GPUN

EVALUATION OF POTENTIAL EFFECTS OF LOOSE OTSG PLUGS IN THE TMI-1 RCS

CONTENTS

PAGE

INTRODUCTION

Recently, increased primary to secondary OTSG leakage led to the discovery that a total of six Westinghouse-design rolled tube plugs were dislodged from the lower OTSG tubesheets. The plugs have not been recovered from the primary system. Pull-tests of the remaining plugs in the upper and lower tubesheets have been performed to determine needed repairs. Qualification of repairs will provide assurance that no increase in the loose plug population is likely. This evaluation examined potential effects of the loose plugs in the RV and RCS during plant operation.

1.0 SUMMARY

The evaluation considered potential damage and safety effects of a small population of loose plugs on RCS pressure boundary, RV internals, fuel assemblies, control elements, instrumentation, and other core components. Components and connections in the remainder of the RCS were also evaluated. Although the plugs are not expected to fragment based on known materials properties, the effects of smaller loose pieces were addressed. The conclusions of the evaluation would not change for loose plug populations an order of magnitude greater than six. OTSG plug repairs will assure that an increase in the number of loose plugs is very unlikely. In the interest of conservatism, certain potential effects possible only with much larger populations of loose parts were also included.

Concerns of impact, wedging, wear, and flow blockage were investigated. No credible effects due to these concerns were identified that could place the plant in a configuration that would exceed existing safety analyses envelopes or prevent safe reactor shutdown and heat removal capability. The likelihood of any significant degradation to the proper functioning and safety margins of the fuel and all other primary system components is extremely low. In the unlikely instance of breached fuel rod cladding or restricted control rod motion plant radiation monitoring limits and previous safety analyses assure safe shutdown. No Tech Spec safety margins have been reduced. The plugs do not introduce any significant increase in the probability of occurrence of events previously analyzed in the safety analyses nor any increase in the consequences of those events. No potential for an accident or malfunction different than those previously analyzed has been created. Thus no unreviewed safety questions have been created. These judgements are based on:

- the small size and mass of the plugs,
- the small loose plug population
- the expected randomness of plug flow paths in the RV.
- the minor consequences of all credible effects

It is concluded that TMI-1 can be safely operated with a small population of loose OTSG plugs in the RCS with reasonable assurance of no significant increase of undue risk to the health and safety of the public or plant personnel.

2.0 PLUG POPULATION

A total of six (6) plugs were released from the lower tubesheets. Repairs to the remaining rolled plugs are expected to minimize the potential for any additional loose plugs. The material properties of the plugs will preclude fragmentation (see below, Section 3.0). Nevertheless, for certain potential damage modes it was assumed that some fragments of the various sizes necessary to cause the effect could detach from the plugs. Because of the small plug size and the expected random flow travel paths, a population an order of magnitude larger than six would not change the results of the evaluation. In certain instances, for the sake of conservatism, the evaluation addressed potential effects that could only be caused by much larger populations of loose pieces.

3.0 PLUG CONFIGURATION AND PROPERTIES

The plug design is shown in Figure 1. It is basically a thin-walled hollow cylinder of Inconel 600 material 3-1/2" long with an OD of about 1/2"; it weighs about 1-1/4 ozs. Inconel 600 is a standard primary system material. It has high strength, high ductility and high toughness properties. Hardness is in the range of RV components materials (primarily stainless steels) with which the plugs will interact.

Ductility and hardness properties have been provided by Westinghouse (Ref. 1). The as-built, thermally-treated plug has an elongation of 40-44% with a Rockwell B hardness of 80. After installation expansion the rolled plug hardness would increase to about R_B90. Figure 2A shows Inco data for Inconel 600 rod and sheet material. According to Westinghouse, their mechanical data for the plugs falls more closely on the Inco sheet and strip plots. This is conservative for minimum elongation properties. For R_B80 the Inco data shows 42-43% elongation confirming the Westinghouse values. At R_B90 Inco elongation decreases to about 33% which is still very ductile. For rod material from which the plugs are actually made the Inco charts give slightly higher elongation values (49% for R_B80; 41% for R_B90), but the difference is about the same as that for sheet and strip. Thus, the conclusion may be drawn that the plugs retain a large amount of residual ductility (at least 33%) after coldworking due to expansion.

Inconel 600 also has good impact strength ("toughness") characteristics. Figure 2B shows manufacturer's Charpy data at room and elevated temperatures for material treatments representative of the plugs. V-notch strength of cold-drawn material at typical RCS temperatures is about 90 to 95 ft-lbs, well above any impact energy that will be experienced by the plugs.

