Westinghouse Safety Evaluation

Cracked CRGT Support Pins

Telecopied 3/5/84

CRACKED SUPPORT PIN AND LOOSE PARTS

EVALUATION FOR POINT BEACH UNITS 1 AND 2

During a recent outage at Point Beach Unit 1, an ultrasonic inspection of the guide tube support pins was conducted. The inspection revealed indications in 67 out of 74 support pins. The indications were located in the shank-to-collar region of the support pin (Figure 1). No indications were found in the support pin leaves. A subsequent visual inspection revealed three missing support pin nuts and threaded shank portions of the support pin assembly. After performing a visual search on a portion of the upper core plate the missing objects were not located. The utility then consulted with Westinghouse and a conclusion to discontinue further search efforts in the upper internals area was reached.

Although the inspections revealed multiple indications and three missing nuts on the support pins only at Unit 1, this evaluation addresses potential conditions at both Point Beach Units 1 and 2. Therefore, the results of this evaluation are applicable to Unit 2 also, should future inspection reveal similar conditions in support pins at that unit.

This safety evaluation will assess the ability of Point Beach Unit 1 to start up and continue safe plant operation with broken guide tube support pins and with the aforementioned loose parts or additional support pin parts present in the reactor coolant system (RCS) for the next operating cycle.

I. MISALIGNMENT BETWEEN GUIDE TUBE AND UPPER CORE PLATE

Assuming the two guide tube support pins on the same guide tube are broken in the collar and leaf area, the remaining pin section consisting of the pin collar will remain trapped between the guide tube flange and the top of the upper core plate. With a failed shank it is possible for the remaining pin section to drop into the 0.1 inch gap between the bottom of the guide tube flange and the top of the upper core plate (maximum dimension cold and at operating temperature). However, since the width of the collar on the support pin is 0.2 inches, approximately 0.1 inch of the collar remains in the guide tube flange (Figure 1). The remaining pin section in the guide tube flange has been shown by analysis to provide the necessary alignment function between guide tube and core plate to allow for proper control rod operation. Since the alignment of the guide tube is maintained even if both pins on the same guide tube crack in the shank and leaves, no safety concern with respect to control rod guide tube misalignment is identified relative to safe shutdown of the reactor with control rod insertion.

Layout studies of the upper core plate were performed to determine the magnitude of guide tube flange displacement if no guide tube support pins were present which would be permitted by adjacent support columns, flow mixers, and orifice plates.

Results of the layout studies indicated that the maximum displacement in the horizontal direction which would be limited by an adjacent orifice plate, assuming that all flanges are at their minimum allowable dimensions, is 0.1917 inches. The maximum displacement in the horizontal direction limited by an

adjacent flow mixer or support column, assuming that all flanges are at their minimum allowable dimensions, is 0.1849 inches. An analysis has shown that a guide tube displacement of 0.1917 inches shows that the required Technical Specification rod drop time of 1.8 seconds will be met.

Since it is not known precisely when the support pin cracking initiated, a review of the control rod scram times recorded over the past 11 fuel cycles was conducted and no increasing time trend was observed. In addition, a review of the control rod wear on 23 rod cluster control assemblies was performed. Wear scars indicative of severe guide tube misalignment were not observed. This is evidence that the cracked guide tube support pins have not degraded rod drop times due to control rod guide tube wear and/or misalignment. A condition has not been created where the Technical Specification required rod drop time of 1.8 seconds has been violated. Significant degradation of rod drop time is not expected to occur during the next operating cycle.

