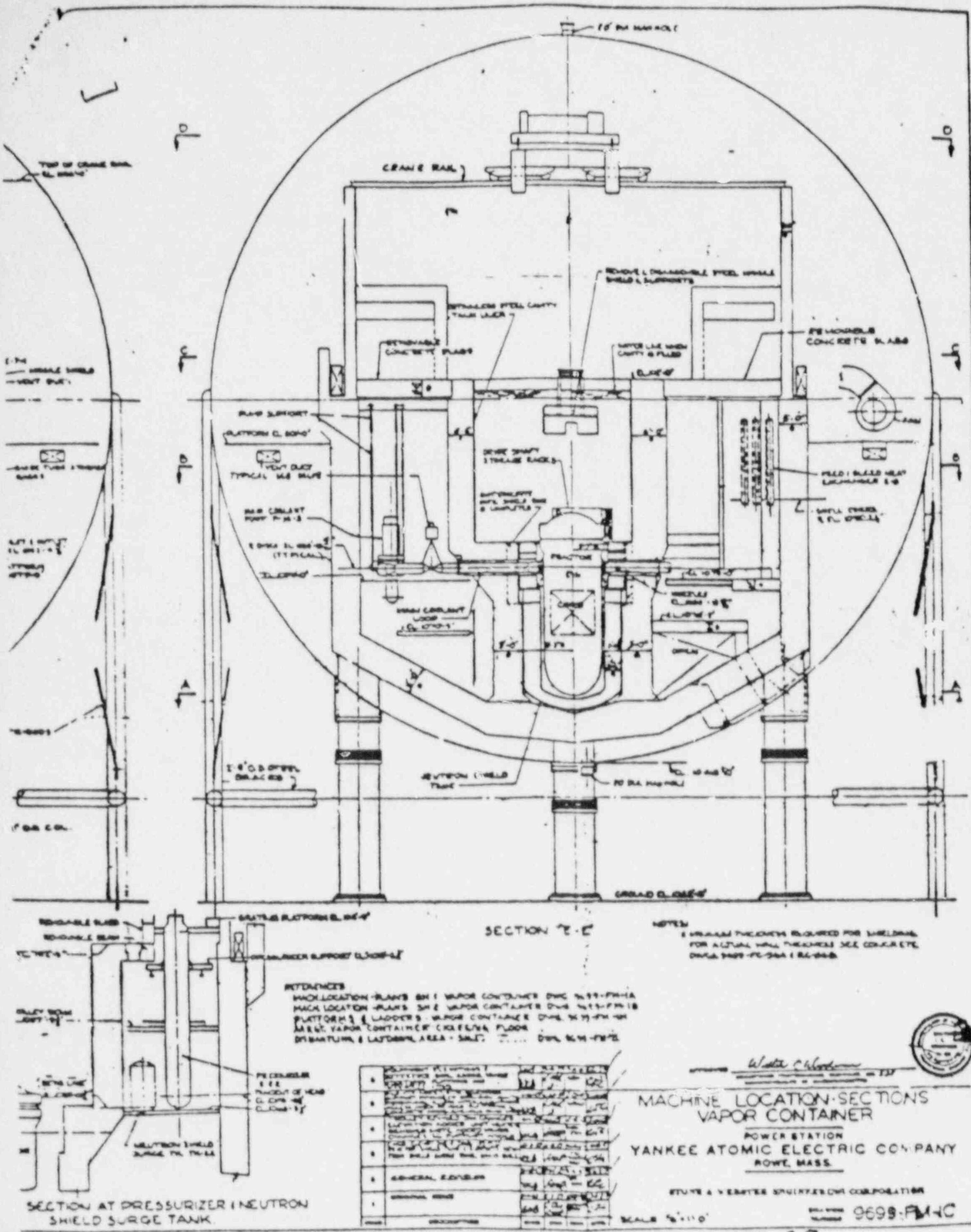
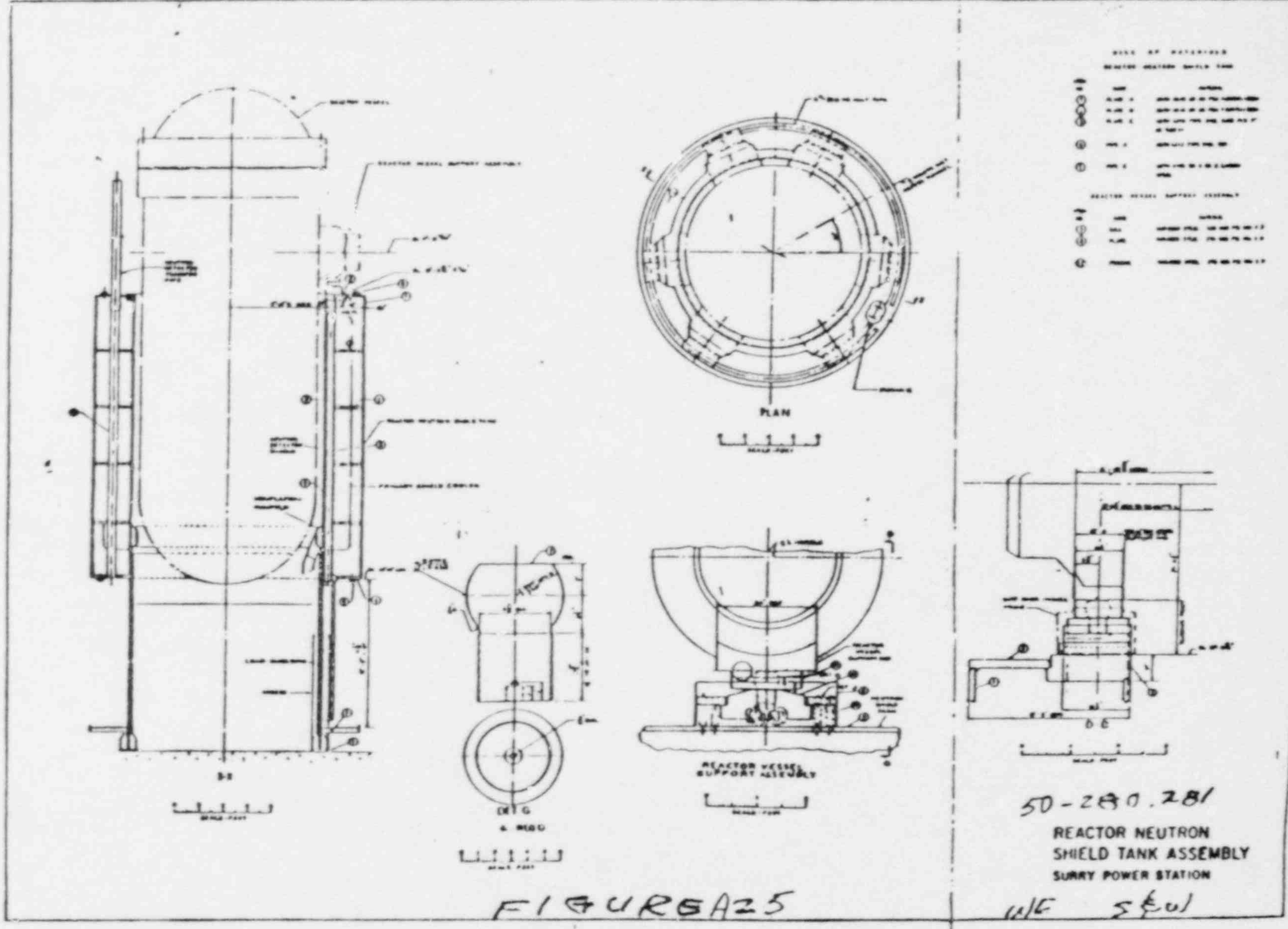


