WESTINGHOUSE PROPRIETARY CLASS 3

WCAP-12345
REVISION 1

WESTINGHOUSE OWNERS GROUP

IRRADIATION EFFECTS ON REACTOR VESSEL SUPPORTS

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1.0 INTRODUCTION

The Westinghouse Owner's Group has received technical information concerning the potential for damage to reactor vessel supports by low flux, low fluence irradiation. This summary of our view of the issue has been prepared from the proceedings of a meeting held with the USNRC on June 15, 1989.

This report contains an independent assessment of the reactor vessel support irradiation damage issue, and a critique of the study performed earlier by Oak Ridge National Labs [1]. A review of the range of support configurations which exist in Westinghouse plants is provided, based on a survey conducted among all the Westinghouse plants. Also included is a summary of the key conservatisms in the integrity studies performed to date.

2.0 IRRADIATION DAMAGE ASSESSMENT

This issue of irradiation damage at low temperature and low fluence has been considered inconsequential since the middle 1960's, when research was redirected to higher fluence and higher temperature conditions. This issue arose again recently with the publication of the ORNL HFIR reactor surveillance results[1]. In this section a summary of all available data will be given, along with a fresh look at the HFIR results.

The best summary of available data is that compiled by Porter[2] of U.S. Steel, and his results are summarized in Figure 2-1. He surveyed the literature, and did a statistical study, constructing tolerance limits on the available data. Note that the slashed points are the only data irradiated at higher than 250^{o}F. Porter's data included both carbon and low alloy steels, and associated welds, and a partial list of materials is shown in table 2-1.

Recent test results from the Shippingport shield tank material provide a good assessment of the level of irradiation damage which might be expected to occur in a typical reactor vessel support system located immediately opposite the vessel core region. These results[3] are summarized in table 2-2, and show an irradiation-induced shift of a maximum of 52F, with a dpa dosage of 0.00167, and fluence = 6.1E17. Plotting these results on Porters data, we find that they fit directly on the mean curve, as shown on Figure 2-2. Note that both 15 and 30 ft.lb. shifts were used for completeness, since the 15 ft.lb shift is often used for carbon steels.

The surveillance results from the HFIR reactor are shown in Figure 2-3. Note that the data all still fit within the 75-90 tolerance bounds developed by Porter. Looking at the data by themselves shows a different slope than Porters, but there appears to be a good reason for such a disparity. The HFIR reactor contains a thick beryllium reflector which prevents a large percentage of the high energy neutrons from reaching the tank wall. Therefore the dpa is a much more accurate measure of total irradiation exposure. The Shippingport and HFIR results compare better based on dpa, as seen in Figure 2-4, but the comparison is marred by the ORNL calculation, which considered only energies greater than 0.1 MeV. This calculation should be redone to consider all energies, as the Shippingport calculations have done. An example of the energies missed in the ORNL calculation is given in table 2-3. Figure 2-5 shows the HFIR results in terms of both fluence (> 1 Mev), and dpa (>0.1 Mev) and illustrates the different slopes which result.

The HFIR results are not typical of those for a power reactor, because the energy spectrum has such a low proportion of low energy neutrons. It is our belief that use of these results is misleading for the vessel support issue. The more appropriate data are from Shippingport, and they match Porter's data very well. Porters data base was used in earlier years to conclude that no integrity issue exists for low fluences, and the same conclusion appears to hold today. The fluence calculations need to be redone for the HFIR reactor, and the whole issue reassessed at that time.

TABLE 2-1

MATERIALS INCLUDED IN PORTERS DATABASE

ASTM A106
ASTM A201
ASTM A212B (HOT ROLLED; NORMALIZED)
E7016 WELD
AISI C 1019
ASTM A203
ASTM A285 (HOT ROLLED; NORMALIZED)
ASTM A293
ASTM A301B (NORMALIZED, ANNEALED)
ASTM A302B, ASSOCIATED WELDS
ASTM A336
ASTM A353
HY 65
HY 80
E 10016 WELD
USS "T-1" STEEL
E 12016 WELD
DUCOL W30, ASSOCIATED WELDS
2.25 CR - 1 MO
Ni-Mo-Cu-V (AUSTEMPERED; QUENCHED)

TABLE 2-2

SHIPPINGPORT RESULTS

USED 15 FT-LB AND 30 FT-LB SHIFTS

LOCATION	FLUENCE	Dpa	30 FT-LB SHIFT	15 FT-LB SHIFT	
3,9,2,8 WR	6.10×10^{17}	0.00167	52F	45F	INNER HALF, INNER WALL(3/89 PROG RPT.)
3,9,2,8 WR	2.03×10^{17}	0.00056	40F	38F	OUTER HALF, INNER WALL(3/89 PROG RPT.)
3,9,2,8 TR	6.10×10^{17}	0.00167	20	30	INNER HALF, INNER WALL(4/89 PROG. RPT.)
3,9,2,8 TR	2.03×10^{17}	0.00056	30	30	OUTER HALF, INNER WALL(4/89 PROG. RPT.)