Based on these ductility and toughness properties it is very unlikely that plug fracture or fragmentation will occur in the RCS.

As described in Section 4.0, the plugs will most likely be held by flow within the fuel assembly lower end fittings during plant operation. In this location they will be exposed to radiation from the core. In general, irradiation effects due to neutron fluence in nickel-based materials will tend to increase hardness and strength and decrease ductility and toughness. Tests on Inconel 600 have shown that material irradiated to neutron fluences much greater than those possible for the plugs to accumulate in TMI-1 retains about 7% ductility at typical reactor temperatures (Ref. 2). Earlier tests at lower fluences (though still higher than possible TMI-1 levels) showed significant retention of ductility for Inconel 600 at TMI-1 temperature conditions (Ref. 3). The tests also showed that virtual loss of all post-irradiation ductility occurs only at temperatures far in excess (1100°F) of TMI-1 RCS conditions. Due to the lower TMI-1 neutron fluences and to uncertainties of exposure times and actual plug locations with respect to core flux patterns the available data indicate that irradiation effects will not significantly degrade the original plug material properties. Thus sufficient plug ductility and toughness will be retained after irradiation to support the conclusion that fracture or fragmentation will remain unlikely.

Due to the small plug mass and relatively low velocities possible in the RCS large plug deformations are not likely. However, due to the hollow thin-wall design of the plugs, some deformation may occur as they impact against the much more massive RV internals components or the upper OTSG tubesheets. Because of their high ductility and toughness, they would bend and flatten, tending to compact into minimum volume configurations, rather than break up into smaller pieces. Nevertheless, the effects of plug fragments are addressed where appropriate.

The plugs will be subject to some wear. Inconel 600 will tend to wear by the abrasion of tiny hardened particles rather than by a spalling effect that might produce chips or slivers. The effects of such wear particles in the RCS would not be distinguishable from those of routine particulate matter in the primary coolant.

4.0 PLUG FLOW PATHS

Calculations have shown that loose plugs from the lower tubesheets would be lifted and entrained in the cold leg flow for all pump flow configurations (Refs. 4 and 5). Also, any loose plugs from the upper tubesheets would not be entrained under one-or two-pump backflow conditions (Ref. 6). Thus, the latter would remain in the OTSG upper plenums and have no potential effects in other RCS regions.

The most probable flow path for a plug is out of the lower dome of the OTSG, up the vertical cold leg, through the RC pump, into the horizontal cold leg, through the inlet nozzle, downcomer, lower plenum, lower internals, into the lower end fittings (LEFs) of the fuel assemblies where they will be trapped by inlet flow. Inlet flow paths and RV component arrangements are shown in Figures 3, 4 and 5.

Flow holes in the lower core grid assembly are all large enough for the plugs to pass through to the LEFs. An intact plug (1/2" OD) cannot pass through the 0.4" wide slots in the LEF grillage (Fig. 6). It is possible that a flattened plug or large fragment would fit through one of the larger slots in the LEF. If so, it would be trapped by the lower spacer grid which has tighter clearances within an "egg-crate" matrix (Fig. 7).

It is, therefore, expected that all loose plugs will be confined to the lower portions of the RV and fuel assemblies. However, based on available drawings, secondary flow paths can be defined through the periphery of the lower core plate grid and into the core former baffle. From the baffle a plug could reach either the upper RV plenum (through the upper core plate grid) or the fuel (through the vertical baffle plates). The likelihood of this occurring is very low. First, less than 3% of total core flow enters the baffle region. Second, even if a plug passes through the periphery of the lower grid plate it will most likely be trapped beneath the lower horizontal baffle former plate. There are eighty (80) 1-5/16" diameter flow holes uniformly distributed around each of the eight levels of former plates. The chance of a plug passing through all eight levels and reaching the upper plenum without being trapped is extremely small.

There are also eighty (80) 1-3/8" diameter flow holes through the vertical baffle plates at each of several levels around the core. These could permit a plug (or fragment) to interact with a fuel assembly. This interaction is very unlikely since there is little pressure difference between the core and baffle regions and thus low flow through the holes. The greater likelihood, if a plug reached the baffle at all, is that flow within the baffle would carry it past the hole and it would remain trapped.

Although not expected to form, small plug fragments could be carried past the lower spacer to the upper FA grids where they would lodge. Very small fragments could pass out of an assembly to the upper plenum. Another possible pathway for very small pieces to the upper RV regions is through the FA control rod guide tubes either by entering the 1/8" flow hole in the lower GT end plug or the 3/16" flow holes near the bottom of the GT wall. Such entrance is unlikely since the flow rate in the guide tubes is low relative to the external coolant velocity.