FIGURE A24

WE 57-29 SW



50-280, 281

REACTOR NEUTRON
SHIELD TANK ASSEMBLY
SURRY POWER STATION

FIGURE A25

WF S&W

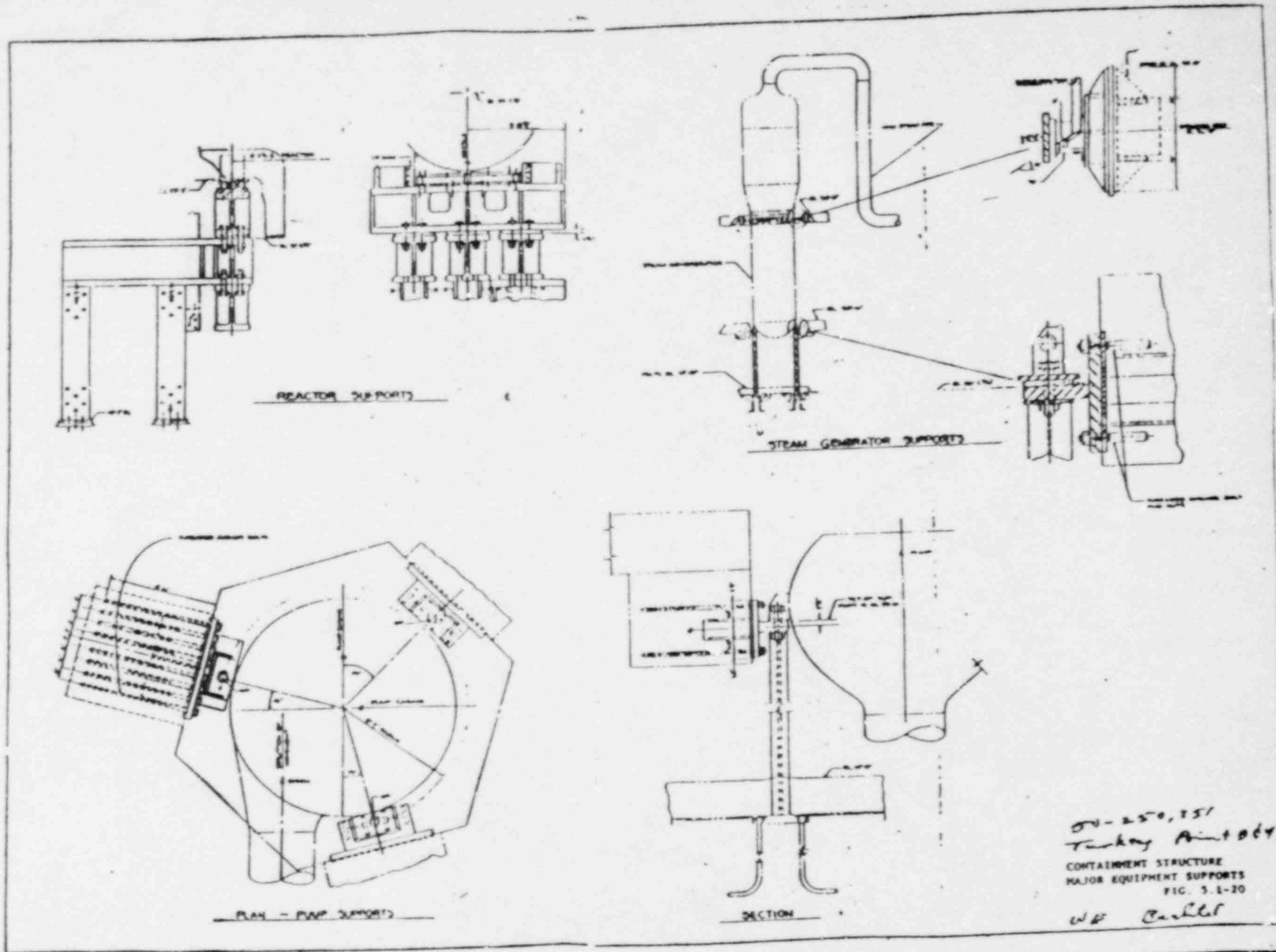
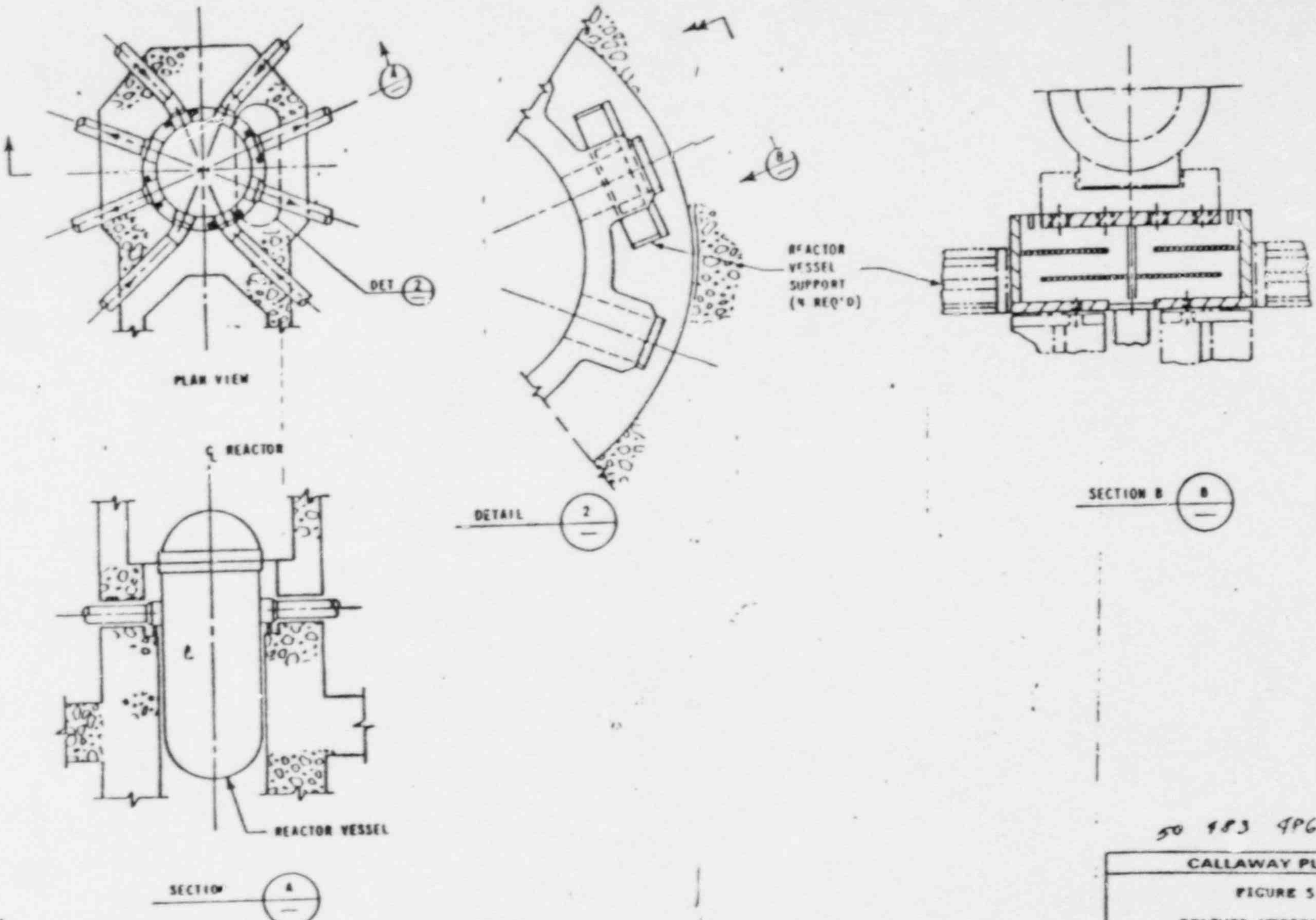