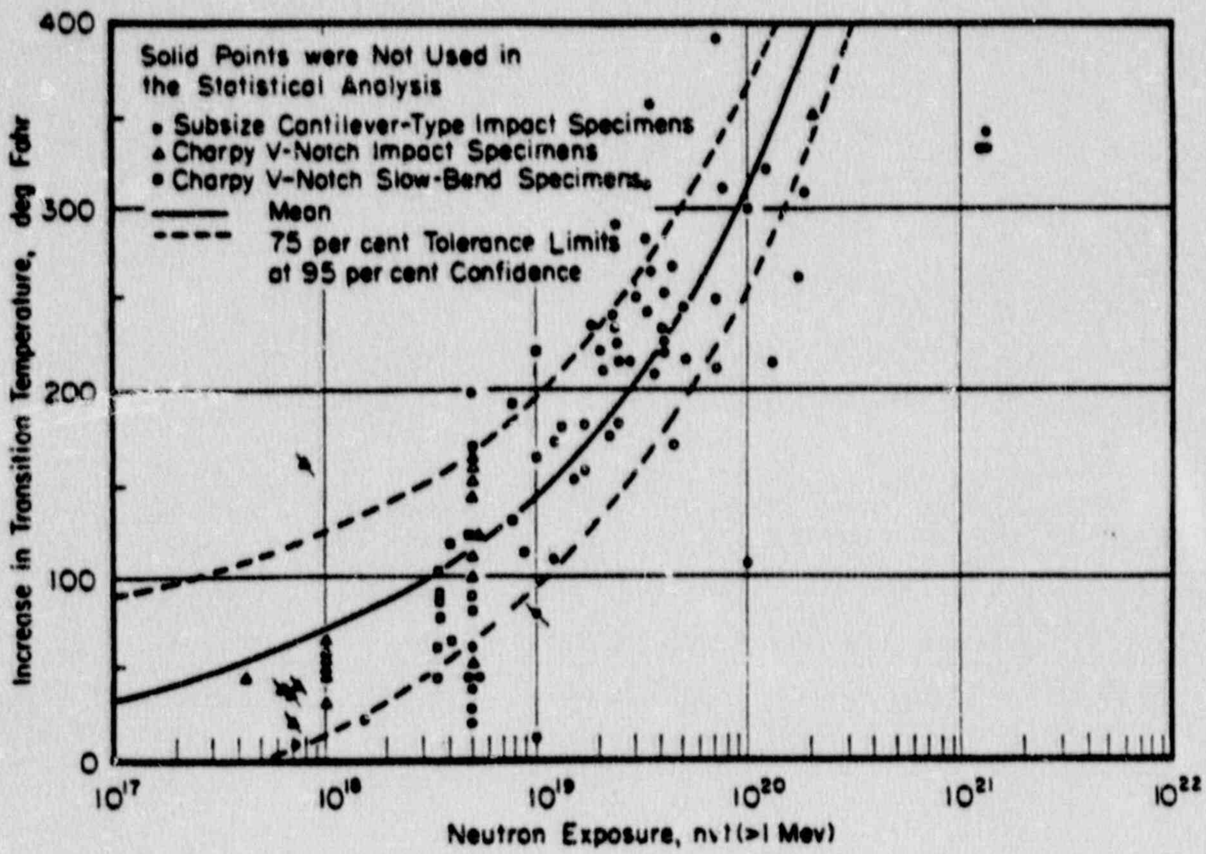
TABLE 2-3

DAMAGE CROSS SECTIONS BY GROUP

GROUP	LOWER ENERGY LIMIT	DAMAGE CROSS SECTION (B)			
		σ_{d1}	σ_{d4}	σ_{d1}'	σ_{d4}'
1	7.79 MeV	1374	6895	456.2	1826
2	6.07	1365	5743	537.9	1726
3	4.72	1261	4633	598.7	1682
4	3.68	1847	4248	641.5	1551
5	2.86	1799	3583	709.6	1340
6	2.23	1531	2696	744.8	1188
7	1.74	1308	2054	594.1	916.5
8	1.35	1114	1573	469.8	684.8
9	1.05	826	1095	347.0	471.0
10	0.821	683	789	239.8	289.3
11	0.639	843	843	285.2	290.2
12	0.498	535	535	163.7	167.3
13	0.387	729	729	272.1	274.6
14	0.302	402	402	109.2	111.0
15	0.235	327	327	118.7	119.9
16	0.183	268	268	148.3	149.2
17	0.143	264	264	82.57	83.32
18	0.111	176	176	32.77	33.42
19	86.5 keV	192	192	91.52	92.08
20	67.4	138	138	44.35	44.83
21	40.9	115	115	55.59	55.99
22	24.8	350	350	64.65	64.90
23	15.0	21.8	21.8	5.851	5.890
24	9.12	31.6	31.6	13.95	13.95
25	5.53	28.1	28.1	19.45	19.45
26	3.35	11.2	11.2	7.757	7.757
27	2.03	6.7	6.7	5.714	5.714
28	1.23	4.2	4.2	4.691	4.691
29	0.749	1.7	1.7	—	—

$\left.\begin{matrix}\text{Groups 19–29}\end{matrix}\right\}$ NOT CONSIDERED TO DATE FOR HFIR