II. LOOSE PARTS EVALUATION

This section addresses the potential effects of loose parts on the reactor coolant system and the residual heat removal system (RHRS). The RHRS is addressed since it is open to the RCS during cooldown and therefore creates a path for the loose part to enter the RHRS. Systems which are not safety related were not evaluated, nor were systems connected to the RCS in the upper half of the piping, since there is a low probability of the loose parts entering these systems. See Table 1 for a description of the potential loose parts from support pins.

o Rod Cluster Control Assembly (RCCA) Operation

The potential for a loose part resulting from a broken guide tube support pin to adversely affect RCCA operation is extremely remote. For RCCA operation to be affected a loose part would have to enter a guide tube through one of the flow openings in the lower guide tube area. Once inside the guide tube the loose part would have to move between the rod channels into the center of the guide tube and rise approximately 30 inches above the continuous guide tube region. Refer to Figures 2, 3 and 4. Once above the continuous region the loose part would then have to orient itself in such a way as to lay across a guide plate and block the spider channel. It is considered an extremely remote possibility that any loose part could effect all the movements and placements necessary to affect RCCA movement.

Even if this were to occur, such a circumstance does not constitute an unresolved safety question since analyses have shown that the reactor can be safely shutdown with the highest worth control rod stuck in the fully withdrawn position.

o Reactor Coolant Pumps

It is expected that the reactor coolant pumps will not be affected by loose parts from a guide tube support pin. The loose parts should pass through the pump with no change in pump vibration characteristics or increase in locked rotor accident probability. Significant mechanical damage to the pump impeller is considered to be unlikely.

o Steam Generator

If loose parts from a support pin were to enter the steam generator channel head at Unit 1 the only effects would be peening-type impacting of the tubesheet cladding and channel head. Tube ends in the Model 44F steam generators at Unit 1 are flush with the tubesheet face and would therefore not be exposed to impact damage. The Model 44 steam generators at Unit 2 do have protruding tube ends and could therefore be impacted by loose parts in the channel head. Additionally, the steam generator tube sleeves installed in Unit 2 extend slightly beyond the tube ends. This would then expose the sleeve ends to impacting from loose parts and exert forces on the sleeve. Sleeve joint integrity is not expected to be adversely affected if impacting occurs due to the resistance provided by the lower joint hard roll. Tube and sleeve end deformation may occur but the impact forces are not expected to be large enough to create sleeve movement relative to the tube. Loose parts impacting on un-sleeved tube ends may also result in tube end deformation but there are no safety concerns. Tube ends may require rework before being able to insert an eddy current probe at the next inspection but the integrity of the tube-to-tubesheet weld and the tube hard roll would minimize the potential for a primary-to-secondary leak to occur. Previous experience at another Westinghouse plant has shown that severe tube end deformation has only a small effect on available coolant flow capacity. Tube end fragments which may be generated from impacting are not expected to create a safety issue due to prevention of control rod insertion. If such fragments are generated, the tortuous path

they must follow and the unique position in which they must come to rest in order to prevent rod movement is judged to be so unlikely as to not create a safety concern.

The nut and locking disc diameters are too large to enter into the steam generator tubes. However, the shank, leaves and locking pin could conceivably enter the steam generator tubes. The length of the shanks and leaf fragments from split pins at Point Beach are 1.4 inches and 1.5 inches, respectively. An analysis has shown that an object up to 1.9 inches in length can pass through all U-bends down to and including row 2. These pieces may, however, lodge in the row 1 U-bends. It is considered unlikely that one of the support pin loose parts will enter a row 1 tube since row 1 tubes make up only 2.8% (92/3260) of the tube bundle (some Row 1 tubes may be plugged). Leaf and shank orientation to enter a steam generator tube would have to be perpendicular to the tubesheet whereas the most likely orientation, with the largest surface exposed to flow, would be parallel to the tubesheet. This also serves to minimize the potential that one of these loose parts will enter a U-tube. Even if a loose part does become lodged in a row 1 U-bend, it may wear through wall and create a detectable leak which would initiate plant shutdown for investigation, but it is unlikely that a tube burst would result. This is due to the inherent ductility of the tube material and the expected localized wear region which would be surrounded by tubing with virtually full-depth wall. Assuming that the loose parts enter a row one U-bend and become lodged, they would likely be lodged in such a manner as to be virtually stationary, experiencing minimal vibration. This would

serve to limit the rate and extent of tube wear. Even if a primary-to-secondary leak larger than allowable Technical Specification does develop, analysis performed for the Point Beach sleeving program shows that a 47.5 gallon per minute primary-to-secondary leak per steam generator concurrent with a steamline break would still not violate 10CFR100 site boundary radiation dosage criteria. The plant is designed to respond to a single double-ended tube rupture by achieving a safe shutdown.