REACTOR SUPPORTS
STEAM GENERATOR SUPPORTS
PLAN – PUMP SUPPORTS
SECTION
50-250, 251
Turkey Point 3&4
CONTAINMENT STRUCTURE
MAJOR EQUIPMENT SUPPORTS
FIG. 5.1-20

FIGURE A26



50 483 4P6

NSSS AE Daniel Corr Rev OLO 6/86

CALLAWAY PLANT
FIGURE 5.4-13
REACYOR VESSEL SUPPORTS

FIGUREA27

8555-29

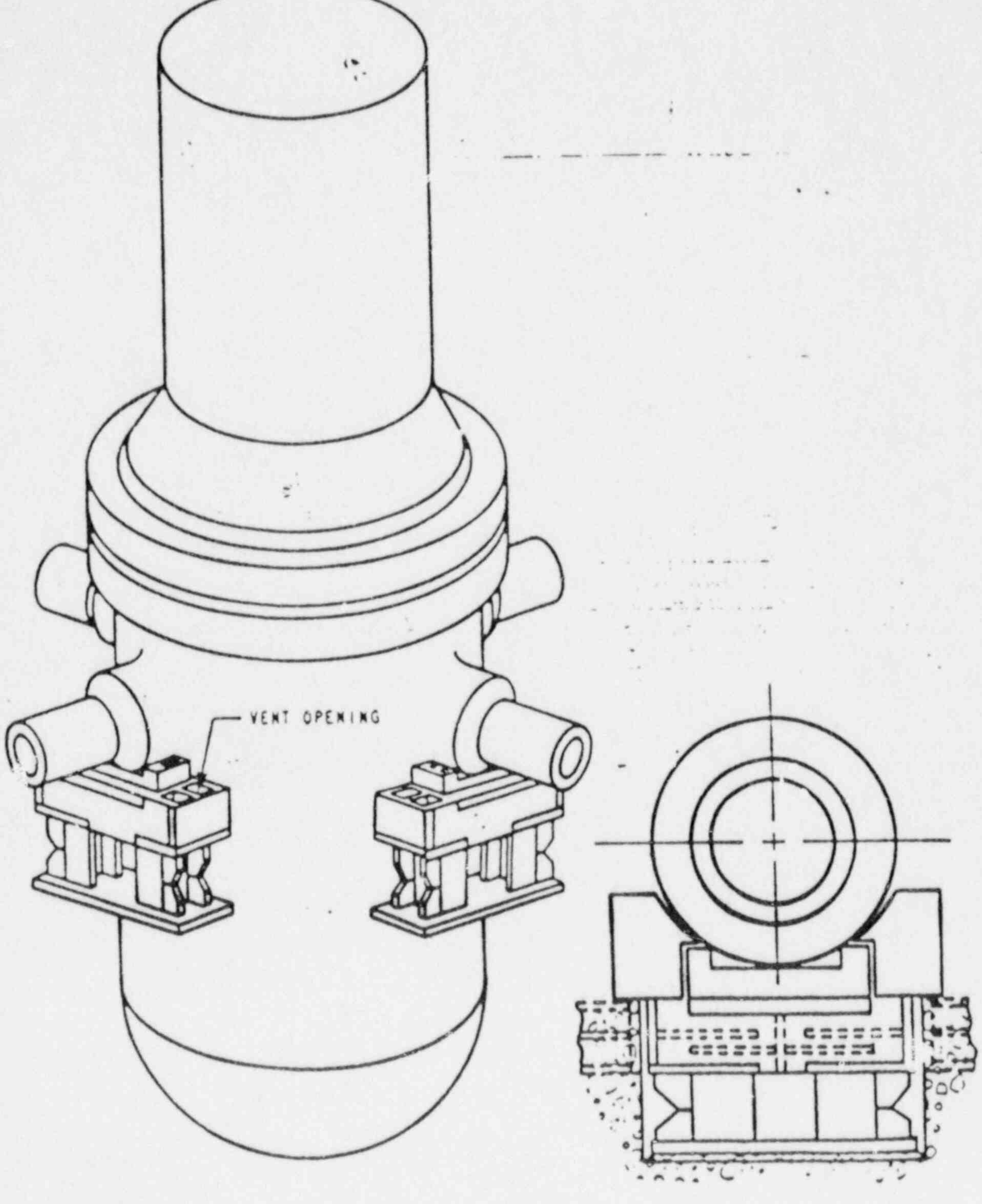


FIGURE A28

Figure 5.5-6 Reactor Vessel Supports

Watts Bar 50-390/391

