PGE Portland General Electric Company

Bart D. Withers Vice President

August 15, 1984

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Director of Nuclear Reactor Regulation
ATTN: Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington DC 20555

Dear Sir:

TROJAN NUCLEAR PLANT
Loose Parts Safety Evaluation

Our letter of August 9, 1984 provided a general description of the loose parts safety evaluation being performed for the Trojan Nuclear Plant as a result of missing components from the Control Rod Guide Tube (CRGT) support pins. Since that letter, one of the missing nuts, locking discs, and locking pins has been found. Nevertheless, a safety evaluation was performed assuming two missing nuts, two locking discs, and two locking pins. It is the conclusion of this safety evaluation that the startup and continued operation of the Trojan Nuclear Plant, with the possible loose parts described above in the Reactor Coolant System, does not have an adverse effect on public health and safety. A summary of that safety analysis is attached. This safety evaluation is similar to those performed for the Zion and Point Beach nuclear plants following their discovery of broken CRGT support pins.

Also attached are replacement pages to Attachment 1 of our August 9 letter, incorporating minor editorial changes to that document. Please replace the original pages of Attachment 1 with those attached to this letter.

Sincerely,

BDWithers

Bart D. Withers
Vice President
Nuclear

Attachments

c: Mr. Lynn Frank, Director
State of Oregon
Department of Energy

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SAFETY EVALUATION FOR TROJAN NUCLEAR PLANT
POSSIBLE LOOSE PARTS IN REACTOR COOLANT SYSTEM
AUGUST 1984

I. INTRODUCTION

The Trojan Nuclear Plant has replaced the Control Rod Guide Tube (CRGT) support pins and in so doing has discovered that some pieces are missing from the original support pins. Also, one of the steam generators experienced some impacting on its tube ends which required a repair effort to restore the tube ends to an acceptable condition. No tube end fragments were generated.

II. LOOSE PARTS EVALUATION FOR TROJAN PLANT - AUGUST 1984

This report addresses the potential effects of the loose parts resulting from failed guide tube support pins (Figure 1). The potential effects of loose parts on the Reactor Coolant System (RCS), Chemical and Volume Control System (CVCS) and Residual Heat Removal System (RHRS) are considered. The RHRS is addressed since it is open to the RCS during cooldown and therefore creates a path for the loose parts to enter the RHRS.

Table 1 contains a description of the potential loose parts resulting from support pins. The loose parts evaluation for Trojan addresses a core with 17 x 17 standard fuel.

- o Nuclear Fuel

  Because of the size of the components given in Table 1 and their possible location in the core during operation, it is highly unlikely that they could make their way into the fuel assemblies. If all the pieces were somehow able to migrate through the RCS and into the lower plenum of the reactor, they could possibly become entrained in the coolant flow and subsequently become entrapped in the bottom nozzle area. The chance that all of the pieces would congregate in one location is small. Information and discussions pertinent to each condition are given below.

  <u>Material Within the Fuel Assembly</u>

  The majority of loose parts considered have sizes such that most of the parts could not pass through the bottom nozzle plate. Those pieces that could pass through the bottom nozzle would not pass through the lower grid however, and would not affect the DNB evaluations for this core. Tests (Ref. 1) on open lattice fuel assemblies indicate that a blockage of up to 41% is acceptable, with disappearance of the stagnant zone behind the flow blockage after 1.65 L/D_E (length/equivalent hydraulic diameter). These types of local blockages have little effect on subchannel enthalpy rise and effect minor perturbations in local mass velocity. In reality, a local flow blockage is expected to promote turbulence, and thus, would likely not affect DNB at all. Local flow blockage within the active fuel would not adversely affect the Trojan ECCS performance analysis.

Material Entrapped by the Bottom Nozzle Plate

Because of the 0.376-inch diameter flow holes in the bottom nozzle plate, only the locking pins are likely to pass through the bottom nozzle plates and up into the fuel assembly. Assuming, conservatively, that several support pins failed and all the associated loose parts could pass completely through the RCS and be trapped by the bottom nozzles, the flow to any one assembly will not be completely blocked off. However, THINC IV predictions (Ref. 2) indicate that even when blockage completely covers the nozzle, full recovery of flow occurs about 30-inches downstream of the blockage. Thus inlet blockage effects would be limited to the lower portion of the active core, where DNB and LOCA are not limiting.

Results of the thermo-hydraulic evaluation show that the presence of these loose parts is not expected to result in any operational problems from a flow blockage/DNB standpoint.