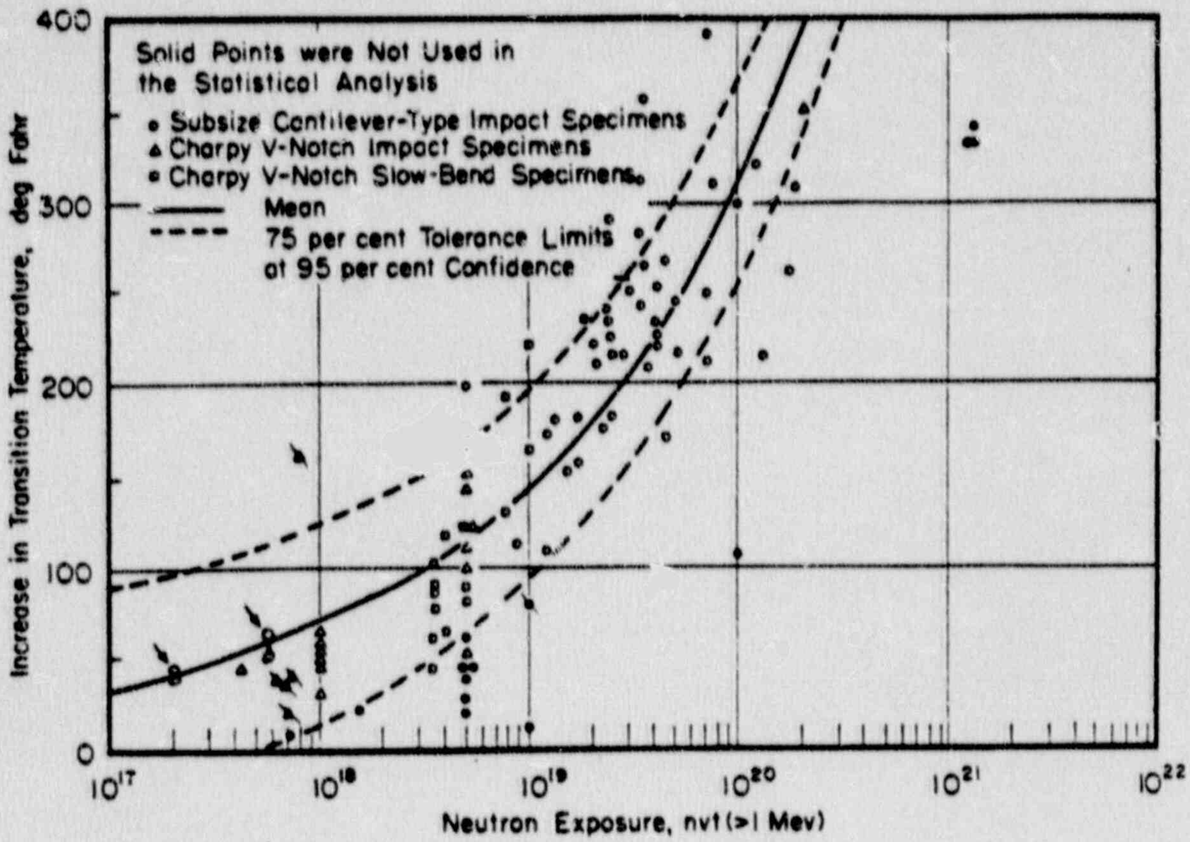
FIGURE 2-1



- LOW TEMPERATURE IRRADIATIONS
- CARBON AND LOW ALLOY STEELS
- STATISTICAL BOUNDS DEVELOPED

SUMMARY OF DATA BY PORTER, ASTM STP 276, 1959

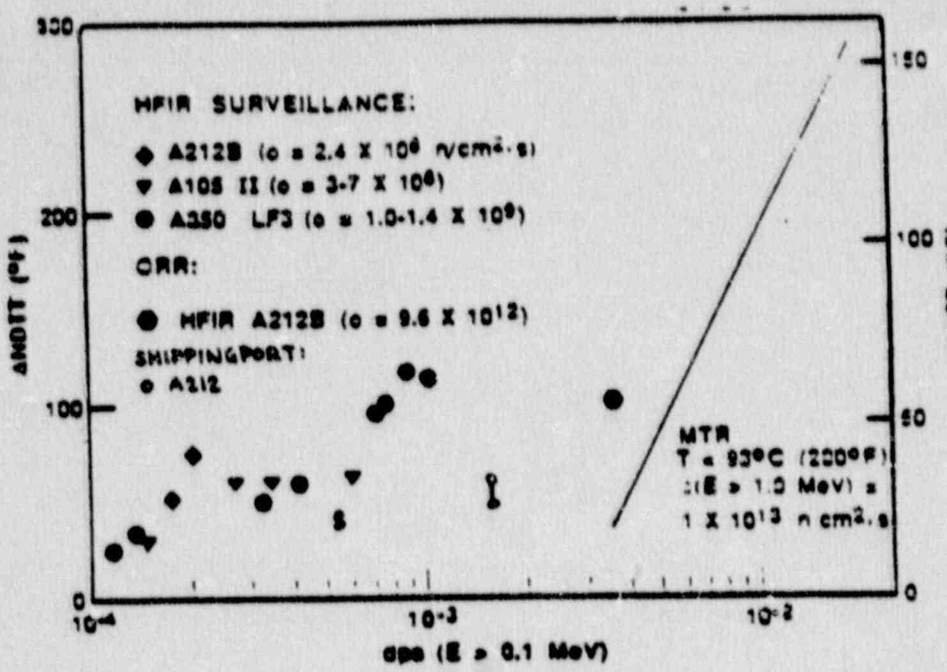
FIGURE 2-2



o SHIPPINGPORT SHIELD TANK RESULTS
NOTE: LOWER POINT IS 15 FT-LB SHIFT, UPPER IS 30 FT-LB SHIFT

EFFECT OF NEUTRON RADIATION ON THE NOTCH TOUGHNESS OF CARBON AND ALLOY STEELS IRRADIATED BELOW 500 F: PORTER (SLASHED POINTS IRRADIATED ABOVE 250 F)

FIGURE 2-3

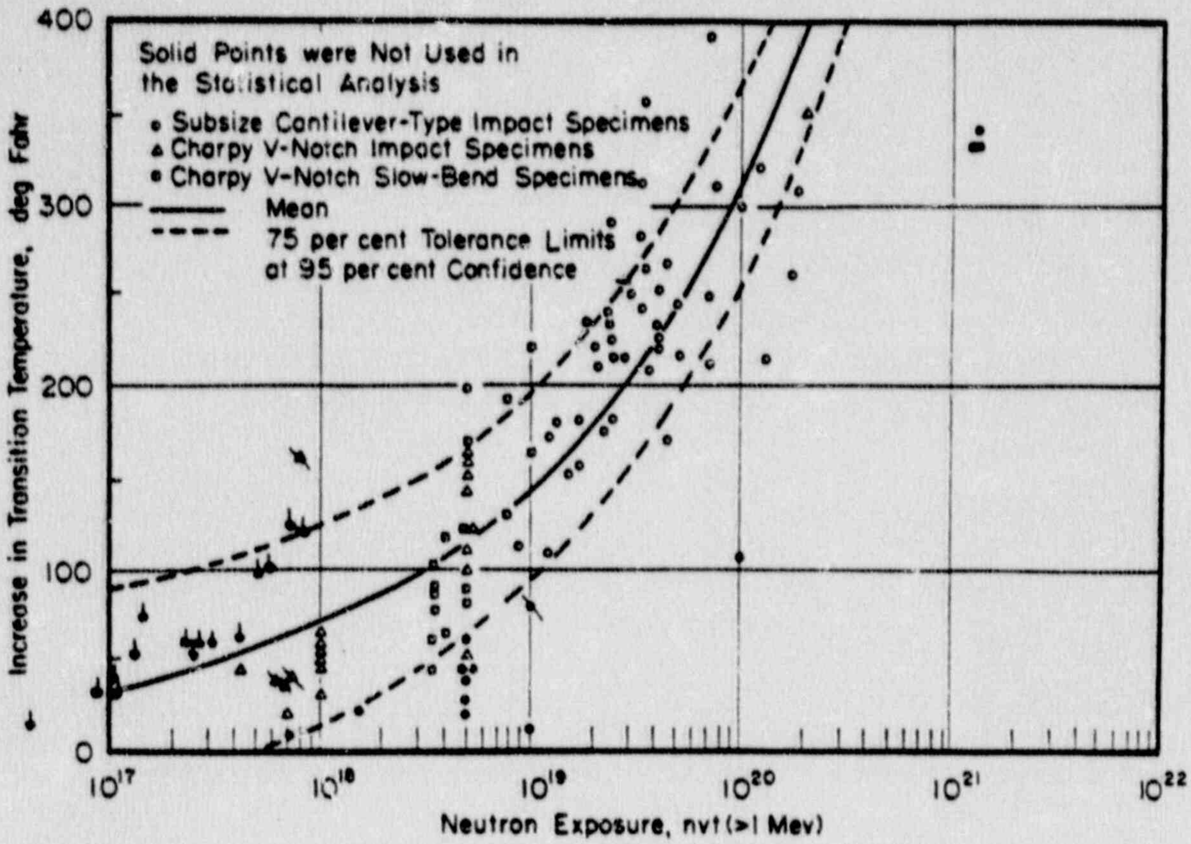


○ SHIPPINGPORT (ALL ENERGIES)