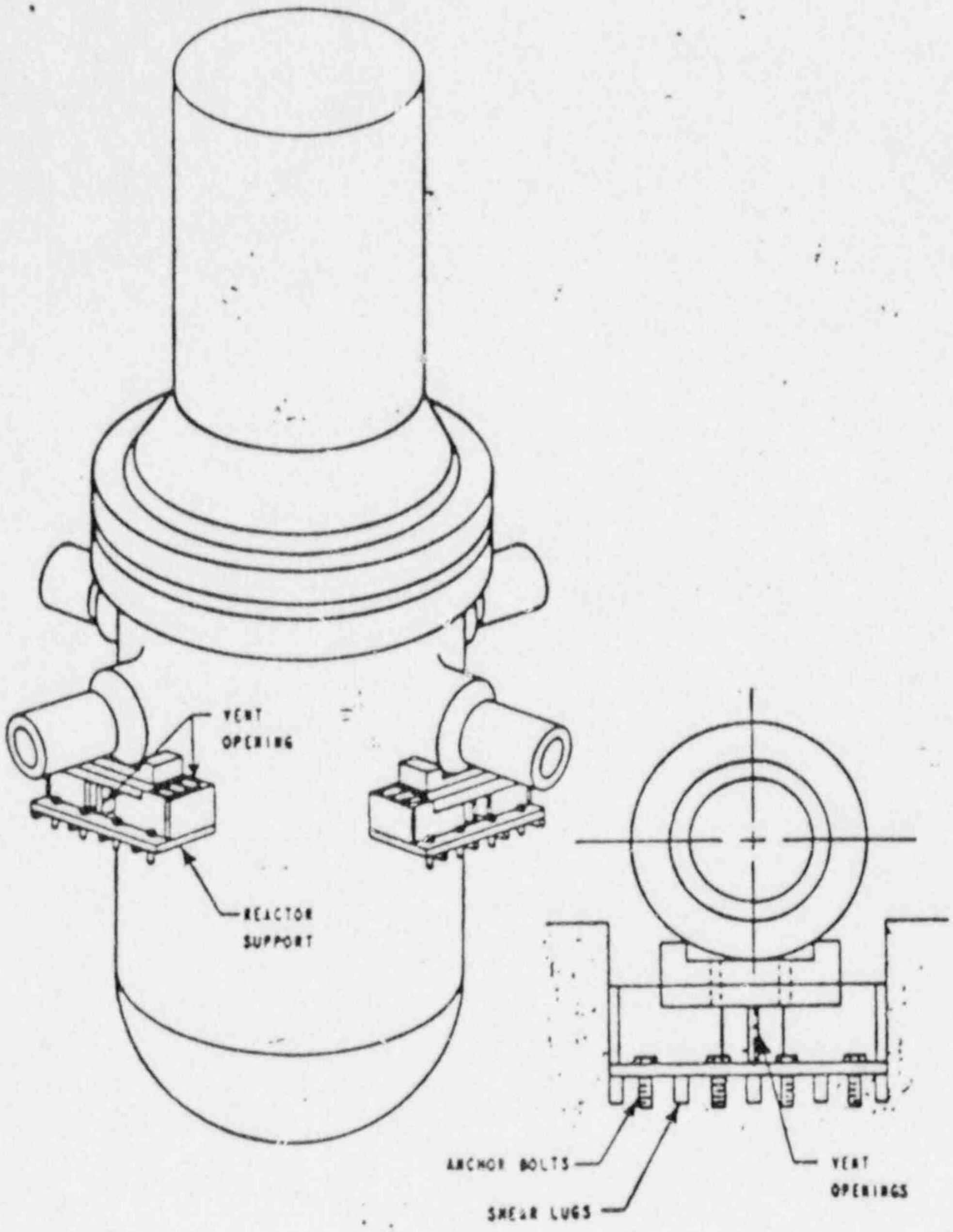


FIGURE A29

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Supports
Figure 5.5-7

50-395

WE

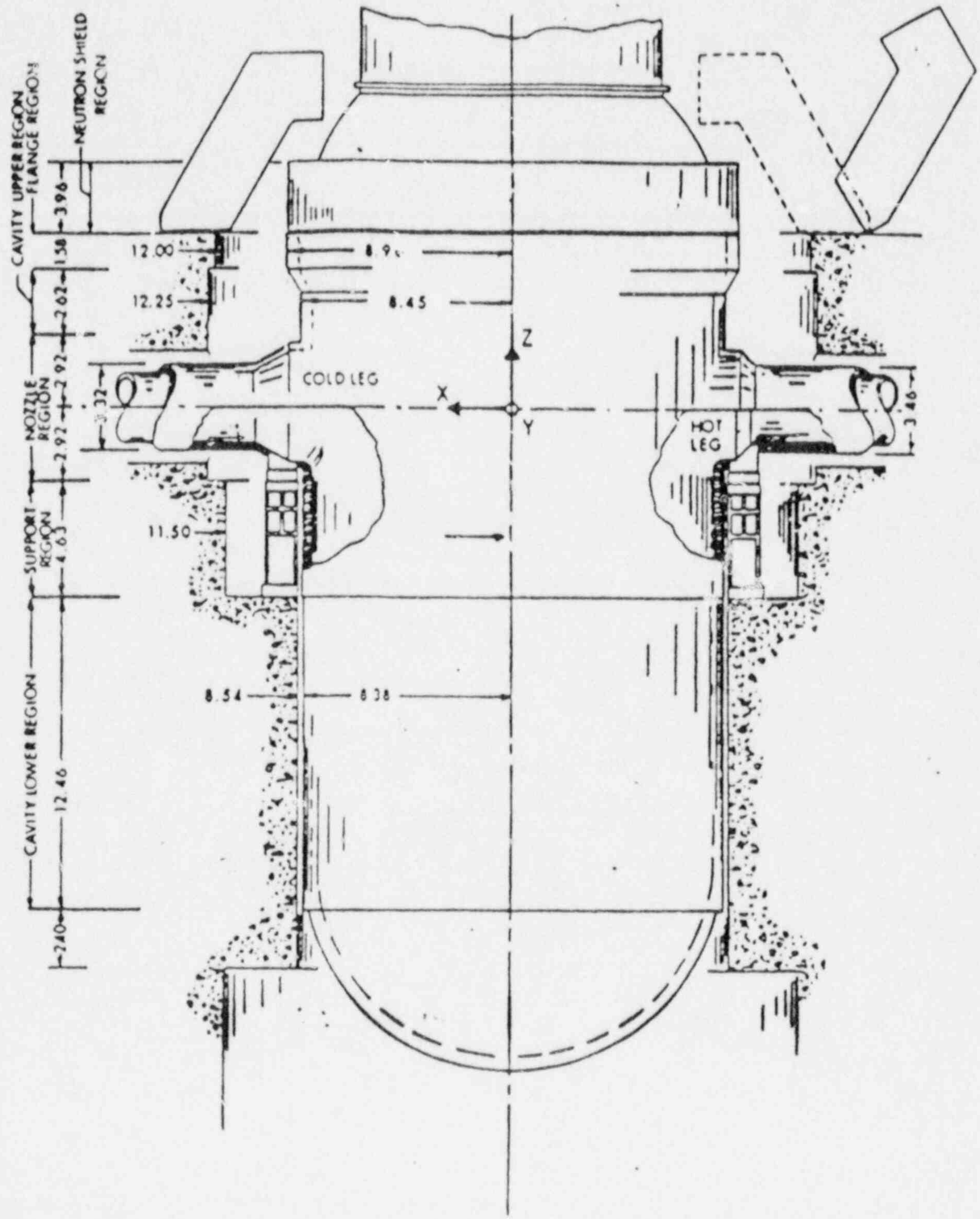


FIGURE A30

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE SEABROOK STATION - UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	WE 443/444 VERTICAL SECTION THROUGH REACTOR CAVITY FIGURE 6.2-26