One concern is the possibility of additional debris being generated by movement of the support pin components throughout the RCS. Small pieces of debris could be trapped in the fuel assemblies and lead to fuel rod fretting. Nevertheless, the number of fuel rod failures that could occur from these known missing pieces is within the lowest limit of fuel rod failures assumed in the safety analyses.

References

1. Basmer, P., Kirsh, D. and Schultheiss, G. F., "Investigation of the Flow Pattern in the Recirculation Zone Down Stream of Local Coolant Blockages in Pin Bundles," ATOMWIRTSHAFT, 17, No. 8, 416-417 (1972).

2. Hochreiter, L. E., Chelemer, H. and Chu, P. T., "THINC IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

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 or through the bottom flange of the RCC guide tube. 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

34 inches above the continuous guide tube region (refer to Figure 2). 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.

- 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.

- Steam Generator

If loose parts from a support pin were to enter the steam generator channel head the effects would be peening-type impacting on the tube-sheet cladding, tube ends and channel head. The steam generators have protruding tube ends that could be impacted by loose parts in the channel head. Loose parts impacting on tube ends may result in tube end deformation but there are no safety concerns. Experience during the current refueling outage has shown that repairs to damaged tube ends can be made without creating any new loose parts in the system. Eddy current testing of the tubes in all four steam generators demonstrated their integrity. Previous experience has shown that severe tube end deformation has only a small effect on coolant flow capacity.

The nut and locking disc diameters are too large to enter into the steam generator tubes in their original configuration. However, after impacting the tube sheets and tube ends several times, the locking disc can become misshapen to the extent that they and the locking pin could conceivably enter the steam generator tubes. 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.

Impacting of these loose pieces (pins) 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, and create constant wear at a localized region. In larger radius 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 any steam generator tube without becoming lodged.

o CVCS Components

<u>Letdown Line Isolation Valves</u>

Based on the size of potential loose parts in Table 1, the possibility exists that a loose part could wedge into one of the letdown line isolation valves and prevent it from closing. If this were to happen, Plant operation could continue as long as letdown isolation was not required. Should letdown line isolation be required, the redundant letdown line isolation valve could be closed or either of the Containment isolation valves further downstream could be closed.

<u>Letdown Line Orifices</u>

Loose parts smaller than about 1-inch diameter may pass through the letdown isolation valves and reach the letdown line orifices. Construction of the orifice is such that structural failure of the orifice is incredible, however, the orifice may become plugged. In the unlikely event of orifice plugging, one of the other two orifices may be placed in service.

Based on the size of the loose parts shown in Table 1, it is judged extremely unlikely that any loose pieces (approx. 1/4-inch diameter or less) will pass through the letdown orifices. Even if any of these small pieces do pass through the orifices, redundancy of isolation valves ensure the ability to isolate letdown if required. These small parts would eventually be trapped in the reactor coolant filter.

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 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, nor a support pin nut alone is able to pass through the steam generator in their designed configuration. However, after being deformed from impacting the RCS components, passage through a steam generator tube can occur.

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 support pin nuts are able to follow this path, the 1.130-inch diameter nut is too large to enter the 1-inch gap between the bottom of the vessel and the four secondary core supports. Because of the support pin nut, locking pin, and locking disc dimensions, they do not create a concern for wedging effects in the lower vessel region. That is, the nut cannot enter the above mentioned gap, and the other items have dimensions that preclude wedging effects.

Because of the low probability of the various assumed conditions occurring during the next fuel cycle and because of the loose parts dimensions, wedging of broken support pin parts is not a safety concern for Trojan.

- Control Rod Drive Mechanisms

  The foreign objects in the RCS would not be expected to reach the control rod drive mechanisms or to have an adverse effect on their operation. Proper operation of the mechanisms will be confirmed during Plant startup.

- Pressurizer

  Based on the most probable movement of the foreign objects and the flow rates experienced within the RCS, it is unlikely that the foreign objects would cause mechanical damage or become lodged in the pressurizer inlet piping or the pressurizer.

- Safety Valves

  Due to the physical separation of the safety valves from the remainder of the RCS, no adverse effect is expected to result from the foreign objects. These valves are located at the top of the pressurizer and since the loose parts will not get to the pressurizer, they will not affect the safety valves.

- Materials

  There is no anticipated long or short term metallurgical effect expected from the Inconel X-750 material in contact with the RCS metals.