Possible paths into RCS connections are discussed in Section 6.8.

Potential effects of both expected and low probability plug paths are addressed in the evaluation.

5.0 LOOSE PARTS DETECTION

There are two Loose Parts Monitoring (LPM) sensors for the lower RV plenum region located outside the vessel on the guide tubing for incore instrument strings 5 and 13 (core locations E-9 and M-7). There are also two LPM sensors at the upper tubesheet level of each OTSG on the external seismic supports. LPM alarm sensitivity is about 0.5 ft-lbs surface impact within three feet of a sensor. For normal RCS flow conditions tests have shown a lower detection limit of about 0.25 lbs. (Ref. 7).

The weight of a plug is on the order of 0.08 lb. Therefore, unless a plug impacts as closely as possible to a sensor at maximum attainable flow velocities it is unlikely that it will be detected by the LPM system.

6.0 EFFECTS EVALUATIONS

This section provides detailed evaluations of the potential damage and safety effects of the loose plugs on all structures and components in the RV and RCS. Table 1 provides a summary of the evaluations. Potential plug effects in RCS connecting systems are summarized in Table 2.

6.1 RCS Pressure Boundary

Based on available drawings no area was identified where loose plugs can challenge the RCS pressure boundary. Plug impact energies are too low to cause significant damage to the more massive RCS boundary components. Also, no locations in the RV design were determined in which a plug piece wedged between major RV boundary components could cause mechanical loadings (e.g., due to differential thermal expansions during heatup) or wear of such magnitude to damage the pressure boundary or create a safety concern.

It is very unlikely that a plug will enter the upper head of an OTSG. If one did, it could not cause sufficient damage to the tube-to-tubesheet weld area to affect the pressure boundary. Average inlet flow velocity of primary fluid to the OTSG is about 60 ft/sec (Ref. 8). A plug at that velocity would have a kinetic energy of about 4 ft-lbs. Even repeated impacts of this magnitude could cause only minor indentations at the tube connections (or to existing plugs). TMI-1 tube ends have been machined off down to the weld region and do not extend into the OTSG plenum, thus further reducing the potential of impact damage to the connection. Also, since all unplugged tubes are kinetically expanded the actual primary-to-secondary pressure boundary is well below the top of the tubesheet. Therefore, OTSG leakage due to plug impact is not feasible.

Another concern was identified regarding breach of an incore detector string resulting in a change of normal pressure boundary configuration. This is addressed below in Section 6.6.

6.2 <u>RC Pumps</u>

The small plugs will pass through an RC pump with no significant effect on the pump. Any impact damage to the more massive impellers would be minor. Also, there is a minimum clearance of 1" between the impeller blades and the pump casing that will easily pass the 1/2" plug. Thus jamming is not a concern.

Fragmentation of the plugs in the pumps or elsewhere is unlikely due to their high ductility. If any fragments form it is highly improbable they will harm the pumps. The only potential concern of metal fragments in an RC pump is seal wear. During normal operation seal injection flow pressure is at 50 psi greater than RC pressure, thus preventing primary coolant (and any entrained chips) from entering the seal regions. In the event seal injection is lost only a very small fragment (about 0.020" thick) could fit through the existing diametral gaps to reach the seals. Seal wear could occur causing increased leak-off. This would be an operational problem within the capabilities of the makeup system. In the extreme, it can be postulated that if both the #1 and #2 seals were damaged a potential for a small break LOCA would exist if the pressure differential on the #3 seal were greater than 50 psi. This size break is covered by existing safety analyses and does not represent a new safety concern. Given the characteristics of the plugs and pump flow patterns as described above the latter scenario is considered implausible.

6.3 RV Internals

Most loose plugs interactions will be with portions of the downcomer and the lower RV internals. As with the vessel, the geometry and mass of internals components such as the core support and thermal shields, incore nozzles, incore guide tubes, flow distributor and lower grid plate assembly preclude significant damage due to plug impact.

These conclusions are supported by an in-depth study done by Westinghouse for Zion 1 (Ref. 9) and accepted by the NRC (Ref. 10) on the impact effects of similar size loose parts on PWR internals. Results showed that mechanical effects such as perforation, denting, bending, and dynamic responses of the internals were all well within allowable limits.

As the least massive structures in the lower plenum the incore instrumentation penetration nozzles were examined. The nozzles are stainless steel vertical right cylinders with a minimum OD of 1-3/4" and about 1-ft high. Minimum wall thickness is 0.56". The nozzles are root-welded in the lower head. Assuming a conservative maximum plug velocity of 70 ft/sec (Ref. 4), direct impact energy on a nozzle would be about 6 ft-lbs. Significant damage would not be expected due to impacts of this magnitude.