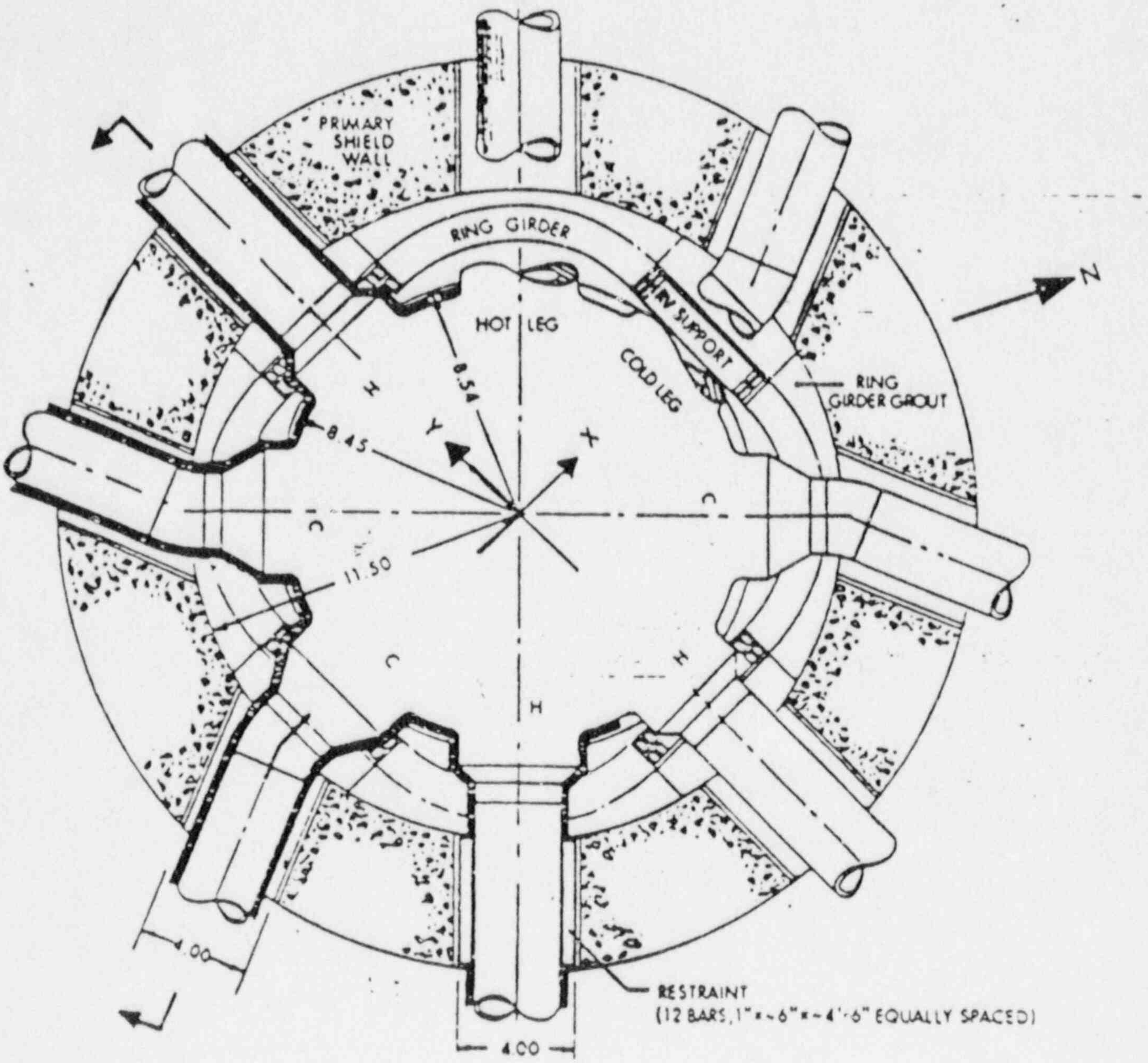
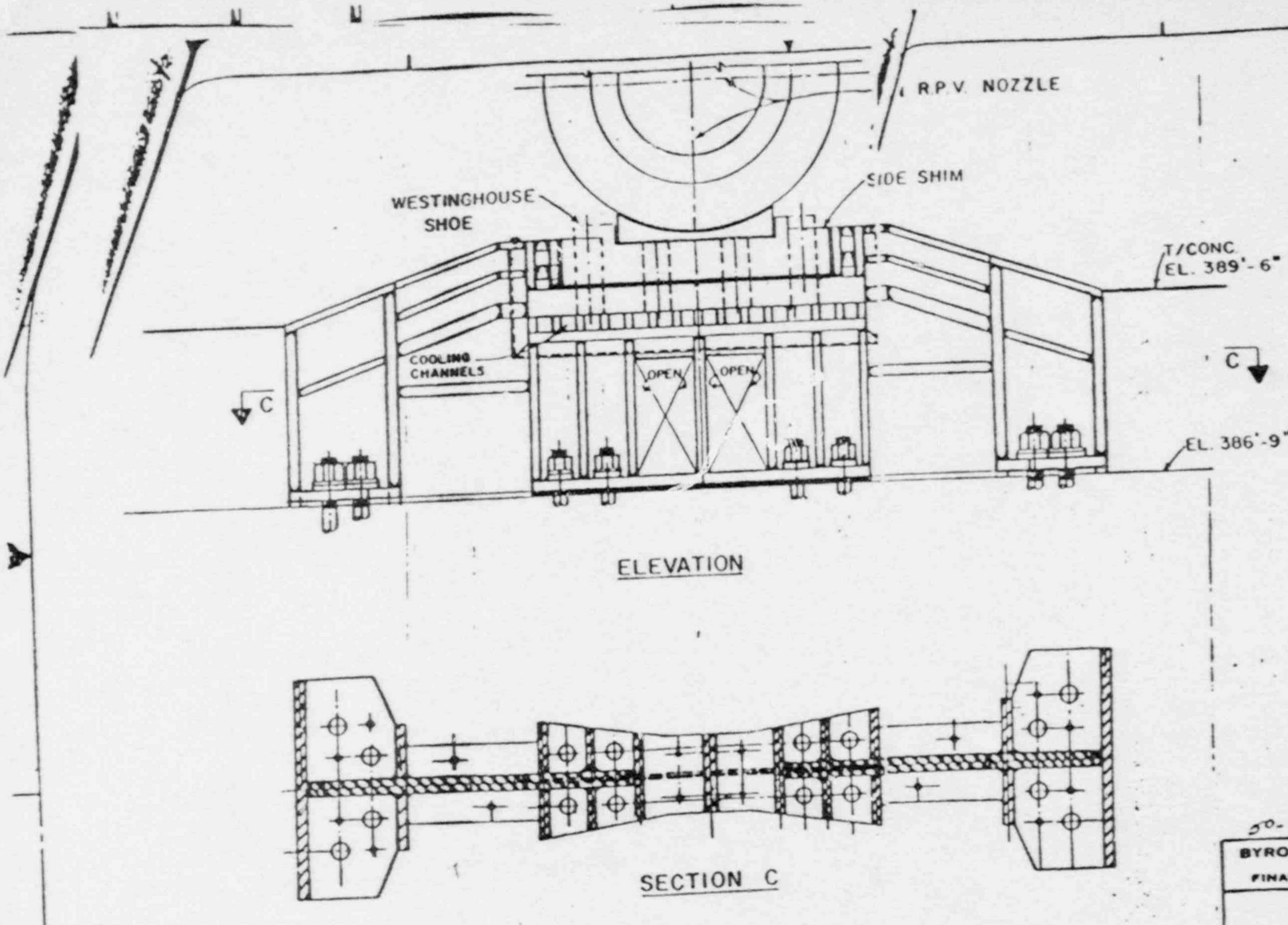


FIGURE A3.1

WE UEC 443/444

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE SEABROOK STATION - UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	PLAN VIEW OF REACTOR CAVITY AT ELEVATION OF NOZZLES
	FIGURE 6.2-27