◆ ● ▼ HFIR

COMPARISON OF SHIPPINGPORT AND HFIR DATA, BASED ON dpa

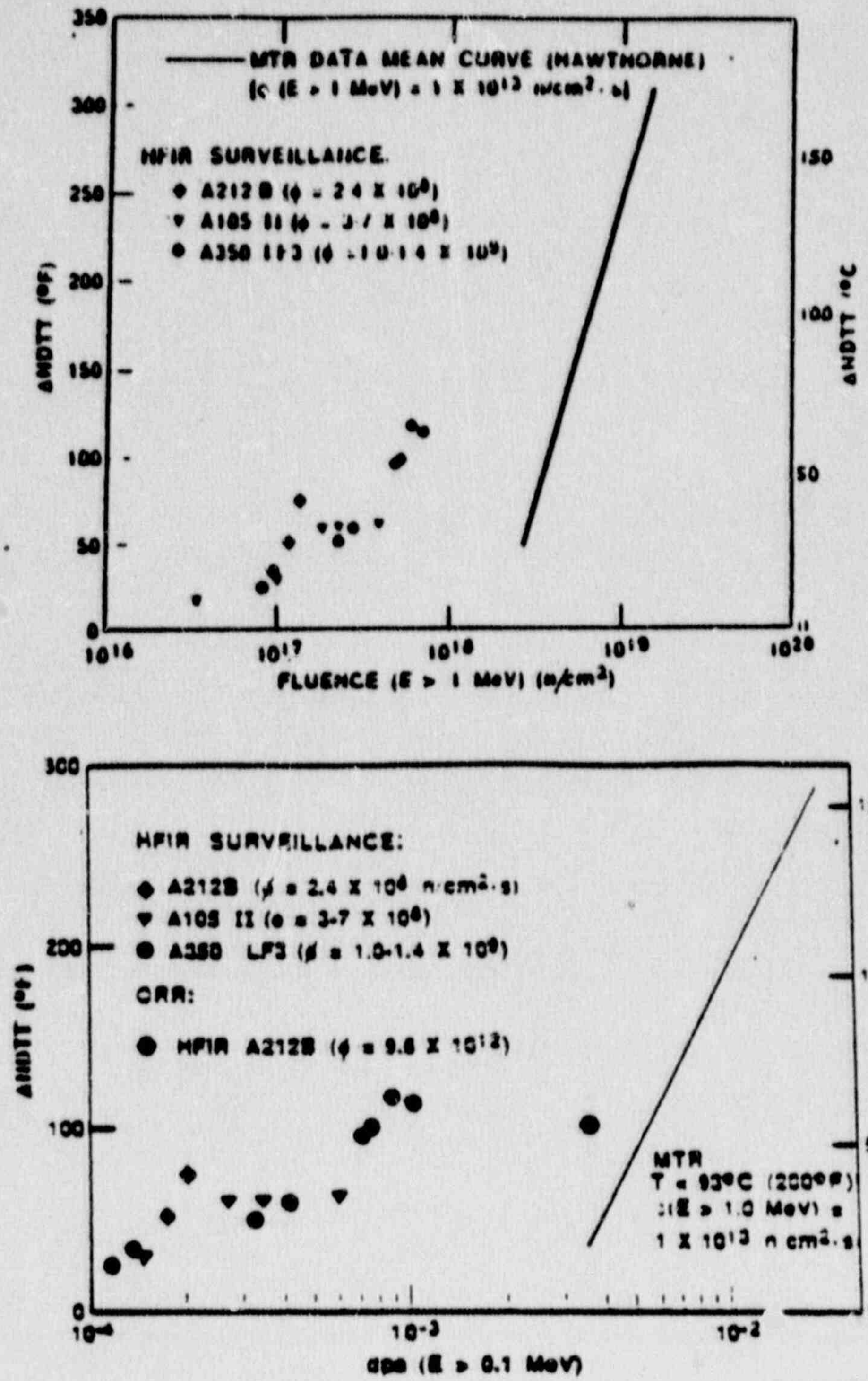
FIGURE 2-4



HFIR SURVEILLANCE RESULTS

EFFECT OF NEUTRON RADIATION ON THE NOTCH TOUGHNESS OF CARBON AND ALLOY STEELS IRRADIATED BELOW 500 F: PORTER (SLASHED POINTS IRRADIATED ABOVE 250 F)

FIGURE 2-5



COMPARISON OF HFIR SURVEILLANCE DATA: FLUENCE > 1 Mev vs dpa

3.0 FRACTURE ANALYSIS CRITIQUE

The fracture analysis performed by Cheverton et.al.[1] appears to be technically sound, and there are only a few items worthy of discussion. The strain rate effect on toughness is a real effect, but the use of strain rates equivalent to impact loads seems unrealistic for a seismic event. The strain rate of 0.1 inch/inch per second is a realistic upper bound. The loadings used are upper bound design loadings, and are therefore very conservative. Realistic loadings have not been calculated for the Trojan supports. Elimination of the need to consider large break loca loads for Turkey Point by utilizing leak before break could significantly increase the critical crack size for this plant to the point that a concern would not exist.

As stated in the ORNL report, low cycle fatigue is not viable mechanism for creation of flaws on the order of the critical crack size calculated, thus such flaws would have to exist at the time of fabrication. ORNL further stated that at the two locations considered for the Trojan supports other than the grout hole, the critical crack size is the full width of the flange (16 inches) and that a flaw of this size would be readily detected during fabrication. The credibility of the existance of a flaw of critical crack size magnitude relies on the validity of the assumption that such a flaw exists at the flame cut grout hole. We believe that the existance of a 0.4 inch flaw at the grout hole to be unlikely especially, as was mentioned in the ORNL report, since the flame cut hole was dressed. The grinding operation employed during the dressing operation would lower the potential for any pre-existing flaws of critical crack size magnitude. The net effect is that the likelihood of cracks anywhere in the support configuration is very low, either from fabrication or service.

A best estimate of the NDT shift in the supports is 50F, but much higher values were used in the analysis[1]. A more realistic assessment, including use of leak before break to reduce the postulated loads, and use of best estimate instead of bounding loads, would lead to no integrity concerns, even at the governing plants.

4.0 REVIEW OF SUPPORTS CONFIGURATIONS FOR WESTINGHOUSE PLANTS

A review was made of the support configurations for all Westinghouse plants, and a few discrepancies were found. These are listed in table 4-1. The dimensions of the various support configurations are listed in table 4-2, which refers to the geometry shown in Figures 4-1, 4-2 and 4-3.

The stresses for all support configurations were reviewed, and results showed that vertical tension loads existed on only three plants: Trojan and Turkey Point Units 3 and 4. All other plants have supports loaded entirely in compression during normal operation. Therefore, the key plants identified in reference [1] are correct. It should be emphasized that the loads available for all support configurations are upper bound faulted loads used for design type calculations, and these loads were not intended to be used in integrity calculations such as those in reference [1].

TABLE 4-1
REACTOR VESSEL SUPPORT
CONFIGURATIONS COMPARISON

	NUREG DATA	W DATA*
– San Onofre Unit 1	Short column?	Long column
– Point Beach Units 1&2	Short column	Long column/ Ring girder
– Seabrook Units 1&2	Ring girder	Short column on concrete
– S. Harris Units 1&2	Ring girder	Short column on concrete
– Prairie Island Units 1&2	Long column	Short column

—ALL OTHER PLANT DATA ARE IN AGREEMENT—

*Preliminary data, to be confirmed by each utility.

TABLE 4-2

DIMENSIONAL COMPARISON OF SUPPORT CONFIGURATIONS

DESIGN	VERTICAL(1) DIMENSION (INCHES)	HORIZONTAL(2) DIMENSION (INCHES)
TYPICAL SHORT COLUMNS(3) WITH BOX	+26	7 TO 9
TYPICAL BOX (NO COLUMNS)(4)	-4 TO 2	7
TROJAN (NUREG-CR-5320) (COLUMNS ON CANTILEVER BEAMS)	+94	26
TURKEY POINT (NUREG-CR-5320)	+28	26

