NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Point Beach, Unit 1	0 5 0 0 0 2 6 6	1 OF 1 3

TITLE (4)
Cracked & Missing Control Rod Guide Tube Split Pins

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 2	2 5	8 4	8 4	0 0 1	0 0	0 3	2 8	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[X] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER: AREA CODE	
C. W. Fay	4 1 4	2 7 7 - 2 8 1 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	A A	R C T	W 3 5 1	N					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 02/25/84, a visual inspection of the control rod guide tube split pins revealed that 3 nuts were missing. The visual inspection was performed after an ultrasonic inspection of the pins identified cracks in 67 of 74 pins (2 pins per guide tube). The pins are made of Inconel X-750 and because of the heat treatment temperature are subject to stress corrosion cracking. The pin design and material is generic to Westinghouse PWR's, and the cracking problem has been addressed in a Westinghouse bulletin and an NRC information notice.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach, Unit 1	0 5 0 0 0 2 6 6	8 4	— 0 0 1	— 0 0	0 2 OF 1 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

In Westinghouse letter WEP-79-524 dated 04/07/79, it was reported that a control rod guide tube support pin was found broken and a significant number of pins had UT indications of cracks at a Japanese PWR plant built by Mitsubishi Heavy Industries. No problem with support pins at Westinghouse plants had been reported at that time, and Westinghouse judged the failures not to be a safety issue. In Westinghouse letters WEP-82-536 and WEP 82-543 dated 07/01/82 and 07/22/82, it was reported that support pin cracking occurred at a domestic Westinghouse plant (North Anna 1) for the first time and the potential existed for stress corrosion cracking of the support pins at Point Beach Nuclear Plant, Unit 1, due to the heat treatment of the pins at less than 1800°F. In NRC IE Information Notice 82-29, "Control Rod Drive (CRD) Guide Tube Support Pin Failures at Westinghouse PWR's," dated 07/23/82, similar support pin failures were reported at Westinghouse-designed plants in Japan and France. Westinghouse's position was that a cracked or broken support pin did not constitute a safety concern; however, an inspection of the pins was recommended at the next convenient outage for Point Beach Nuclear Plant Unit 1.

The 37 control rod guide tubes (CRGT's) at PBNP, Unit 1 each contain two support pins (also called "split" pins) located at the bottom flange of the lower guide tube. The support pins horizontally align the bottom end of the guide tube at the upper core plate relative to the fuel assembly and provide lateral support for the guide tube against reactor coolant system flow and other forces (see Figures 1, 2, 3 & 4). The CRGT's provide lateral alignment and support for the rod cluster control assemblies (RCCA's) when the RCCA's are withdrawn from the core (see Figure 5). The support pin design allows removal of the CRGT from the upper internals and accommodates axial thermal expansion relative to the upper internal support columns (see Figure 6). The support pins each consist of a partially threaded shank, a collar, and a dual-leaf spring. The pins are bolted into the lower CRGT flange using a sleeve nut with a locking disk and pin. The leaf spring of each pin fits into a hole in the upper core support plate. The support pin assemblies are constructed of Inconel X-750 material, which has been heat treated and age hardened.

An ultrasonic inspection of the CRGT support pins at PBNP, Unit 1, was recently completed. The inspection revealed UT indications in the shank-to-collar region of 67 of 74 support pins (i.e., approximately 90%) (see Figure 7). No indications were found in the support pin leaves. A subsequent visual inspection of the sleeve nuts and threaded shank portion of the support pin assembly revealed 3 missing nuts and shanks (one each on the CRGT's above core positions G3, I5 & K7) (see Figure 8). Visual inspection of the peripheral portion of the upper core support plate (the inner portions are essentially inaccessible), the core former plate just below the upper core support plate, the lower core plate, the lower portion of the reactor vessel, and all upper nozzles of fuel

assemblies discharged during this outage (the full core) and the previous outage did not locate the missing portions of the pins. Subsequent inspections of the hot leg side of both steam generator channelheads resulted in the retrieval of one of the three nut and shank parts.

Westinghouse has performed an evaluation of the ability of both Point Beach Nuclear Plant units to start up and continue safe operation with broken CRGT support pins. Two potential safe concerns must be addressed to justify safe startup and continued operation with the three missing CRGT support pin nuts and shanks and the potential for additional support pin breakage during operation. These potential concerns are misalignment of the CRGT which could prevent RCCA insertion and adverse effects on safety-related components resulting from loose support pin parts in the reactor coolant and connected auxiliary systems.