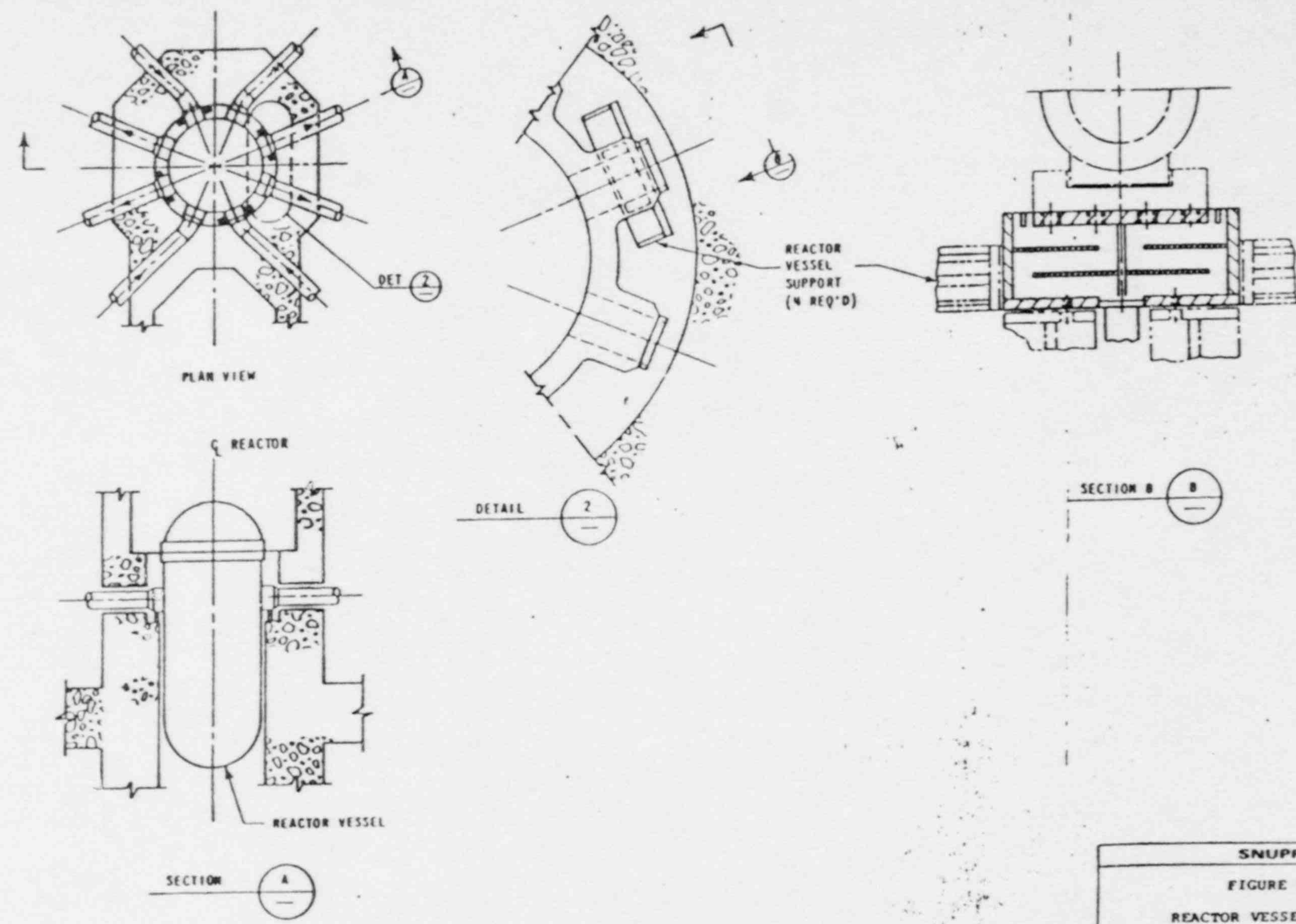
50-454, 455, 456, 457

BYRON/BRAIDWOOD STATIONS
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.9-4
REACTOR VESSEL SUPPORT

WE SL

FIGURE A32



SNUPPS

FIGURE 5.4-11

REACTOR VESSEL SUPPORTS

FIGURE A33

14. W. G. Hopkins, Chapter 7. Reactor Pressure Vessel Supports for Pressurized Water Reactors and Boiling Water Reactors, in report by V. N. Shah and P. E. MacDonald, Editors, Residual Life Assessment of Major Light Water Reactor Components - Overview, Volume 1, Idaho Natl. Eng. Lab., EGG-2649, Volume 1 (NUREG/CR-4731) dated June 1987.

15. R. C. Cipola et al., Aptech Eng. Serv., Inc., Requirements and Guidelines for Evaluating Component Support Materials Under Unresolved Safety Issue A-12, Electric Power Res. Inst. Report EPRI NP-3528, June 1984.

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General Electric BWR Plants

Facility Name	Docket No.
Humboldt Bay	133
Big Rock Point	155
Oyster Creek	219
Nine Mile Point 1	220
Dresden 2	237
Dresden 3	249
Millstone 1	245
Quad Cities 1	254
Quad Cities 2	265
Browns Ferry 1	259
Browns Ferry 2	260
Monticello	263

Babcock and Wilcox PWR Plants

Facility Name	Docket No.
Indian Point 1	3
Oconee 1	269
Oconee 2	270
Oconee 3	287
Three Mile Island 1	289
Crystal River 3	302
Rancho Seco	312
Arkansas Nuclear 1	313
Three Mile Island 2	320
Midland 1	329
Midland 2	330
Davis Besse 1	346
North Anna 3	404
North Anna 4	405
Bellefonte 1	438
Bellefonte 2	439
WPPS 1	460
WPPS 4	513
Babcock Std. 41	---

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13. Stone and Webster Eng. Co. engineering drawings of Neutron Shield Tank Assembly for the Surry Power Station of Virginia Electric and Power Company.

Combustion Engineer'ng PWR Plants

Facility Name	Docket No.
Palisades	255
Ft. Calhoun 1	285
Maine Yankee	309
Calvert Cliffs 1	317
Calvert Cliffs 2	318
St. Lucie 1	335
Millstone 2	336
San Onofre 2	361
San Onofre 3	362
Arkansas Nuclear 1	313
Arkansas Nuclear 3	368
WPPS 3	508
WPPS 5	509
Palo Verde 1	528
Palo Verde 2	529
Palo Verde 3	530
CE 80 Plant	---

Westinghouse Electric PWR Plants

Facility Name	Docket No.
Yankee Rowe	29
San Onofre 1	206
Conn. Yankee (Haddam Neck)	213
Ginna	247
Indian Point 2	247
Turkey Point 3	250
Turkey Point 4	251
H. B. Robinson 2	261
Surry 1	280
Surry 2	281
Zion 1	295
Zion 2	304
Cook 1	315
Cook 2	316
Beaver Valley 1	334
Beaver Valley 2	412
North Anna 1	338
North Anna 2	339
Byron 1	454
Byron 2	455
Braidwood 1	456
Braidwood 2	457

ISSUE 15: RADIATION EFFECTS ON REACTOR VESSEL SUPPORTS

DESCRIPTION

Historical Background

This issue was identified as a Candidate USI in NUREG-0705[44] but was recommended for further study before a judgment is made on its designation as a USI.

The potential problem is caused by the radiation embrittlement of structural materials. In the past, most neutron damage has been associated only with those whose energy is ≥1 MeV. However, it has also been recognized that neutrons whose energy is between 0.1 and 1 MeV also contribute to damage.[64] An upwards shift in the nil ductility transition temperature (NDTT) has been related to high fluence exposure of low energy neutrons. Essentially, all of the neutrons reaching the vicinity of a reactor vessel support structure (RVSS) have low energies because most of the fast neutrons (E>1 MeV) have been moderated or shielded in leaving the reactor vessel. The transition temperature for brittle failure of many structural steels begins in the neighborhood of -50°F, but after high exposure to a neutron fluence, the transition temperature can become as high as 200°F. This means that loss of fracture toughness may occur in the RVSS. Given a transient stress or shock, a rapidly propagating fracture of the RVSS and consequent movement of the reactor vessel could occur. An earthquake could provide such a transient mechanical loading.

Safety Significance

A large seismic event can cause an embrittled RVSS to fracture thereby allowing the reactor vessel to move. Such movement can then lead to a LOCA from the rupture of piping attached to the reactor vessel.

Possible Solutions

There are two possible solutions to this issue: (1) provide local heating and insulation for the RVSS to keep it well above the NDTT, and (2) reinforce the RVSS in those areas where fracture toughness loss may no longer allow the RVSS to be capable of meeting seismic requirements.

The first of these solutions will be considered as far as possible to develop both cost and occupational risk assessments for a candidate resolution. This general approach is necessary for the generic plant risk assessment because, for each plant where the problem really exists, the resolution will be plant specific.

The possible solution would probably only be required during the latter one-third of a reactor's life. Thus implementation of the solution is assumed to occur after two-thirds of the plants' operating lives have expired. From NUREG/CR-2800,[64] this vulnerability period is about 9.6 years for PWRs and 9.1 years for BWRs.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

The research needed to understand the problem has been underway since about 1979 but little data is available. The severity of the problem is not well documented, nor are the susceptible RVSS materials identified. Consequently, to represent the technical community opinion as to whether a problem exists or not, a probability of 0.5 is assumed. In addition, to represent the uncertainty of which RVSS materials may be susceptible, a vulnerable plant probability of 0.4 is also assumed. Thus, 20% of the BWR and PWR plants constructed and planned will be assumed applicable, i.e., 18 PWRs and 9 BWRs of those identified in NUREG/CR-2800[64] will be used for determining the risk parameters for PWRs and BWRs, respectively.

For both PWR and BWR plants, it is assumed that the accident scenario begins with an embrittled RVSS and a seismic event of large enough magnitude (0.2g acceleration) to cause the RVSS to fracture allowing the reactor vessel to move. That movement causes a LOCA (pipe blockage or rupture) for piping attached to it. Data from Oconee 3 and Grand Gulf 1 were used for determining risk parameters for PWRs and BWRs, respectively. The frequency of a rupture of RCS or loss of coolant flow due to a seismic event with an acceleration equal to, or greater than, 0.2g was assumed to be 7×10^{-4}/RY for both types of reactors.