Thus no impact safety concern exists for the lower RV internals.

Localized wear or fretting caused by a plug wedged between internals components or caught in local turbulence would also not lead to any feasible safety concerns.

The same conclusions hold true for the upper internals (upper grid plate, plenum assembly, control rod guide tubes) in the remote instance that one or more plugs finds its way into the upper RV plenum. Vent valves are discussed separately in Section 6.7.

6.4 Fuel Assemblies

As described above, the most likely path for the plugs is to the bottom of the core where they will be trapped in the lower end fittings (LEF) of the fuel assemblies. Potentially this can cause impact, wear and flow blockage effects in the FA. Other possible, though much less likely, plug/FA interactions are also addressed. A full fuel assembly is shown in Figure 8.

6.4.1 <u>LEF Impact</u> - Plugs impacting within the LEF region may cause minor damage to the grillage, FA guide tube nuts, or the incore instrumentation guide bar. However, since a plug will randomly collide with many lower internals components and is moving upward against gravity before reaching the LEF, it will not be entrained for a long enough distance to have full flow velocity (approx. 18 ft/sec) upon impact. Therefore, damage is not expected to be significant. Minor fracture or distortion of LEF structural components is not a safety concern.

6.4.2 LEF Wear - Minor damage due to the loose plugs might occur to LEF parts from turbulent abrasion or fretting wear. This is not a safety concern since it would not degrade the function of the LEF.

Only severe wear to the lower guide tube fastener nuts in the LEF would be of some concern. The sixteen guide tubes are the primary structural elements of the B&W fuel assembly. If the majority of the nuts in a given assembly were damaged to the extent that they lost their function as fasteners the structural integrity of the FA would be degraded. Such damage is not considered feasible.

First, since it is extremely unlikely that more than one plug will reside in any LEF, only several nuts would be in contact with a plug (i.e., the likely position of a plug during plant operation is flat against the LEF grillage trapped among only 3 or 4 guide tube nuts; See Figure 6. The nuts are made of 304 stainless steel which has a hardness similar to that of the Inconel 600 plugs. Therefore, material wear losses should be similar for nut and plug. The nut design is such that the portion that would be exposed to plug abrasion is much thicker (1/4") than the thin-wall plug (1/16"). Thus a given area on the nut is unlikely to wear to the point of significant damage before the plug wall would wear through and change the contact configuration. Further, the flow through the LEF is not expected to cause excessive motion of the plug once it is held against the grillage. Thus the small mass plug would not be expected to cause signficiant abrasion.

Finally, even in the unlikely event several nuts were damaged to the extent that the guide tubes were no longer fastened, the fuel assembly structural integrity would still be sufficient for safe removal during refueling. This is because the weight of an entire assembly can be supported by only several intact guide tubes of the sixteen available. No other effects would be expected due to the damaged nuts since the guide tube lower end plugs extend through alignment holes in the LEF grillage and would remain in place even without the nut attached. Therefore, wear of the lower guide tube nuts is not a safety concern.

6.4.3 Fuel Rod Damage - A more significant concern, though also unlikely to occur, is fretting damage to the fuel rods. As described above under "Flow Paths" (Section 4.0), and as shown in Figures 6 and 7, it is possible for a deformed plug (or portion of a plug) flattened along its axis to fit through the 0.4" wide slots of the lower grillage and interact with the fuel rods and lower spacer grid.

It was also determined that a small possibility exists of a plug (or fragment) entering the core baffle and interacting with a peripheral fuel assembly through the flow holes in the vertical baffle plates. As discussed in Section 4.0, such interaction is very unlikely since there is low flow through the holes.

The primary concern if these interactions were to occur is breach of the fuel rod Zircaloy cladding caused by fretting damage and the consequent release of radioactive fission products to the primary coolant. A secondary concern is damage to the Inconel spacer grid structure. Spacer damage can also lead to fretting breach of the fuel cladding.

Another concern if through-wall damage to the cladding occurs is waterlogging of the fuel rod. Differential pressure between the primary coolant and internal pin gases can allow water to enter the rod. Under conditions of very fast power increases, rupture of a waterlogged pin can occur.

A B&W evaluation of this phenomenon based on SPERT experiments was discussed in the Midland FSAR (Ref. 11). Based on a comparison of normal operating conditions for a B&W 177-FA plant and the SPERT test conditions and results the probability of failing a waterlogged rod is quite small. Even in the extreme case that a failure should occur, the pressure pulse from the ruptured rod should not cause significant damage to the rest of the core. The worst effect of SPERT failure tests under severe conditions was bowing of adjacent fuel rods. Such effects would not be expected under TMI-1 operating conditions and lower fuel energy densities (approx. 100 cal/gm).