The first potential safety concern is that broken CRGT support pins could prevent RCCA insertion due to lateral misalignment of the bottom of the CRGT with respect to the upper core plate. Westinghouse analysis shows that, if both support pins in the same guide tube are broken at the shank-to-collar interface and at both leaves (where previous experience at the other plants has shown cracking) "no safety concern with respect to control rod guide tube misalignment is identified relative to safe shutdown of the reactor with control rod insertion." Westinghouse analysis shows that, even if all support pin assemblies are completely missing, the CRGT alignment will be maintained sufficiently by the adjacent support columns, flow mixers, and orifice plates on the upper core plate such that "the required Technical Specification rod drop time of 1.8 seconds will be met." In addition, a review of RCCA rod wear and rod scram time trends over 11 fuel cycles have not indicated any CRGT misalignment problem. The biweekly rod stepping tests performed during operation per Technical Specifications should indicate any significant CRGT misalignment during the next fuel cycle. Therefore, the potential for the misalignment is judged not to be a safety concern.

The second potential safety concern is that loose parts from broken support pins may cause adverse effects on safety-related components in or connected to the reactor coolant system. Westinghouse has analyzed a loose support pin part affecting RCCA insertion and stated that "it is considered an extremely remote possibility that any loose part could affect all the movements and placements necessary to affect RCCA movement." In addition, the reactor can be safely shut down with the highest worth control rod stuck in the fully withdrawn position. If loose support pins parts enter the steam generator channelheads, there is an extremely remote possibility that a support pin shank might separate from the nut, enter a Row 1 tube (approximately 3% of the tubes) get captured in the U-bend, and

vibrate over time causing fretting and eventual small localized tube leakage. The plant is designed for a single double-ended tube rupture so the above scenario is judged not to be a safety concern. Adverse effects of loose parts on the reactor coolant pumps are not expected, since the loose parts should pass through the pump. The remote possibility of a locked rotor accident resulting from a loose part has already been analyzed in the PBNP FSAR. The adverse effects of loose parts on the reactor internals would only be a concern if large numbers of broken support pin shanks separate from the nut, travel to the lower part of the reactor vessel and land under the secondary core support energy absorbers during cold conditions. The resulting wedging effects on the lower internals during heatup were found by Westinghouse to be acceptable using conservative criteria unless an incredible number of shanks were wedged there simultaneously (6 under each support or 10 under one support).

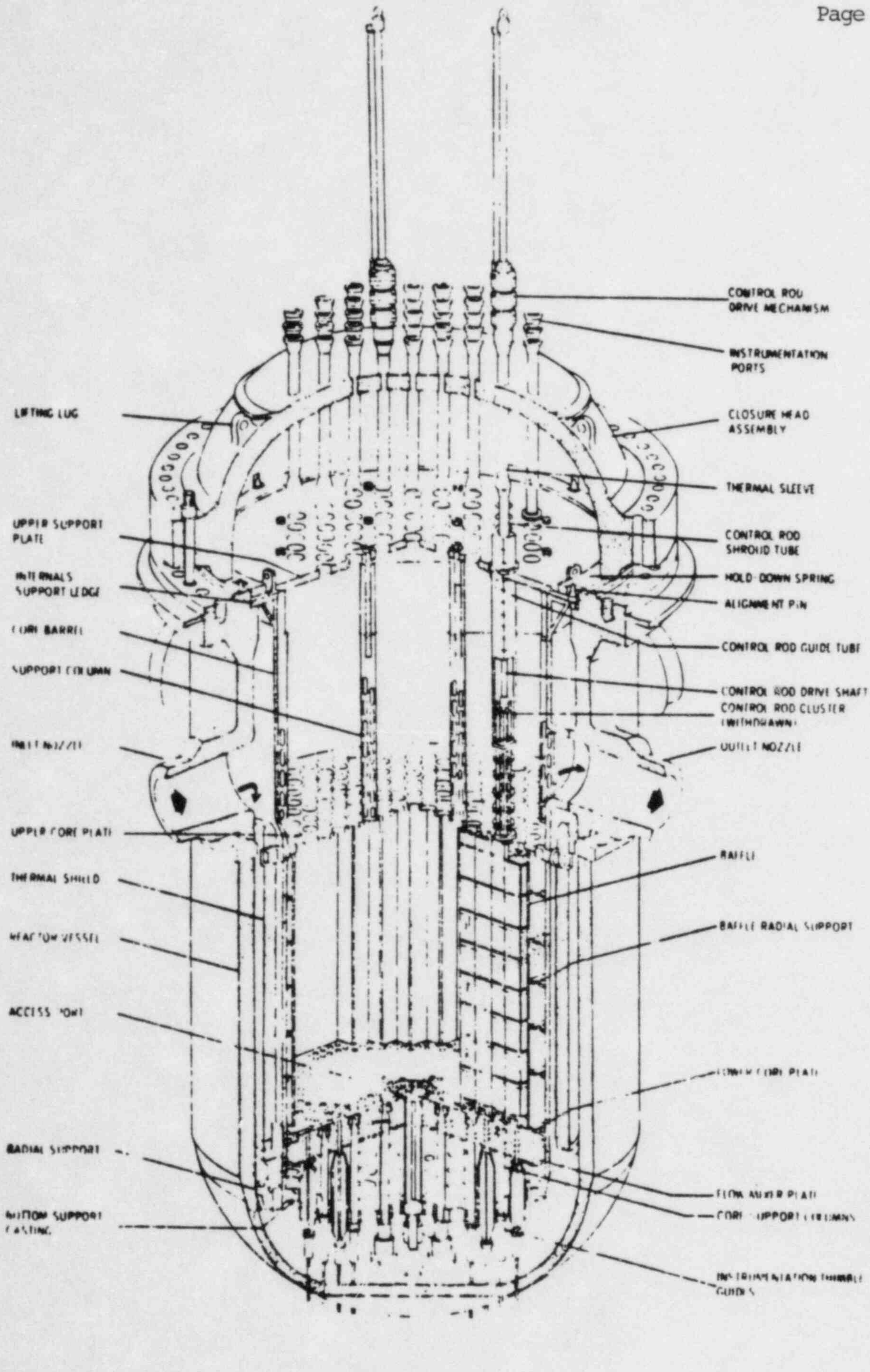
The effects of loose parts on auxiliary systems or components was also reviewed and the RHR system was the only connected system which could be significantly affected. The RHR system valves could not be held shut by loose parts so RHR system startup could be initiated by opening valves. RHR isolation valves held open by loose parts would be detected upon heatup. The RHR pumps are expected to pass loose parts through without seizure but the parts could cause some damage to the impeller. Loose parts in a heat exchanger could result in a tube leak but this would be detected by leakage into the component cooling water system. Since redundant RHR pumps and heat exchangers are provided, the components are accessible for repair if degraded system operation is detected, and safety injection pumps or the steam generators are available for alternate methods of core decay heat removal, the potential effects of loose parts in the RHR system are judged not to be a safety concern. Therefore, the potential effects of loose CRGT support pin parts is judged not to be a concern for any safety function.