For PWRs, the only WASH-1400[16] risk parameters presumed to be affected were SS_1, SS_2, and SS_3. The frequencies of SS_1, SS_2, and SS_3 were based on the assumptions that the probabilities of flow blockage or pipe rupture due to RV movement when the RVSS failed were 0.5, 0.6, and 0.8 for the SS_1, SS_2, and SS_3 cases, respectively. Considering the WASH-1400[16] Release Categories PWR 1 through 8 and substituting the calculated[64] risk parameters in the dominant sequences and frequencies for Oconee 3 resulted in a reduction in core-melt frequency of 1.1×10^{-5}/RY. The public risk reduction associated with this frequency reduction is 16 man-rem/RY.

For BWRs, the only WASH-1400[16] risk parameter presumed to be affected was SS. The frequency of SS was based on the assumption that the probability of flow blockage or pipe rupture due to reactor vessel movement when the RVSS failed was 0.8. Considering the WASH-1400[16] Release Categories BWR 1 and 2 and substituting the calculated risk parameter in the dominant sequences and frequencies for Grand Gulf 1 resulted[64] in a reduction in core-melt frequency of 5.9×10^{-7}/RY. The public risk reduction associated with this frequency reduction is 4.2 man-rem/RY.

Therefore, the total public risk reduction during the vulnerability period for 19 PWRs and 10 BWRs is 3,100 man-rem.

Cost Estimate

The resources required to implement the safety issue resolution at each of the applicable plants are labor and equipment. It is assumed that heaters will be

attached to 4 reactor vessel support columns and that mounting hardware, wiring, metal-sheathed heating cable, switchgear, transformer, and a power controller will be installed. It is also assumed that the equipment would be installed during reactor outages for other purposes and that no replacement power cost would be charged to the solution. It is further assumed that access to the reactor cavity would be possible for the heater installation. The cost of equipment in each affected plant is estimated to be $52,000 for 4 strip heaters clamped to support columns; mounting hardware, materials, and wiring; power controller; switchgear and transformer; and metal-sheathed heating cable. A 50% contingency was included in this cost.

Costs for equipment, labor, and license amendment are estimated to be $520,000 for each of the 14 backfit plants. Thus, the industry cost for implementing the possible solution is $7.3M.

Industry operation cost for maintenance repair, replacement, and power associated with the solution are estimated to be $144,000/RY over the vulnerability period for each of the 14 backfit plants. Thus, the total industry cost for operations and maintenance is $18M, assuming the systems would only be operated during the last one-third of each reactor's life.

NRC costs associated with development and implementation of the possible solution are estimated to be $1.5M for 14 backfit plants.

NRC operation and maintenance review costs associated with the solution are estimated to be $283,000/RY over the vulnerability period of the 14 plants.

Summing all costs outlined above,

Total Cost = $7.3M + $18M + $1.5M + $0.283M = $27M.

Value/Impact Assessment

Based on a total risk reduction of 3,100 man-rem for the affected plants, the value/impact score is given by

$$S = \frac{3{,}100 \text{ man-rem}}{\$27\text{M}}$$

$$\cong 115 \text{ man-rem/\$M}.$$

Other Considerations

There are three factors to be considered that would have an effect on the overall conclusion to be drawn on this issue. These are:

(1) Implementation Occupational Risk Increase

Implementation of the solution would require work to be performed inside the containment of the 14 plants considered for backfit. For each of these plants it is estimated that 54 man-weeks (2,160 man-hours) of labor would be in radiation zones. Assuming a dose rate of 100 millirem/hour, the total occupational dose increase from implementation is 3,020 man-rem.

(2) Operation and Maintenance Occupational Risk Increase

Following implementation of the solution, operation and maintenance would be required during the vulnerability period. Assuming a yearly rate of 520 man-hour/plant for operation and maintenance, the total occupational dose increase is estimated to be 6,490 man-rem.

(3) Accident Avoidance Occupational Risk Decrease

It is estimated that implementation of the solution will preclude NOTT rupture accidents during the last one-third of the remaining lives of all 27 plants. The total occupational dose reduction associated with this factor is estimated to be 39 man-rem.

Summing the above three factors, the total occupational risk increase is [3,020 + 6,490 -39] man-rem or 9,471 man-rem. Consideration of these factors demonstrates that a total occupational risk increase of 9,471 man-rem would be incurred in order to produce a reduction in public risk of 3,100 man-rem. Inclusion of these factors in the value/impact score calculation would result in a negative score.

CONCLUSION

Based on the public risk reduction and value/impact score, this issue would have a medium priority ranking. However, the occupational dose increases associated with this issue far outweigh the public risk reduction and this issue should remain as a LOW priority issue until new data on the severity of the concern is available.

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

October 7, 1987

MEMORANDUM FOR: William Kerr, Chairman, ACRS

FROM: Victor Stello, Jr., EDO

SUBJECT: ACRS COMMENTS ON THE EMBRITTLEMENT OF STRUCTURAL STEEL

In our response to you on August 4, we described the actions underway and being planned to take account of neutron embrittlement of reactor support structures. One important action was a review and assessment of the current state of knowledge of flux, fluence and embrittlement relative to the materials and conditions of neutron shield tanks and support columns that form the primary elements of concern for reactor support structures.

We have now received a report from the HSST Program Staff at Oak Ridge which summarizes the technical information they have assembled relevant to the issue of low dose rate embrittlement of the steel in the support structures of LWRs. A copy of that report is enclosed. The ORNL summary coincides with our evaluation that the neutron shield tanks and support structures do not appear to pose any safety problems. The embrittlement can be conservatively predicted as an increase in transition temperature of the steel of as much as 400°F, based on extrapolation from the HFIR data. This is a factor of 2 to 3 times as much embrittlement as might have been predicted prior to the revelation from the HFIR surveillance program and represents an embrittlement rate 10 to 20 times higher than for test reactor irradiations. The fluence at the HFIR vessel is close to the predicted value, but the flux is 10 to 100 times less than a power reactor cavity region (where the support structures are located) and a factor of 10^4 less than typical test reactors. With the flux reduction schemes now in place, however, the fluence in the shield tank and support column region might decrease about an order of magnitude, with a corresponding decrease in the embrittlement. These structures are in compression, so even with a 0.2 g earthquake, the tensile stresses generated appear to be too low for fracture initiation. The only way to predict enough tensile stresses to initiate fracture of these structures is from the jet forces from a large LOCA of the main cooling lines. By this time, however, the reactor would already be in a serious accident situation, so that failure of the support columns or shield tanks would have little effect.

Clearly, an embrittlement level of some 400°F in the support structures is not a welcome situation. We will incorporate this issue into the RES Materials Engineering Branch programs on fracture, irradiation effects, and neutron

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dosimetry, to refine the predictions and structural implications, with the idea of developing recommendations for limits on embrittlement or methods for mitigation such as heating of the shield tank water and support columns. We note that such a "fix" was already proposed for resolution of "Issue 15," Radiation Effects on Reactor Vessel Support Structures, NUREG-0933, November 30, 1983, a copy of which is enclosed.

Victor Stello, Jr.
Executive Director for Operations

Enclosures:
1. ORNL Letter Report
2. Issue 15

cc w/encls: P. G. Shewmon

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 8, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON MEMORANDUM FROM VICTOR STELLO, JR., EDO, DATED OCTOBER 7, 1987 REGARDING THE EMBRITTLEMENT OF STRUCTURAL STEEL

We are concerned and perplexed by your memorandum of October 7 (referenced). There you conclude that, "the neutron shield tanks and support structures do not appear to pose any safety problems. The embrittlement can be conservatively predicted as an increase in transition temperature of the steel of as much as 400°F." You support your conclusion with the statement, "These structures are in compression, so even with a 0.2 g earthquake, the tensile stresses generated appear to be too low for fracture initiation."