(1) FROM TOP OF CORE DOWNWARD TO BOTTOM OF SUPPORT

(2) FROM VESSEL OUTSIDE DIAMETER RADIALLY OUTWARD TO SUPPORT

(3) SURVEY OF EIGHT PLANTS WITH COMMON CONFIGURATION

(4) SURVEY OF TEN PLANTS WITH COMMON CONFIGURATION

FIGURE 4-1

REACTOR VESSEL SUPPORTS/FUEL CONFIGURATION

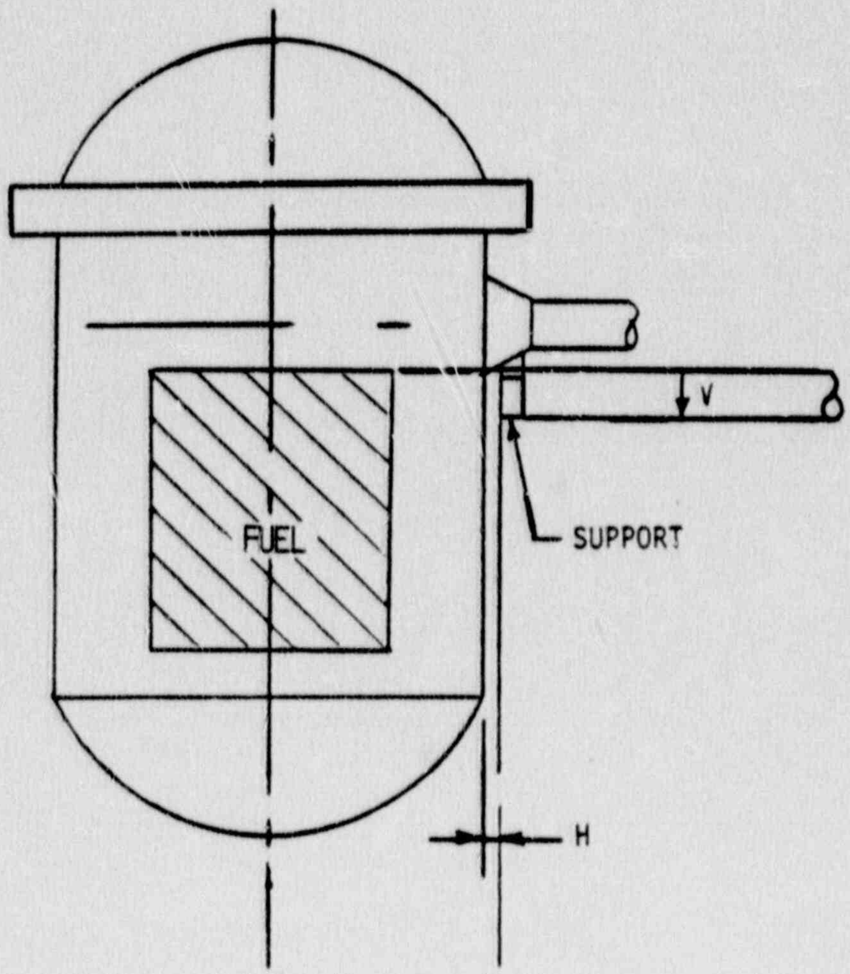
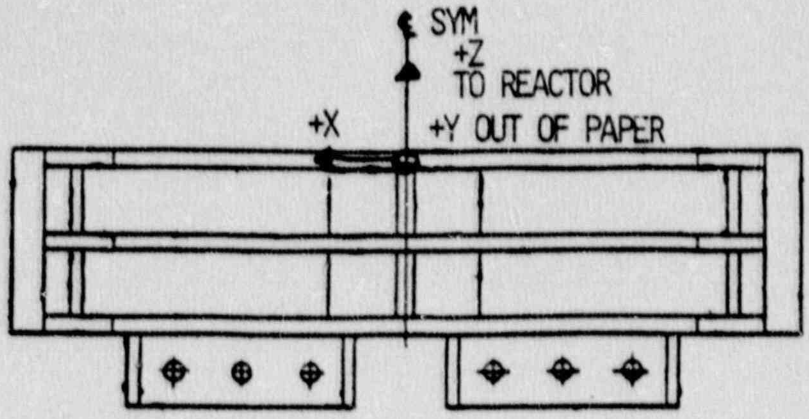
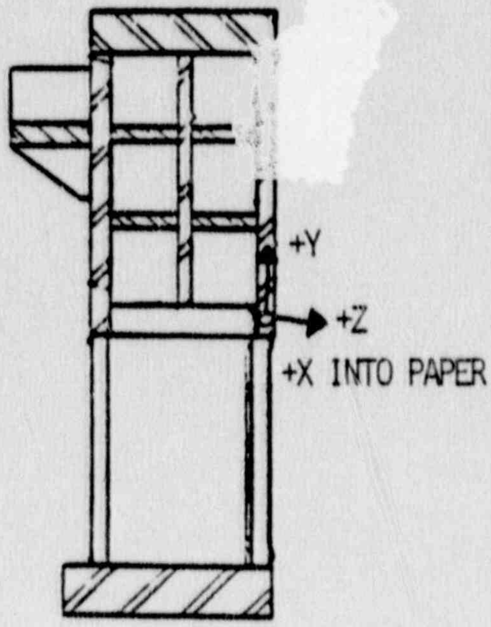


FIGURE 4-2



VIEW

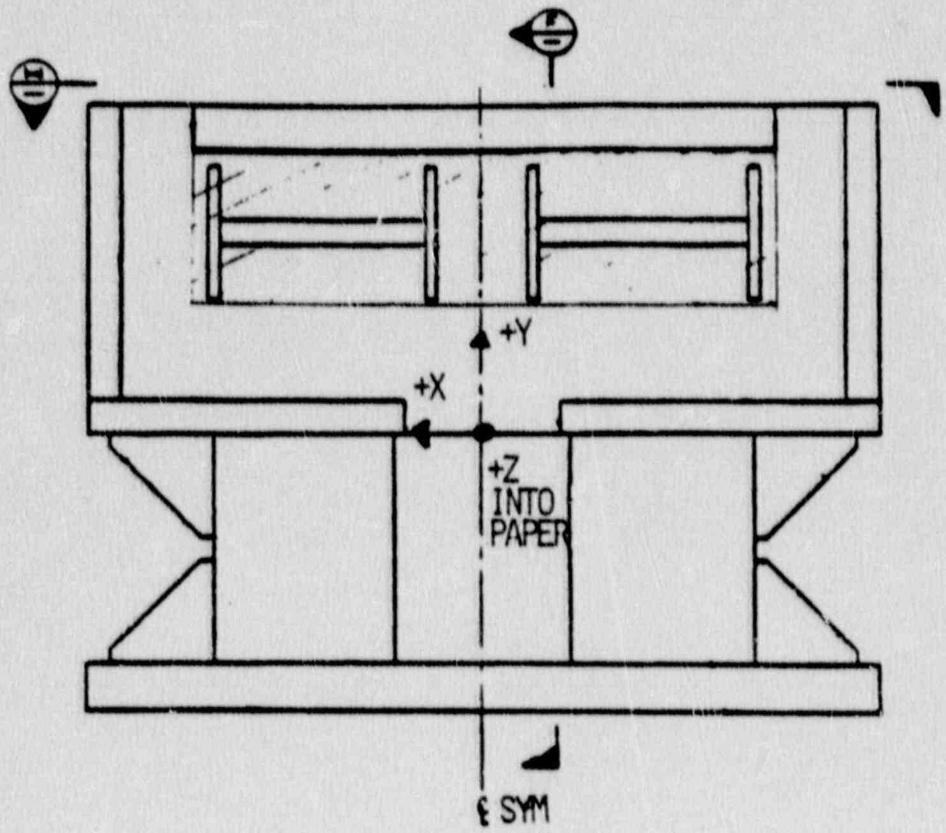
TOP PL NOT SHOWN FOR CLARITY



SECTION

AIR COOLED REACTOR SUPPORTS

FIGURE 4-3



AIR COOLED REACTOR SUPPORTS

5.0 SUMMARY AND CONCLUSIONS

In reviewing the ORNL report on integrity of reactor vessel support systems[1] a number of conservatisms have been identified. It now is evident that the HFIR irradiation spectrum should not be used to model the behavior of power reactor supports. A more realistic model would be Porter's model, which has been confirmed by the Shippingport results.