Impacting of these loose pieces on the inside of steam generator tubes as the pieces flow through the tubes is judged not to significantly affect the integrity of the tubing. Tube wall thinning is only an issue where the loose parts could become lodged, as in the row 1 U-bends, and create constant wear at a localized region. In larger radius U-tubes, and in the straight leg portion of the row 1 U-tubes, the motion of the loose parts through the tubes would not significantly affect tube wall thickness.

The effect of the locking pin on the steam generator is judged to be insignificant due to its small mass, which would not create an impact damage concern and small size, which would allow it to pass through a steam generator tube without becoming lodged.

o RHRS Components

Valves - Loose parts from a guide tube support pin will not hold valves shut. The possibility does exist however that an RHRS gate valve could be partially held open. Since the RHRS operates by opening valves, normal system operation should be achievable. Verification of closure of the RHRS gate valves should be accomplished via leakage testing during heatup.

Heat - Exchanger Loose parts from a guide tube support pin could enter a residual heat removal heat exchanger. If this were to cause a tube leak, it would be identified via high radiation and high surge tank level in the component cooling water system and the affected heat exchanger could be valved out for repairs.

Pumps - The RHR pumps are expected to pass loose parts from guide tube support pins with no pump seizure. The possibility exists that some part of a pump impeller could be cracked or chipped by an impact from a loose part. While it is not believed that this would impair pump operation, it would likely cause mechanical imbalance and shaft vibration which would be observed during pump operation. Two pumps are provided for decay heat removal.

o Reactor Internals and Vessel

Loose parts from guide tube support pins are not expected to affect either upper or lower reactor internals. The mass of the individual loose parts is not considered sufficient to impart any significant impact loads on reactor internals. Fretting and wear from loose parts on internals components are also expected to be insignificant.

In order to determine which types of support pin loose parts may reach the bottom of the reactor pressure vessel and become wedged between it and the reactor internals secondary core support plate, the path that the parts must follow was investigated.

There are two possible paths for guide tube support pin loose parts to reach the bottom of the vessel. The first and most likely path is from the outlet plenum, through the outlet nozzle, through the primary coolant loop into the reactor pressure vessel inlet nozzle and down the downcomer into the lower plenum. For this path the steam generator is the limiting restriction. Neither a support pin locking disc, support pin shank and nut assembly nor a support pin nut alone is able to pass through the steam generator. However, in the unlikely event that a support pin shank and nut assembly separate, the support pin shank could pass through the steam generator and into the vessel lower plenum.

The second unlikely way that guide tube support pin loose parts may reach the bottom of the vessel is that loose parts lying on the top

surface of the upper core plate following plant shutdown fall through an upper core plate flow hole into the core region during removal of the upper internals and into the lower plenum during removal of the core. Although both support pin shanks and nuts are able to follow this path, the 1.130 inch diameter nut is too large to enter the 1.00 inch gap between the bottom of the vessel and the four secondary core supports. However, in the unlikely event that the nut and shank separate, the .625 inch diameter threaded shank could become wedged at this location. Because of the support pin leaves, locking pin and locking disc dimensions, they do not create a concern for wedging effects in the lower vessel region.

Therefore, an analysis was performed to determine the maximum number of shanks which may become wedged whereby the structural integrity of the reactor pressure vessel and internals is not affected. In order to accurately predict loads induced upon the reactor pressure vessel and internals by wedged support pin shanks, test data for load versus interference for the actual wedging situation was obtained. Using this data it was determined that for the extremely unlikely case of all shanks becoming wedged beneath a single energy absorber that ten wedged shanks are acceptable. For the case of the wedged shanks being equally distributed beneath two energy absorbers twenty wedged shanks are acceptable. For the cases of the wedged shanks being equally distributed beneath three or four energy absorbers, at least twenty-four wedged shanks are acceptable.