In conclusion, the UT indications on the shank-to-collar region of the CRGT support pins and the 3 missing support pin nuts and shanks at PBNP, Unit 1, are judged not to be a safety concern due either to CRGT misalignment preventing RCCA insertion or the adverse effects of loose parts on safety systems and components. This problem does not represent a violation or require a change to the PBNP Technical Specifications or involve an unreviewed safety question. The normal startup RCCA tests (rod drops and stepping tests) and the biweekly RCCA stepping tests required by the Technical Specifications will be monitored to verify free RCCA movement. Also, RHR system isolation will be verified during heatup to ensure that loose parts are not captured in the isolation valves. A loose parts monitoring system is also being installed this outage, and will be operational when the plant is returned to service to help identify any loose parts in the system.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach, Unit 1	0 5 0 0 0 2 6 6	8 4	— 0 0 1	— 0 0	0 5 OF 1 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

The long-range solution to this problem is to replace the pins at a future outage when the tooling and parts are available. An inspection of the PBNP Unit 2 pins is also planned during the next refueling outage. Cracking of the split pins in Unit 2 is not considered as likely, however, since they were heat treated at a temperature much closer to the recommended ideal.



REACTOR VESSEL INTERNALS

FIGURE 1

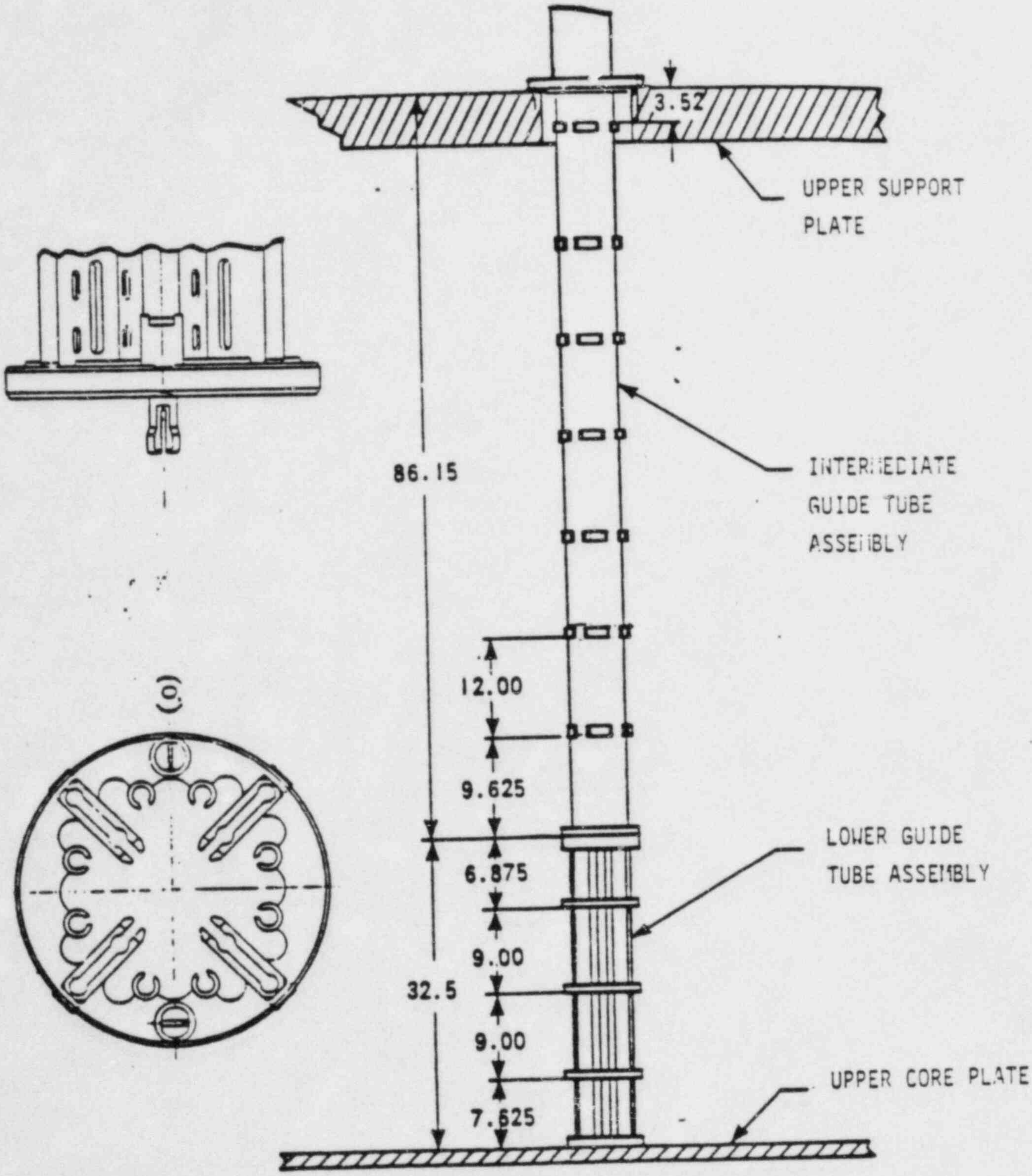

3.52
UPPER SUPPORT PLATE
86.15
INTERMEDIATE GUIDE TUBE ASSEMBLY
12.00
9.625
6.875
9.00
32.5
9.00
7.625
LOWER GUIDE TUBE ASSEMBLY
UPPER CORE PLATE
(0)