Other key conservatisms identified are the relatively large postulated flaws at the grout hole, and the use of worst case design loads instead of best estimate loads.

A realistic assessment of the reactor vessel supports leads to the conclusion that the issue here is a long range one, if it exists at all. The governing support configuration is a short beam, cantilevered from the vessel cavity wall. Only three plants have this arrangement, so this issue is not an Owners Group issue. Based on the available information the priority should be lowered to "low" for generic issue 15.

6.0 REFERENCES

1. Cheverton, R.D. et.al. "Impact of Radiation Embrittlement on Integrity of Pressure Vessel Supports for Two PWR Plants". Oak Ridge National Laboratories Report NUREG-CR-5320, ORNL/TM/10966. January 1989.

2. Porter.L.F., "Radiation Effects in Steel" published in ASTM STP 276, 1959, pp 147.

3. Chopra, O.K., Monthly Reports on "Aging Studies on Material from the Shippingport Reactor", March through May 1989, transmitted by letter to E. Woolridge, USNRC.

APPENDIX A
REVIEW OF REACTOR VESSEL SUPPORTS SURVEY

During the summer of 1989, a survey was carried out of all members of the Westinghouse Owners Group. The survey was designed to obtain as much information as possible from all Westinghouse plants regarding the configuration and operating conditions for the reactor vessel supports.

The survey was conducted in response to a concern which arose over the potential for damage to the supports by low flux, low fluence irradiation. A meeting was held on this subject between the Westinghouse Owners Group and the Nuclear Regulatory Commission on June 15, 1989, at which time an independent assessment of the issue by Westinghouse was presented. This presentation is documented in the main body of this report. A study had been completed earlier for the NRC by Oak Ridge National Laboratory [1], and contained in this reference is a summary of all support configurations for operating plants. One of the key goals of the survey was to verify the support configurations listed in that document.

The survey results are summarized in table A-1, which reveals that responses were obtained from 53 of 56 operating plants. The survey showed that several plants have support configurations which are different from those reported in [1]. Seven plants reported differences, and some provided detailed drawings, which have been included here as figures A-1 through A-3.

Excore dosimetry measurements are available on six plants to date, and others have measurements in progress. The values shown in table 2 have been calculated from actual measurements taken in the reactor cavity, over one fuel cycle. Results show fluence values ranging from 1 to 4 x 10^{17}n/cm^2 at the top of the core, and from 6 to 16 x 10^{17}n/cm^2 at the core mid-plane. Only one measurement is available (thus far) at the nozzle bottom, where most vessels are supported, and this measurement shows a very low value of 5.1 x 10^{16}n/cm^2.

Cavity temperatures were reported for 18 units and revealed a range of 80-150°F. This result is consistent with the maximum value of 150°F reported in [1]. Materials of construction were requested, and the responses revealed a variety of structural materials used. Some responses included bolting materials as well, but these were not included on the summary in table A-1. The survey revealed that very few utilities had records of preservice inspections performed on their supports. Very few unusual features were reported, and those reported did not appear to be of concern relative to the structural integrity of the supports, as shown in table A-1.

Virtually all of the vessels supports received some form of preservice inspection during the construction period. Most of the more recent plants have conducted volumetric examinations on the reactor vessel supports, and some of the early plants have done so as well. The variety of support configurations and range of construction dates make further generalizations difficult.

In summary, the survey has shown that in general the configurations reported in reference [1] are correct, and where they are incorrect, the corrected configurations have been identified. Reactor cavity temperatures were found to be bounded by the maximum value of 150°F reported in reference [1]. A wide range of materials were identified, but all carbon and low alloy steels fall in the same general category relative to susceptibility to low fluence irradiation, as shown by Porter [2].

The survey also provides a variety of other useful information, including cavity dosimetry results showing expected dosages for 32 effective full power years. The fluence in the nozzle support region was found to be extremely low, for the one plant which reported available data (5×10^{16} n/cm^2). Since the majority of plants have their key support configurations located in this area, there should be no cause for concern here. The results of the survey support the conclusions of [1] as to the governing plants for possible susceptibility, and these two geometries have already been analyzed in detail [1].

TABLE A-1: SUPPORTS SURVEY SUMMARY

PLANT	RESPONSE RECEIVED	NUREG CONFIRMED?	EX-CORE DOSIMETRY? (See Table 2)	CAVITY TEMP.	MATERIAL	UNUSUAL FEATURES?
A	X	Y(4C)		-	A588	None
B	X	Y(4C)		-	A588	None
C	X	Y(3)		100-150F	A516 Gr 60	None
D	X	Y(3)		100-150F	A516 Gr 70	None
E	X	Y(4E)		-	-	-
F	X	Y(4E)		-	-	-
G	X	Y(4E)		-	-	-
H	X	Y(4E)		-	-	-
I	None	-	-	-	-	-
J	X	Y(4G)		123F	A516 Gr 70, A36	-
K	X	Y(4G)		123F	A516 Gr 70, A36	-
L	X	Y(4C)		80F	-	-
M	X	Y(4C)		80F	-	-
N	X	N(3B) [1]		-	A36	-
O	X	N(4E) [6]		-	A588 Gr B	-
P	X	N(4E) [6]		-	A588 Gr B	-
Q	X	Y(4F)	Yes	-	-	-
R	X	Y(4F)	Yes	-	-	-
S	X	Y(4C) [2]		143F	T-1	None
T	X	Y(4C)	Yes	-	A441, A36	None
U	X	Y(4F)		-	-	None
V	None	-		-	-	-
W	X	Y(4C)		104F	A572, A516 Gr 70	None
X	X	Y(4C)		104F	A572, A516 Gr 70	None
Y	X	Y(4C)		90F	A588	None
Z	X	Y(4G)		123F	A516 Gr 70	8 - 4" diam. vent holes in 3" plate
AA	X	Y(4G)		123F	A516 Gr 70	

NOTE: Brackets refer to notes on last page of Table A-1.

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TABLE A-1: SUPPORTS SURVEY SUMMARY (cont.)