- Radiological Effects

  The radiological effect of any material from the Inconel guide pins getting into solution into the reactor coolant is insignificant. Although Cobalt-58 (Co-58) can be produced from an (n, p) reaction on the nickel contained in the Inconel, the following steps are involved:

  1. Release of base metal to the RCS.
  2. Deposition of base metal in active core region.
  3. Activation of base metal.
  4. Release of activated metal from core region.
  5. Deposition of activated metal in components.

  Cobalt-58 is normally found in the reactor coolant and as deposits on component surfaces, but this is due primarily to the nickel contained in the massive amounts of Inconel steam generator tube material. The presence of these loose parts in the RCS could increase the level of Co-58 in the primary systems due to the fact that these parts may be impacting components of the system and wearing away. However, based on the experience during Cycle 6, where at least three parts were present in the RCS, the increase in levels would not adversely affect Plant operation or maintenance.

The major known radiological effect is the added exposure accumulated by personnel performing the additional inspections, maintenance or other activities, created by the presence of the foreign objects.

II. CONCLUSION

As presented in the above evaluation the ability of Trojan to startup and continue safe Plant operation with CRGT support pin loose parts does not constitute an unresolved safety question. The basis for this evaluation is the loose parts analysis contained in this evaluation.

The Trojan Technical Specifications and procedures require an evaluation of rod drop times before returning to power, and the performance of biweekly rod stepping tests to assure rod movement.

The evaluation of missing support pins does not affect the ability of the Plant to startup and continue safe Plant operation.

TABLE 1

POTENTIAL LOOSE PARTS GENERATED
FROM GUIDE TUBE SUPPORT PINS

Part	Dimensions	Weight
Nut (2)	1.3" long 1.13" diameter	0.2 lbm
Locking Disc (2)	0.812" diameter	0.03 lbm
Locking Pin (2)	1.125" long 0.25" diameter	0.016 lbm

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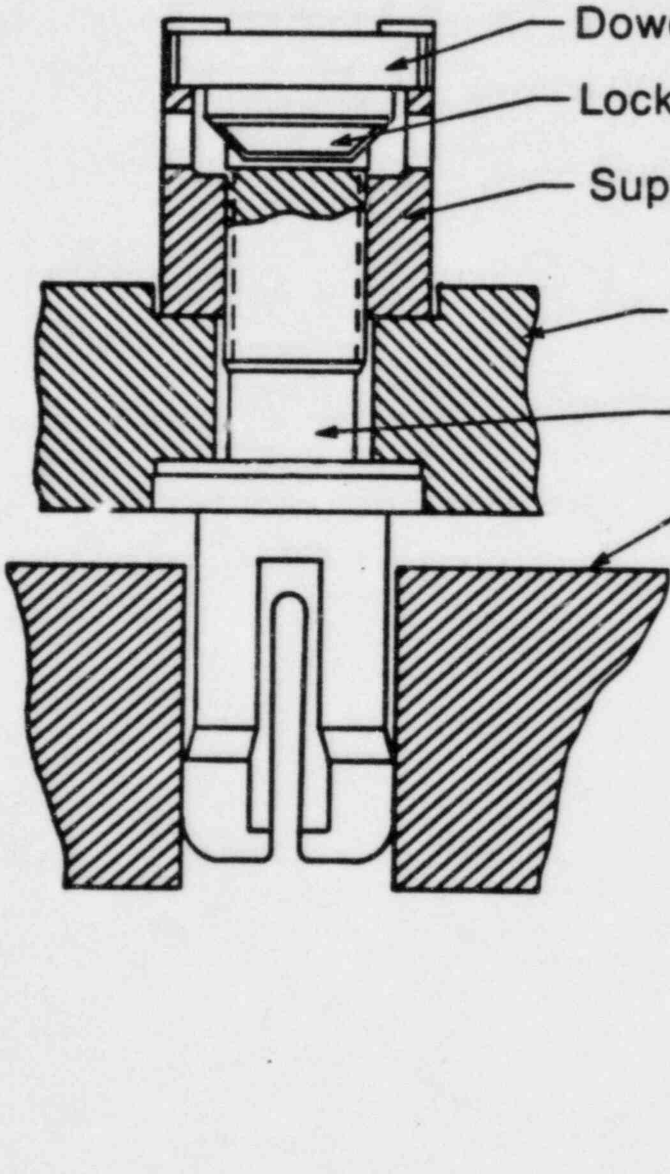