FIGURE 2

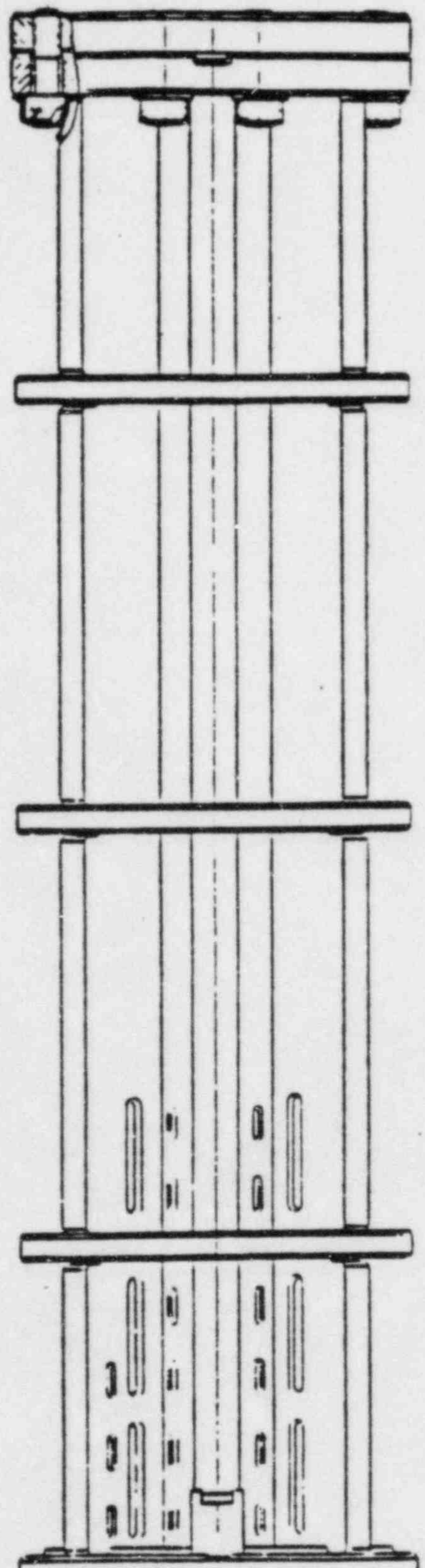