It should be noted that the criteria used to determine acceptance in this evaluation were either (a) stresses above the yield point in the secondary core support or (b) a reduction of the core hold down load to less than 100,000 lbf at full flow conditions. While violation of these criteria would not lead to safety concerns, they were chosen to determine that no damage could occur to the reactor internals components.

Because of the low probability of the various assumed conditions occurring during the next fuel cycle and because of the conservatism in the criteria, wedging of broken support pin parts is not a safety concern for the Point Beach plants.

Conclusion

As presented in the above evaluation the ability of Point Beach Unit 1 to startup and continue safe plant operation for the next fuel cycle with failed guide tube support pins does not constitute an unresolved safety question. The safety of Point Beach Unit 1 is not affected due to operation with failed guide tube support pins or the loose parts that might be generated. Operation beyond the next fuel cycle would require verification that the bases of this evaluation are still met. Should similar support pin conditions be discovered at Unit 2, the conclusion of this evaluation will also be applicable. The bases for this evaluation are: 1) the component wear and misalignment analysis contained in this evaluation, 2) the loose parts analysis contained in this evaluation, 3) review of rod drop time over the previous operating cycles to assure no degradation of the required drop time and, 4) detailed

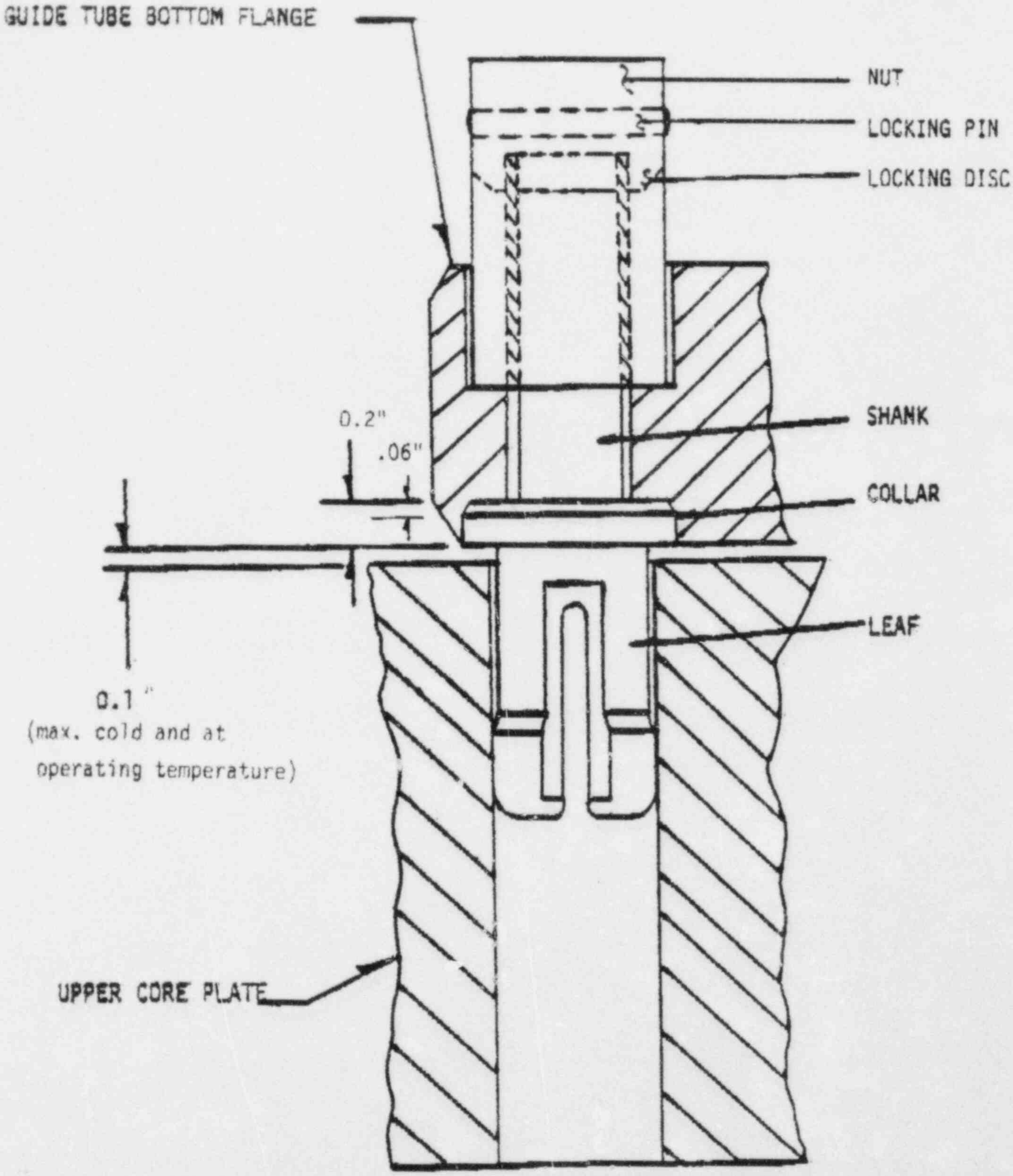
attention given to the Technical Specification requirement for bi-weekly rod stepping tests to assure rod movement and evaluation of any abnormalities noticed during these tests.