Figure 1
Cross Section of a Support Pin Assembly

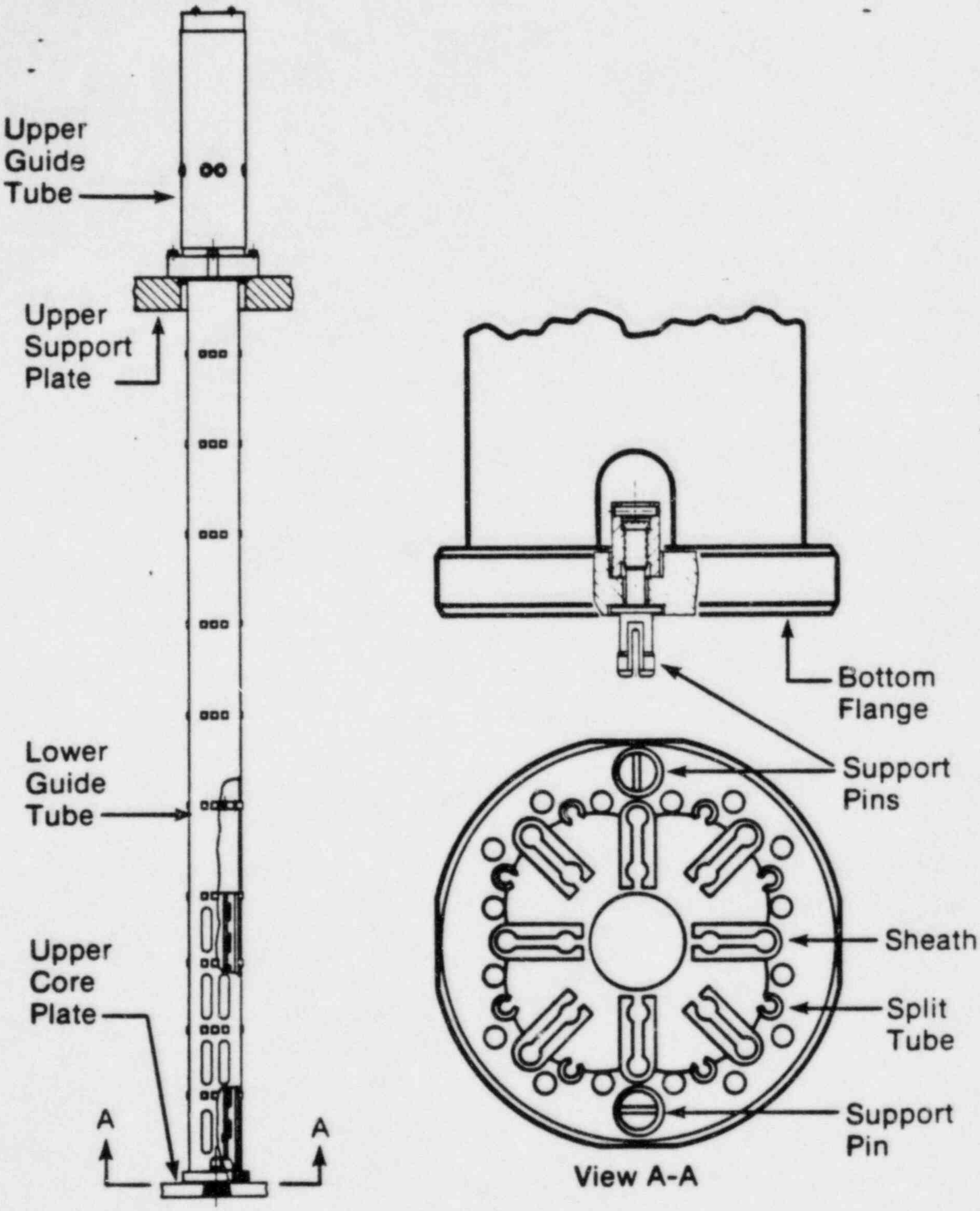
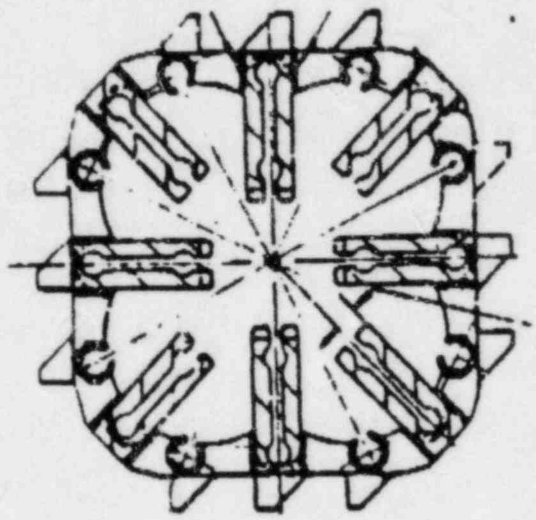