Studies indicate that the highest risk of sudden pipe rupture in the primary system arises from the failure of supports of a major component. We can see no reason to be sanguine about the safety of operating nuclear power plants with the largest, heaviest component in the primary system supported on a structure, parts of which are fully brittle. This is unsafe by any type of analysis. The average stress may be compressive, but it isn't the average stress which would determine the failure of the structure. These supports are welded structures so there are regions with tensile stresses as high as the yield stress. They operate in a temperature gradient so there will be thermal stresses which are tensile in the cold (less ductile) regions. They are uninspected so we have no real idea of what kinds of flaws are present, and flaw size is critical in any meaningful failure analysis.

It would be imprudent to operate nuclear power plants with brittle structures supporting the pressure vessels. We recommend that an early effort be made to gain answers to the following questions:

1) Is the temperature of the support structure of the reactor pressure vessel in any operating plant now below its nil ductility transition temperature (NDTT)?

2) Will the temperature of the support structure of the reactor pressure vessel in any operating plant drop below its NDTT before the plant's license expires?

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We hope and suspect that the answer to the first question is "no." However, it is not clear that we know this with any certainty. The research program mentioned in your memorandum is necessary and desirable, but it is not clear that it will answer the safety-related questions noted above in a timely manner.

Sincerely,

W Kerr

William Kerr
Chairman

Reference:
Memorandum from Victor Stello, Jr., EDO, to William Kerr, ACRS, dated October, 7, 1987, Subject: ACRS Comments on the Embrittlement of Structural Steel

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

JAN 2 8 1988

MEMORANDUM FOR: William Kerr, Chairman, ACRS

FROM: Victor Stello, Jr., EDO

SUBJECT: ACRS COMMENTS ON THE EMBRITTLEMENT OF STRUCTURAL STEEL IN REACTOR SUPPORT STRUCTURES

This memorandum is in reply to your December 8, 1987 letter concerning the embrittlement of structural steel in reactor support structures. In summary, we have restudied the 1983 NRC analysis of this issue as well as the more recent ORNL data that emerged from the HFIR analysis, and, as discussed in more detail below, our judgment is that this issue does not constitute an immediate threat to the safety of nuclear power plants that the NRC has licensed.

With regard to the two questions posed in your letter on the current and future status of the NDTT of the reactor support structures, our responses are:

(1) We have not studied enough plant specific information to unequivocally answer this question; the generic analyses that we have done to this point have not led us to identify any specific plants where the support system NDTT is above the operating temperature.

(2) Our judgment is that the NDTT of the support system structure in some plants may exceed the operating temperature before the end of the plant license.

We note that both of these responses are judgmental because we do not yet have sufficient information to be more definite, but we intend to perform such plant specific analyses in our continuing program.

We have responded to your questions as you posed them -- in the context of support structure material NDTT relative to the operating temperature of the support structure. While we agree that NDTT can provide a convenient indication of fracture toughness, and that one can put the problem in perspective by referring to NDTT, we note that resolution of this issue will be done in terms of a comprehensive fracture mechanics analysis, combining the stresses, the material properties (of which the NDT temperature is an important part), and a flaw size large enough to initiate a fracture.

Based on the analyses that have been performed and on our engineering judgment, we believe that the embrittlement of reactor vessel support structures does not pose an immediate threat to the safety of nuclear power plants. An important consideration that has contributed to our belief is that the failure of vessel supports does not lead to the failure of the piping. Work done by LLNL on a different project used a modelling assumption that support failure automatically resulted in piping failure. For that project, this was a conservative,

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simplifying assumption. The staff has reviewed other analyses that consider the effects of component support failure on pipe breaks. Those analyses show that support failure is not likely to result in pipe failure. Further, recent structural analyses performed by the staff, and employing very conservative assumptions, show that reactor vessels can be supported on their inlet and outlet piping alone, without benefit of the support structures. In addition, our preliminary investigations indicate that shield-tank and column supports, although exposed to neutron fluxes at mid-height of the core, and skirt-type supports have a low likelihood of failure because (1) there is a large degree of redundancy associated with these designs, (2) the loading for these designs is primarily compressive, (3) the columns have no welds and essentially no thermal stresses in the high-flux zone, and (4) the end-of-life fluence for the skirts is very low.

Investigations regarding the impact of the HFIR data on the likelihood of failure for other vessel support designs are still underway. While we recognize that there may be tensile stresses within the structures that could cause cracking, and that there may be more embrittlement than originally anticipated, we do not believe that there is an immediate threat to the safety of nuclear power plants.

The Department of Energy is aware of this issue of embrittlement of reactor support structures having recently conducted a workshop on this topic at Sandia National Laboratory. The conclusion of the workshop was that there is no immediate problem with LWR support structures. However, the workshop focussed primarily on shield tanks and reached its conclusion primarily on the basis of industry work which did not emphasize embrittlement rate effects, nor did it include some of the additional sources of tensile stresses that we have been considering such as welding residual stresses or excessive friction in the slider assembly. EPRI has prepared an analysis of the support structure embrittlement on behalf of the industry that shows an increase in NDTT of 200°F for a 40-year life rather than the 400°F as determined in the ORNL work. We must point out that at this time, our contractor ORNL has reviewed the EPRI analysis and does not agree with it. EPRI, in cooperation with the NUPLEX Technical Subcommittee is, however, continuing to investigate the impact of embrittlement on the neutron shield tank and columnar support designs.

The NRC view of the DOE and industry work is that while we agree with the conclusion of no immediate safety problem, we are working to develop an adequate data base of information on designs, toughness, embrittlement, loads and flaws so that we can make a prompt and correct assessment of the situation.

We are actively researching this issue with the objectives of (1) identifying those plants with support structures that currently are susceptible to brittle failure at the operating temperature, (2) identifying those plants with support structures that may become susceptible to brittle failure at the operating temperature, and (3) defining appropriate actions for those plants. To achieve the first two objectives, we have asked the ORNL HSST Program staff to summarize the reactors by categories of support structures and submit, by February 1988, an estimate of time and cost for analysis of up to five distinct support structures. Our intent is to complete the analyses by the end of FY88. The ORNL analyses will include considerations of rates of embrittlement, peculiarities of support structures, variability of both the materials involved

and the potential loads, and combinations of stresses resulting from fabrication effects and normal and accident loading conditions. From the results of these analyses, we expect to be able to achieve our first two objectives. Subsequently, the staff will define appropriate actions for any plants that have, or are expected to have, supports susceptible to brittle failure.

The issue of support structure degradation involves the combined effects of low dose rate, low temperature, and spectrum softening on irradiation damage; analyses will make use of the HFIR results as these are the only such results currently available which include at least combined low dose rate and low temperature effects. Spectrum softening effects will be included in the ongoing research program. Final resolution of these questions on combined effects will require several years of work before enough data can be developed to provide a confident answer. A principal reason for the extended time for resolution comes from the need for study of materials from decommissioned reactors, such as the testing of steel to be removed from the shield tank of the Shippingport reactor. We plan to remove the samples in the Spring of 1988 with the testing to follow as soon as practical. Samples from other reactors undergoing decommissioning will be obtained and tested as they become available in the future. We also are in touch with Naval Reactors to determine if they have any pertinent information which could be shared with us on this issue.

Finally, the staff will review the regulations covering protection against brittle fracture for components and their supports and will review whether further requirements are needed, such as inspection or sampling of materials.

As we continue to develop pertinent information on this subject, we will keep you informed.

Original Signed by
V. Stello

Victor Stello, Jr.
Executive Director for Operations

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EDO R17	

SEE ATTACHED FOR PREVIOUS CONCURRENCES

RES:MEB	RES:DD/DE	RES:DE	RES:DD	RES:DIR	NRR:DEST	NRR
CSerpan	RBosnak	GArlotto	DRoss	ESBeckjord	LCShao	TEMurley
1/4/88	1/5/88	1/ /88	1/ /88	1/6/88	1/ /88	1/ /88