TABLE 1

POTENTIAL LOOSE PARTS GENERATED

FROM GUIDE TUBE SUPPORT PINS

Part	Dimensions	Weight
Shank	1.4" long 0.625" diameter	0.13 lbm
Leaves (2)	1.5" long 0.750" diameter	0.2 lbm (0.1 lbm/leaf)
Nut	1.3" long 1.13" diameter	0.2 lbm
Locking Disc	0.812" diameter	0.03 lbm
Locking Pin	1.125" long 0.25" diameter	0.016 lbm



GUIDE TUBE SUPPORT PIN

FIGURE 1

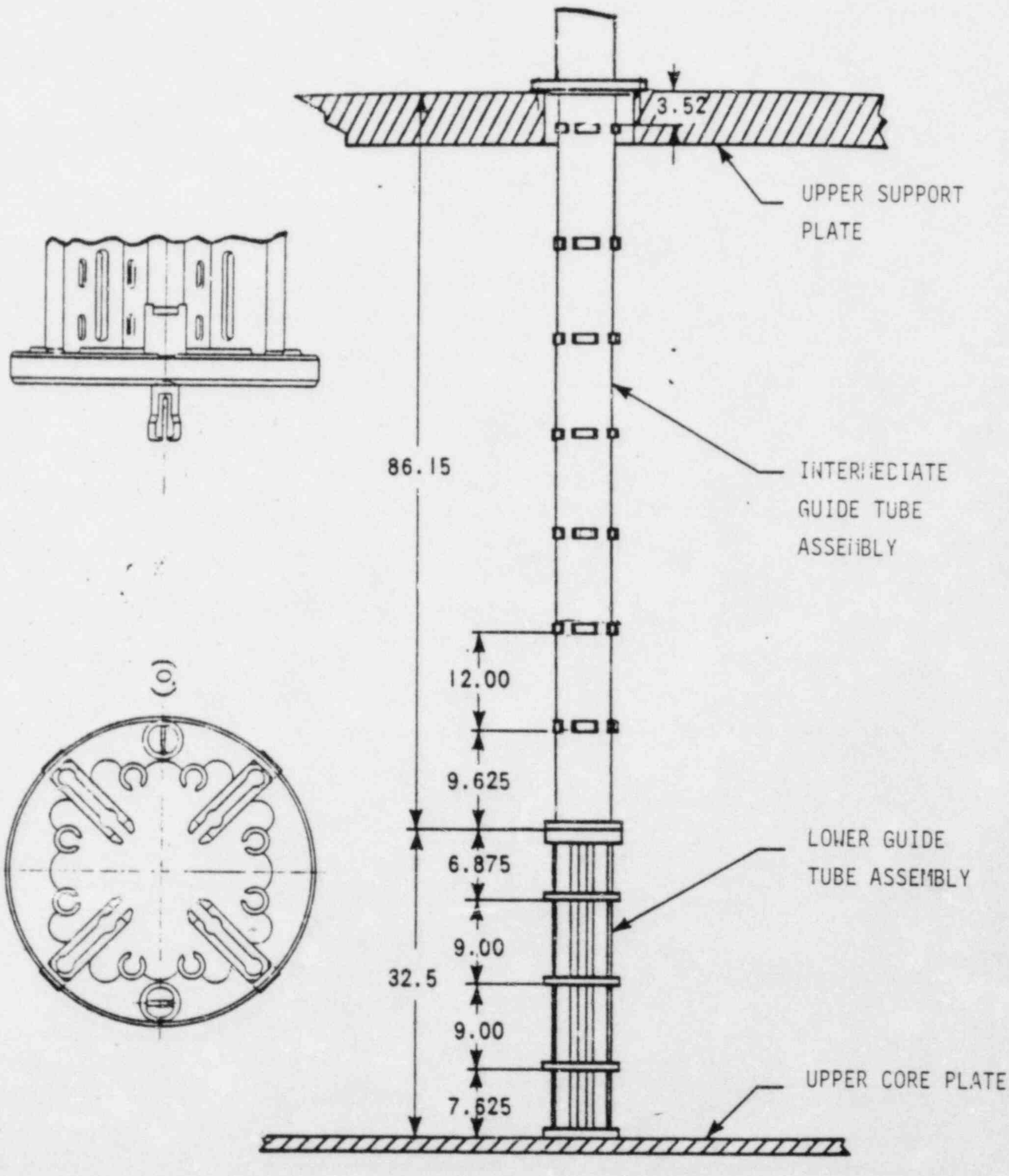

3.52
UPPER SUPPORT PLATE
86.15
INTERMEDIATE GUIDE TUBE ASSEMBLY
12.00
9.625
6.875
LOWER GUIDE TUBE ASSEMBLY
9.00
32.5
9.00
7.625
UPPER CORE PLATE