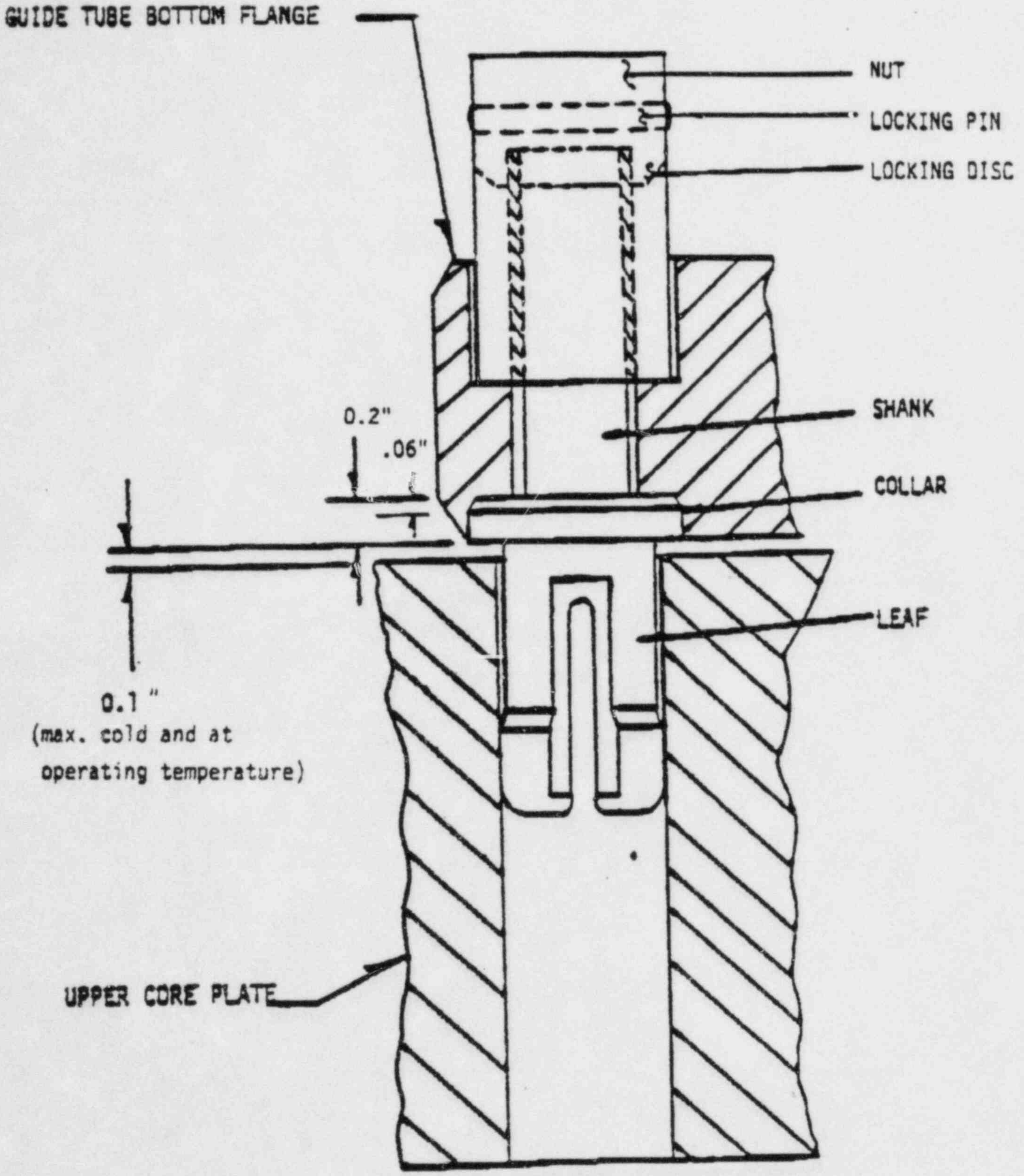
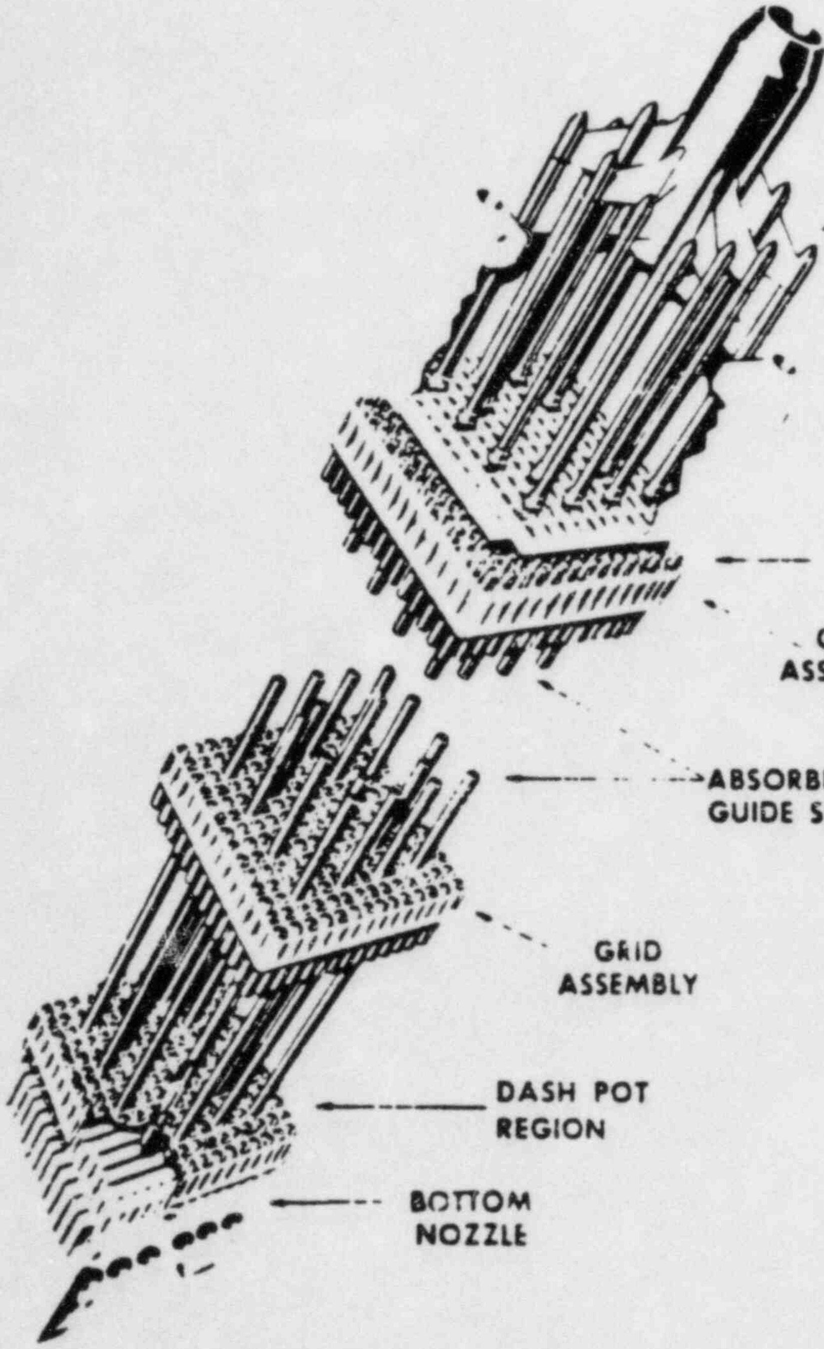