Figure 2
Control Rod Guide Tube Assembly



LOWER GUIDE TUBE
INTERNAL CROSS-SECTION

FIGURE 3

power plants. The Westinghouse correspondence identified the mechanism of failure as intergranular stress corrosion cracking (IGSCC) and emphasized that they did not feel the failure of support pins was a safety problem in operating nuclear power plants.

On July 23, 1982, the NRC issued IE Information Notice 82-29, "Control Rod Drive Guide Tube Support Pin Failures at Westinghouse PWRs". The NRC pointed out that the safety consequence of a support pin failure as a loose part is still under consideration by the NRC. In general, the NRC appeared to be accepting the Westinghouse position that CRDM misalignment was not credible in non-upper head injection (UHI) plants. This Information Notice was issued because of failures at the North Anna Nuclear Power Plant in 1982.

PGE evaluated the NRC IE Information Notice 82-29 as well as the recommendations provided by Westinghouse in a November 3, 1982 letter. PGE agreed at that time that cracked or broken support pins were not a safety hazard but more of an economic concern. PGE did not feel that an immediate problem was presented because the torque applied to the lock nuts on the support pins at the Trojan Nuclear Plant had been 135 ft-lb; the torque on those support pins that had failed were 200-210 ft-lb. Although Westinghouse was recommending solution annealing at 2,000°F for one hour, with subsequent age hardening at 1,300°F for 20 hours, the pins installed at Trojan were heat treated at temperatures of less than 1,800°F and age hardened. In 1983, support pin failures were discovered at Beaver Valley and Farley Unit I nuclear plants. These support pins that failed had been final torqued to 210 ft-lbs. In 1984, support pin failures were found at Point Beach I, Zion II, and Salem I. The support pin failures at Salem were the first pins that were similar to these at Trojan. In May 1984, following the removal of all fuel and the upper and lower internals from the Trojan Nuclear Plant reactor vessel, loose parts were found. Subsequent identification of the loose parts confirmed that they were in fact CRGT support shanks without the nuts. PGE decided to replace the CRGT support pins during the 1984 refueling outage, extending the outage approximately 20 days.

<u>Evaluation</u>

Figure 2 shows the new CRGT support pins used at the Trojan Nuclear Plant. The support pins have been evaluated to be less susceptible to failure for the following reasons:

1. The shank length has been increased.

2. The shank has been changed to a parabolic radius.

3. The installation torque has been reduced.

4. The material has been heat treated to improve its resistance to IGSCC.

5. The leaf spring design has been changed.

6. The locking mechanism has been altered.

Modification of the support pin shank provides improvement in the stress resistance of the pin. The longer shank length results in a lower thermal growth stress. The change in shank radius to a parabolic radius reduces the stress in the transition region between the shank and pin shoulder (the area where most failures have been observed to occur). The stress concentration factor of the pins has been reduced from 3.3 to 2.9 (assuming a pseudo-elastic analysis). These shank modifications will reduce the stresses in the pin and its susceptibility to IGSCC.

The installation torque of the old support pins was approximately 140 ft-lb, resulting in a preload stress of 63,200 psi. The new support pins have been torqued to a value between 120 to 130 ft-lb, resulting in a maximum preload stress of 45,300 psi. The installation torque of 120 to 130 ft-lb has been determined experimentally to be adequate to retain the locking nut and yet not create preload stresses on the support pin that make it susceptible to IGSCC. Installation of the new pins in water instead of air, as the old pins were, will assist in restricting the preload stress.

The same material used in the old pins, Inconel X750, is also used in the new pins with improvements in its microstructure. The heat treating process used for the new pins has enhanced the Inconel X750 microstructure, making it more resistant to IGSCC. Serious consideration had been given to using other materials, such as Inconel 718. However, Inconel 718 was rejected primarily because there was insufficient data available on it, and there was so much data, tests, and actual operational data available on Inconel X750. At this time, Inconel X750 is still the best material to be used in an application of this nature.