FIGURE 2

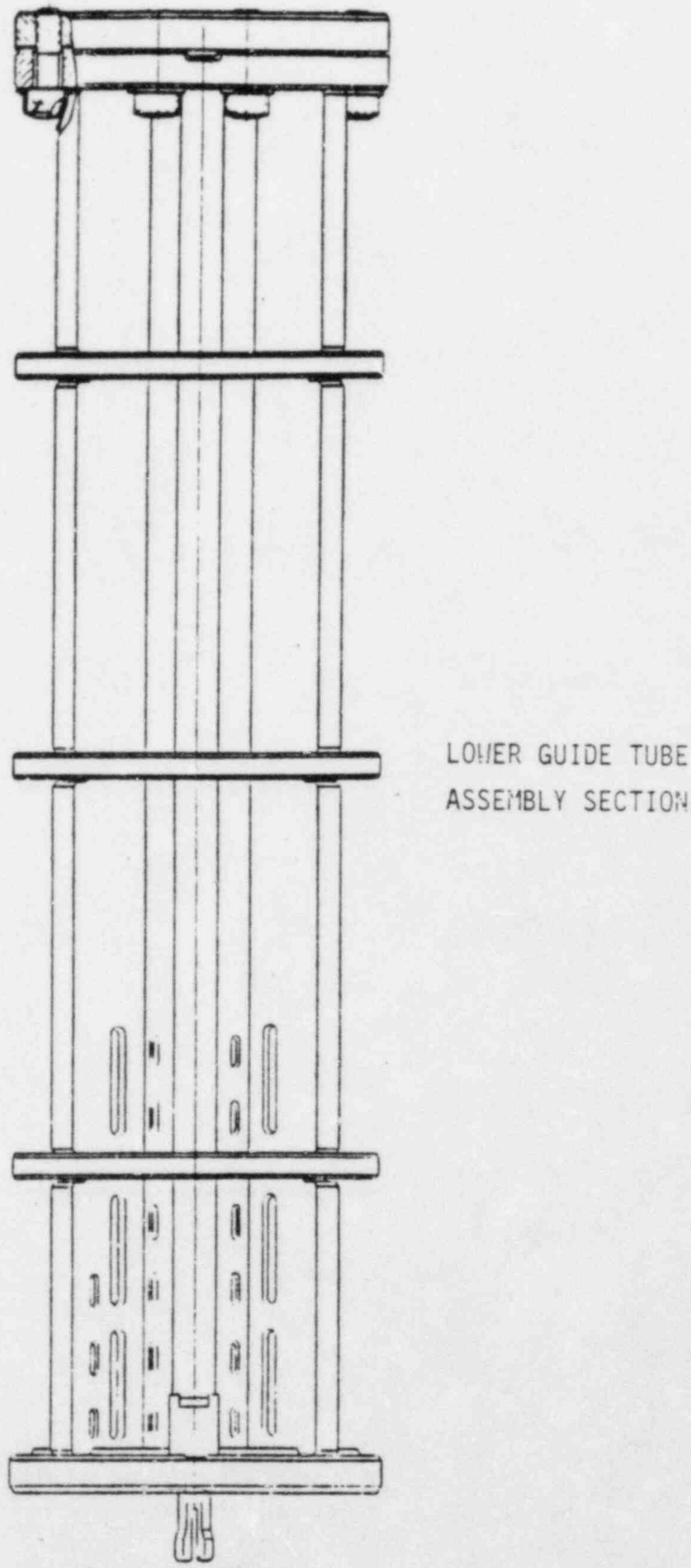


FIGURE 3

LOWER GUIDE TUBE