FIGURE 3

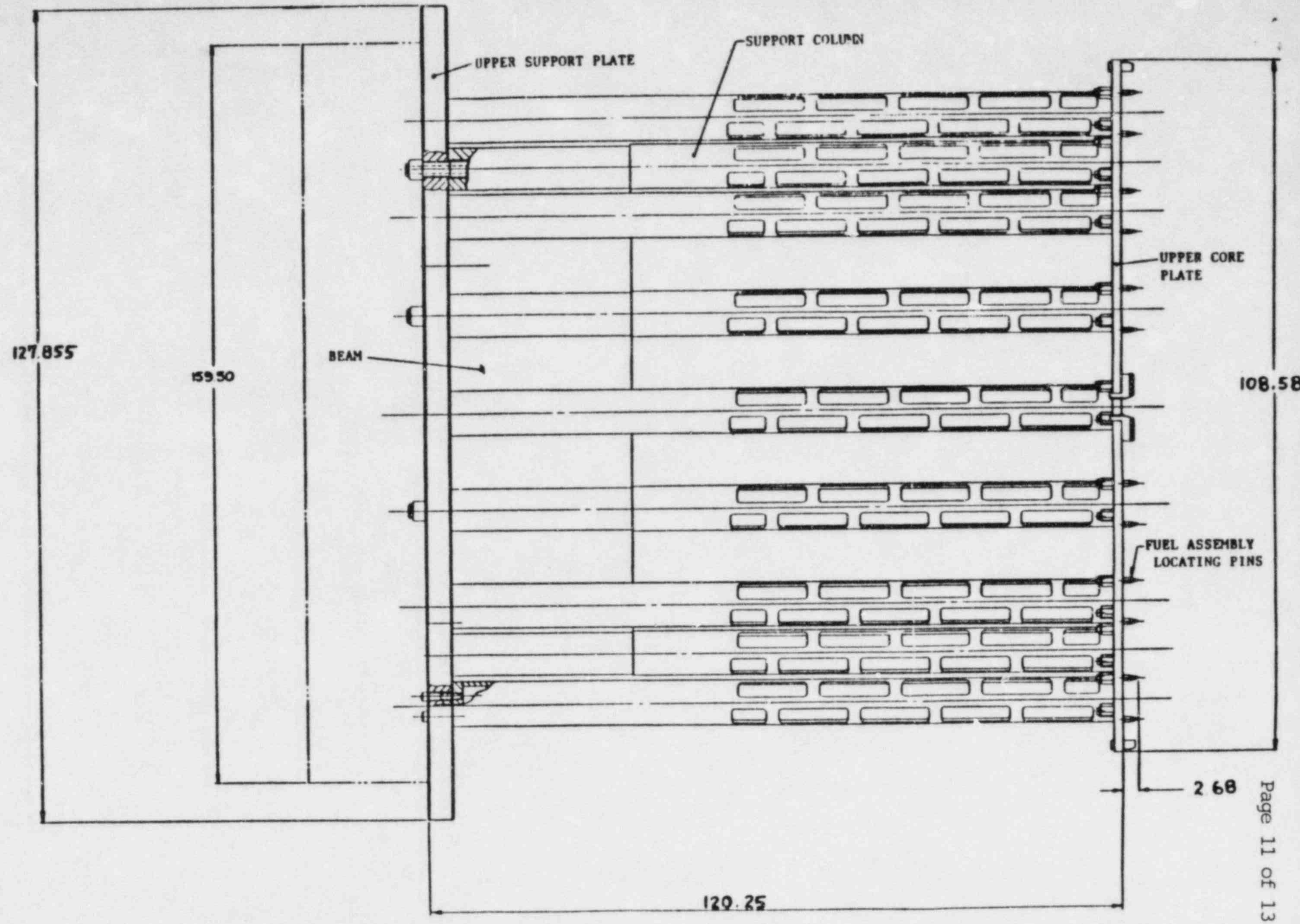


GUIDE TUBE SUPPORT PIN

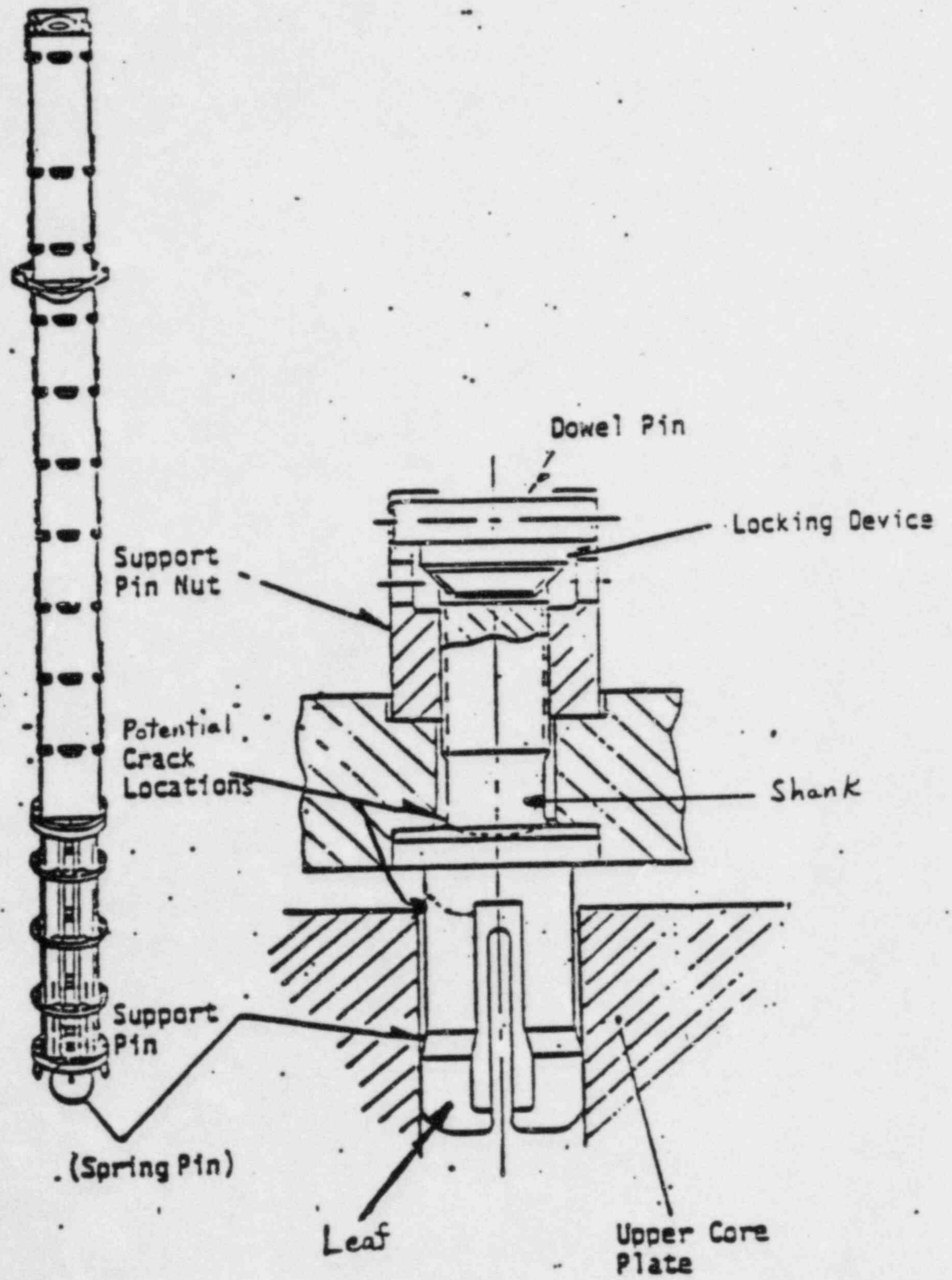
FIGURE 4



TYPICAL ROD CLUSTER CONTROL ASSEMBLY
FIGURE 5

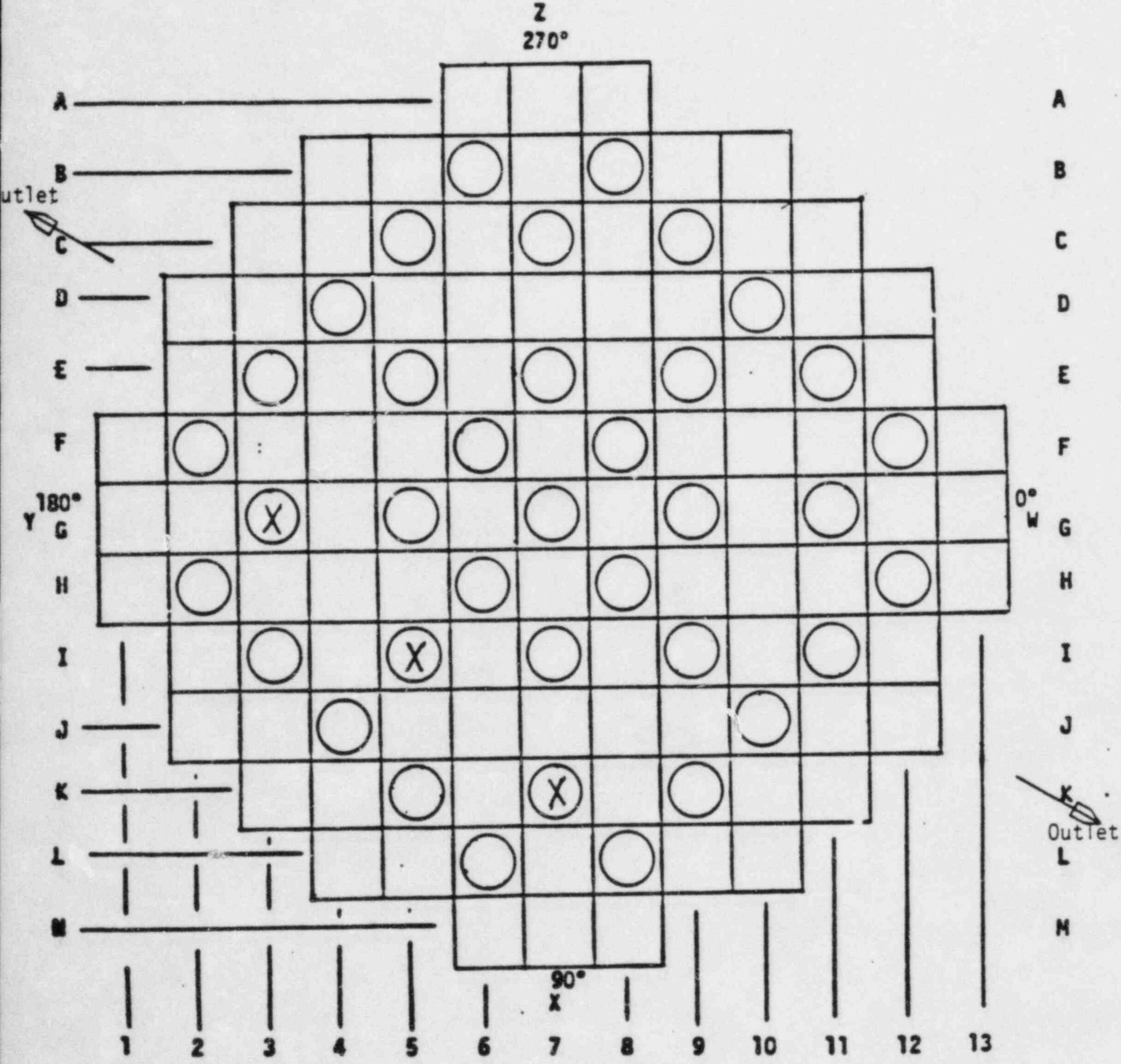


UPPER CORE SUPPORT ASSEMBLY
FIGURE 6



GUIDE TUBE ASSEMBLY

FIGURE 7



○ Control Rod Guide Tube Locations

X CRGTs with a Missing Split Pin Nut

FIGURE 8

DmB

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

March 28, 1984

Mr. J. G. Keppler, Regional Administrator
Office of Inspection and Enforcement,
Region III
U. S. NUCLEAR REGULATORY COMMISSION
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

DOCKET NO. 50-266
LICENSEE EVENT REPORT NO. 84-001-00
CONTROL ROD GUIDE TUBE SPLIT PINS
POINT BEACH NUCLEAR PLANT, UNIT 1

Enclosed is Licensee Event Report No. 84-001-00 which provides a description of an event which we are reporting because of its potential generic significance. This report does not fall under any specific 10 CFR reporting requirement but is being reported under the "other" category.

Very truly yours,

CW Fay

Vice President-Nuclear Power

C. W. Fay

Enclosure

Copy to NRC Resident Inspector

APR 5 1984

IE22
1/1