The Inconel X750 used for the support pins has been solution annealed at 2,025°F for 1 hour and 15 minutes and age hardened at 1,300°F for 22-1/2 hours. Cooling from both processes was by air, although cooling from the solution annealing process was expedited by fan. The material was ultrasonically inspected and determined to be free of defects and indications. Samples of the material were subjected to a rising load test with acceptable results. (The rising load test is described by Military Specification MIL-N-24114C.) Analysis of the material after heat treating indicated the overall grain size averaged 5.0, which provides good resistance to IGSCC. Experimental results indicate that Inconel X750 heat treated in this manner is very resistant to IGSCC without cracks or notches.

Part of the machining process for the support pins included polishing to a 30-microinch finish or better in the shank radius. The pins were carefully examined to ensure the surface was free of cracks or other

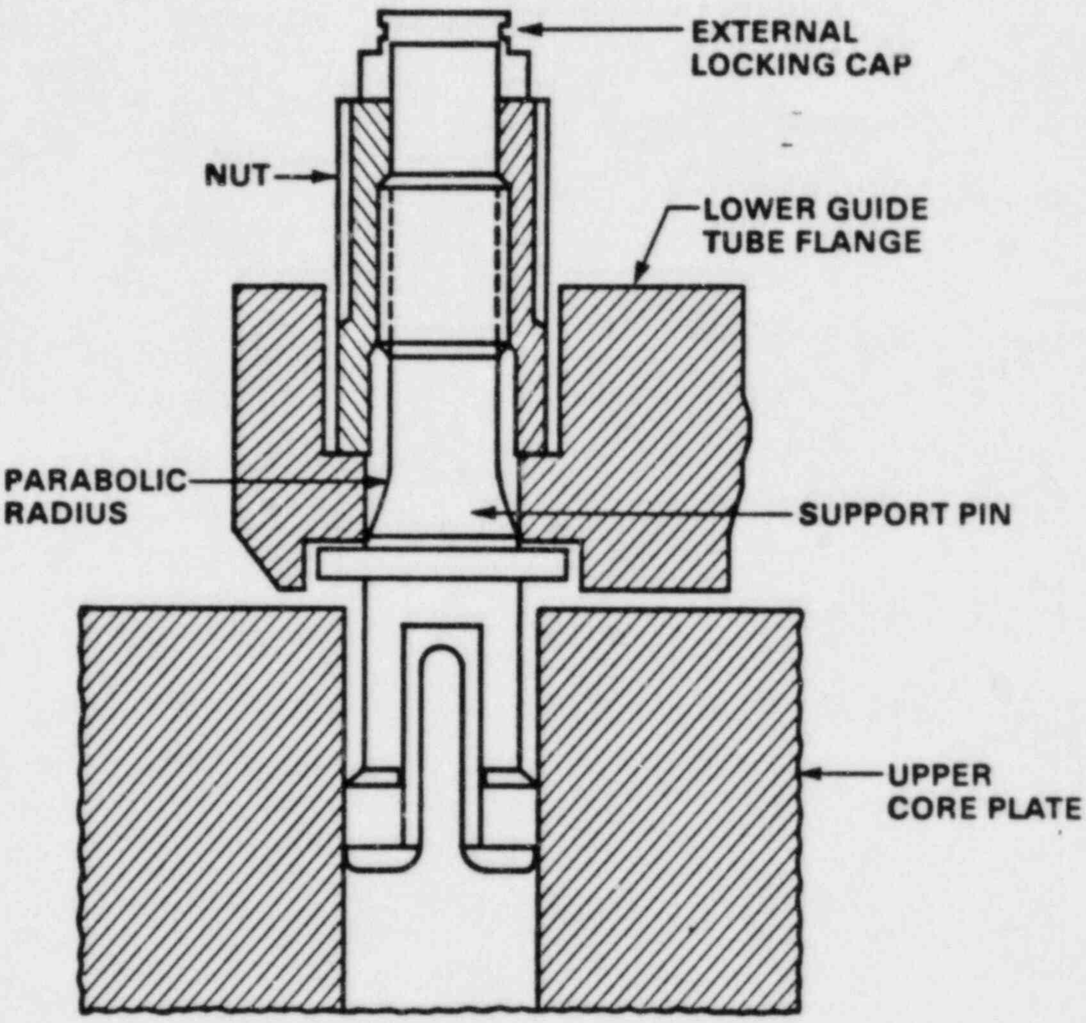


Figure 2
Cross Section of a Replacement Support Pin