CONTINUOUS SECTION
INTERIOR CROSS-SECTION

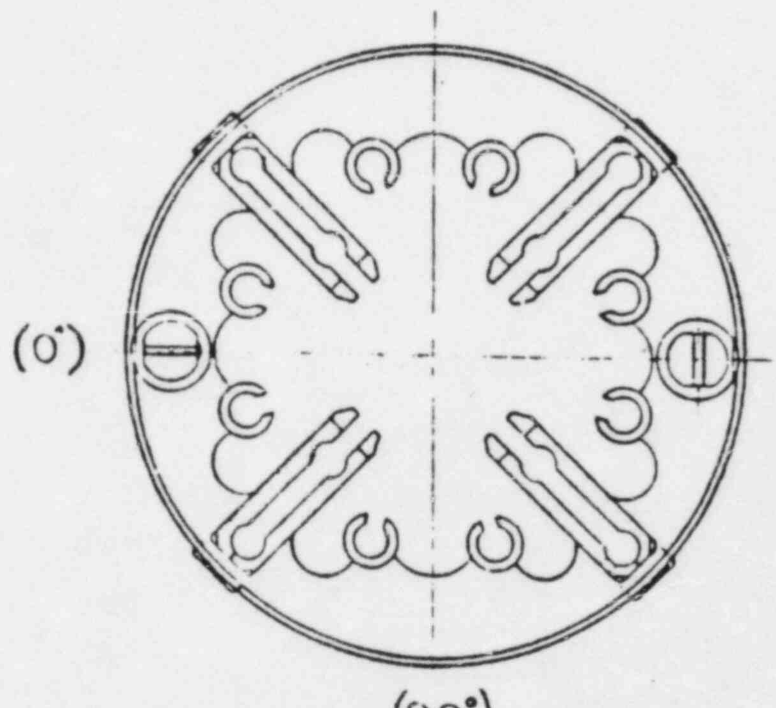


FIGURE 4