PLANT	RESPONSE RECEIVED	NUREG CONFIRMED?	EX-CORE DOSIMETRY? (See Table 2)	CAVITY TEMP.	MATERIAL	UNUSUAL FEATURES?
BB	X	Y(3)		-	A516 Gr 70, A537	
CC	X	Y(3)		-	A516 Gr 60	None
DD	X	Y(3)		-	A516 Gr 60	None
EE	X	N(2F)	Yes	80-90F	T-1, A53	None
FF	X	N(2F)	Yes	80-90F	T-1, A53	None
GG	X	Y(4c) [3]		133F	A588, Gr A	None
HH	X	Y(4c) [3]		133F	A588, Gr A	None
II	X	Y(4F)		110F	A441	Drilled and machined
JJ	X	Y(4F)		110F	A441	shear pin hole
KK	X	N(2B) [4]		111F	A302B, A36	None
LL	X	Y(4F) [5]		-	-	-
MM	X	Y(4C)		120-130F	A572	None
NN	X	Y(4C)		120-130F	A572	None
OO	X	N(4C)		-	-	None
PP	X	N(4C)		-	-	None
QQ	X	Y(4C)		-	-	-
RR	X	Y(4C)		-	-	-
SS	X	Y(3)		-	A516 Gr 60	None
TT	X	Y(3)		-	A516 Gr 60	None
UU	X	Y(4A)		-	-	-
VV	X	Y(4A)	Yes	-	-	-
WW	X	Y(4A)		-	-	-
XX	X	Y(4C)	-	-	A572, A302B, A588	3.5" diam. holes in embedded steel
YY	X	Y(4C)		120-130F	A572	None
ZZ	X	Y(4C)		120-130F	A572	None
ZA	None	-		-	-	-

NOTE: Brackets refer to notes on last page of Table A-1.

TABLE A-1: SUPPORTS SURVEY SUMMARY (cont.)

PLANT	RESPONSE RECEIVED	NUREG CONFIRMED?	EX-CORE DOSIMETRY? (See Table 2)	CAVITY TEMP.	MATERIAL	UNUSUAL FEATURES?
ZB	X	Y(4B)		122F	A201 Gr B	None
ZC	X	Y(4E)	Yes	-	-	-
ZD	X	Y(4E)	Yes	-	-	-

[1] 3B - four block supports on steel cylindrical shell
[2] 4C is the closest geometry, but the actual is slightly different - see figure A-1
[3] Slightly different, see figure A-2
[4] 2B - long column attached to vessel support brackets between nozzles, see figure A-3
[5] 4F* This girder is not continuous
[6] Weldment pedestal is embedded in concrete; support configuration looks like 4C, but no figure provided

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TABLE A-2
EXCORE DOSIMETRY

Plant	Core Midplane (n/cm^2)	Core Top (n/cm^2)	Nozzle Bottom (n/cm^2)
ZC	6.5×10^{17}	1.7×10^{17}	
ZD	5.8×10^{17}	4.3×10^{17}	
Q	8.1×10^{17}	2.4×10^{17}	
R	8.2×10^{17}	2.1×10^{17}	
W	1.4×10^{18}	3.8×10^{17}	
T	1.6×10^{18}	3.4×10^{17}	5.1×10^{16}

Notes: 1. Core top is near bottom of box beam supports.
2. These values are for 32 EFPY, estimated from measurements made over one fuel cycle.
3. All values are taken at energies greater than 1 Mev.

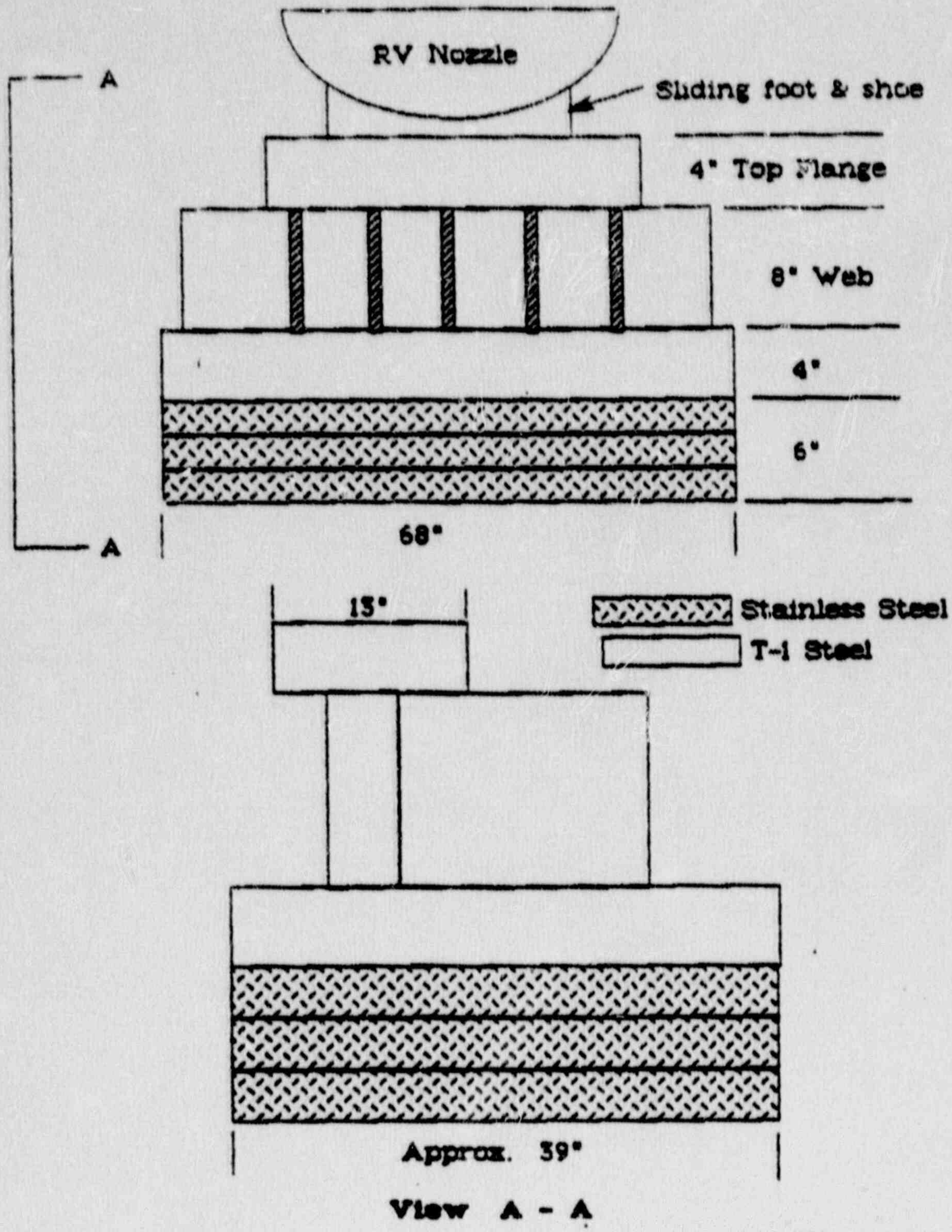

Figure A-1. Reactor Vessel Support Configuration for Plant S

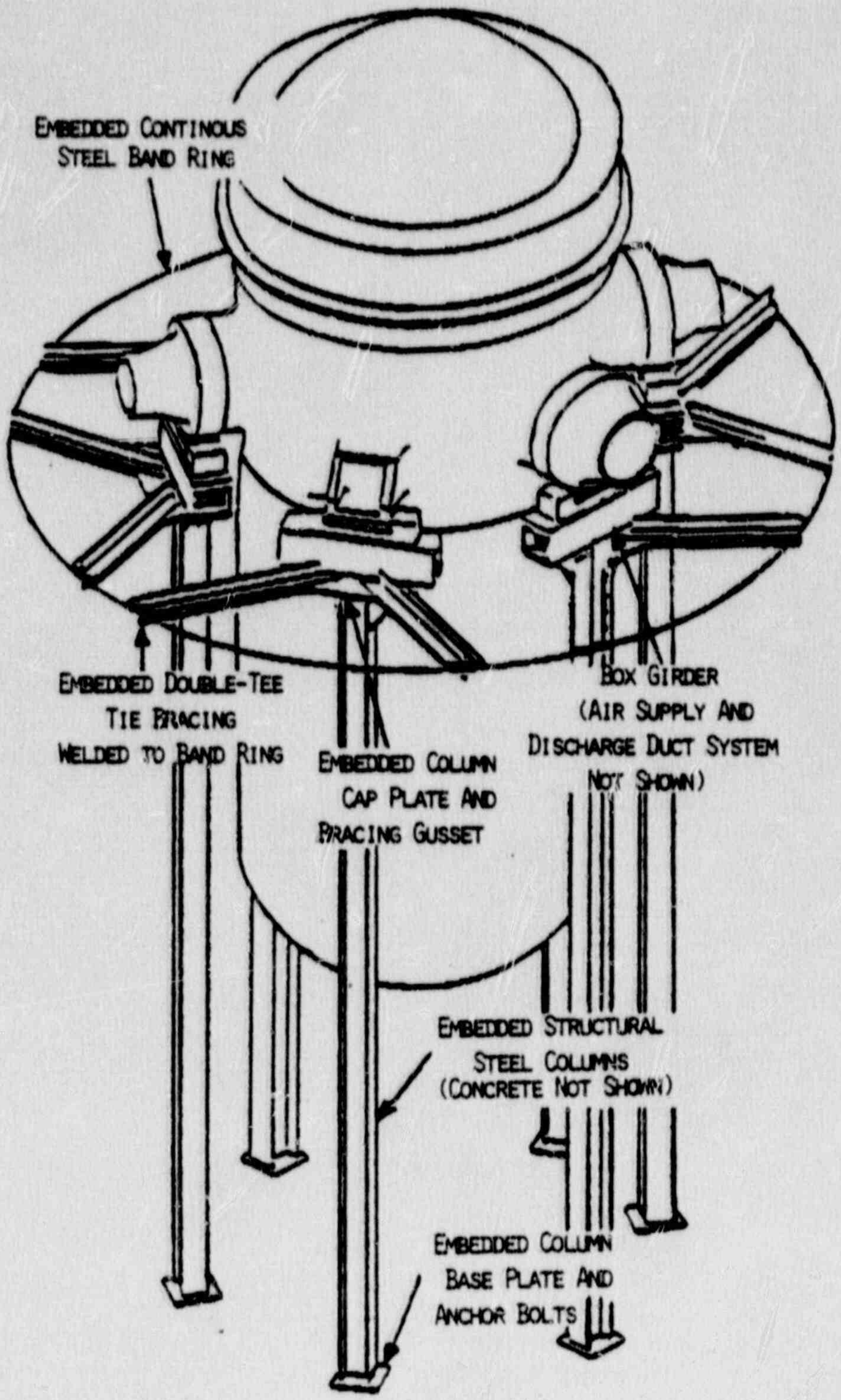

Figure A-2. Reactor Vessel Support Configuration - Plants GG and HH

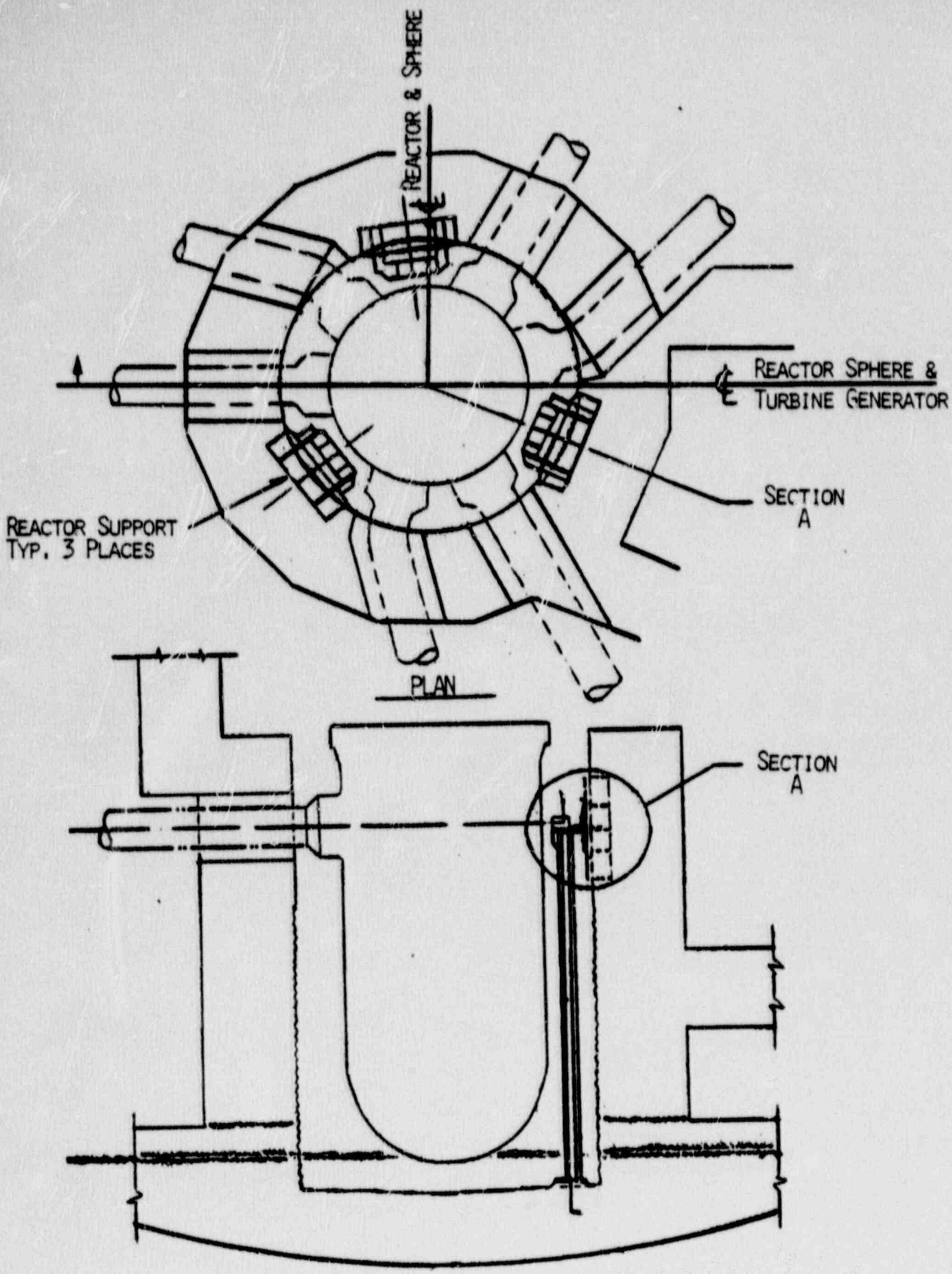

Figure A-3. Reactor Vessel Support Configuration - Plant KK