Based on B&W plant operating experience, fretting breach of fuel rods is a low probability occurrence.
Given the small population of plugs and the low probability of a plug reaching the fuel or of fragment formation, the operational risk of fuel rod failure at TMI-1 is acceptably low. Coolant activity levels are closely monitored and controlled by Tech Spec limits. Therefore, any fuel rod damage caused by loose plugs would be detectable and would not decrease existing safety margins or threaten safe plant shutdown.

6.4.4 Fuel Assembly Blockage - An intact, nondeformed plug (approx. 1/2" OD) cannot fit through the 0.4" slots of the LEF. It will, therefore, cause local flow blockage. A totally flattened plug seated across the flow slots will have a similar effect. Totally flattened plugs will not actually occur, but represent the maximum area of a deformed plug. Figure 9 shows these two plug configurations to scale overlaid on the LEF flow area.

Dimensional comparisons of the plugs and the LEF flow area show that with ideal positioning it would take a minimum of about 30 intact plugs to effectively block an entire fuel assembly inlet. For totally flattened plugs (3-1/2" x 7/8") it would take a minimum of about 20. In the operating core (due to deformed plug geometries and irregular positioning) it would actually require many more plugs to even approach total FA blockage.

TMI-1 inlet flow is distributed to all fuel assemblies. There is some flow bias to ensure that the hotter interior core region receives greater than average flow. The even flow and interactions with the lower internals will distribute the loose plugs in an essentially random pattern to the bottom of the core.

These dimensional and flow considerations show that the small loose plug population cannot cause significant core flow blockage concerns. There is a very low probability that more than one plug will lodge in any single fuel assembly. One intact plug would create about a 3% inlet flow blockage.

Flow blockage in the bottom of a fuel assembly raises two concerns: 1) Whether a change in the hydraulic lift characteristics of the FA occurs; and 2) whether the thermal-hydraulics performance of the FA is affected.

6.4.4.1 FA Lift - Fuel assembly hydraulic lift forces are a direct function of the total pressure drop along the assembly. The pressure drop for a single assembly is largely controlled by the overall core pressure drop which is determined by the essentially uniform pressures over the core inlet and outlet plenums. Partial blockage of a single assembly will thus have little effect on pressure drop. Local effects of the blockage are compensated for by inlet velocity changes and flow-mixing due to pressure differences between assemblies. Therefore, no decrease in liftoff margin is expected to occur due to the small blockages caused by the plugs.

Conservatively ignoring compensatory pressure drop effects, the static flow force on an intact plug held against the LEF inlet under worst-case flow conditions (500°F, 4-pump operation) is about 1 lb (Ref. 12). This is less than 1% of the FA holddown force margin for TMI-1 which exceeds 100 lbs under these conditions.

The extreme postulated case of fuel assembly lift has been evaluated previously in a GPUN submittal accepted by the NRC (Ref. 13). That study determined that an assembly could lift no more than 1.5" with low consequent impact energy. Effects due to reactivity changes, FA vibration, and spacer grid mismatches would be minimal.

Based on the above considerations FA lift due to loose plug flow blockage will not occur and is not a safety concern.

6.4.4.2 DNBR Effects

The thermal-hydraulic characteristics of a fuel assembly can be affected by flow blockage. Significant flow reduction could cause margin decreases or violation of DNBR limits, particularly if the reduction occurred in the hot assembly. In the Midland FSAR B&W evaluated DNBR effects as a function of hot channel flow blockage using the BAW-2 critical heat flux correlation. Results are shown in Figure 10. Allowing for differences in core parameters, these results may be taken as representative for TMI-1. A DNBR of 1.3 is not reached until about 70% blockage occurs.

PWR loose parts evaluations performed for Zion-1 (Ref. 14) and accepted by the NRC (Ref. 10) have shown that flow recovery takes place within fairly short distances downstream of even a totally blocked fuel assembly inlet. Tests referenced in the same evaluations have verified this quick recovery and shown that local flow blockage (up to 41%) has little effect on subchannel enthalpy rise and causes only minor perturbations in local mass velocity. Also, small blockages create turbulence which tends to reduce enthalpy in a hot channel, thus further reducing potential DNBR effects.

Given the small blockages possible due to the loose TMI-1 plugs and the location at the LEF inlet, any flow perturbation effects will likely disappear before reaching the bottom of the active fuel. Therefore, no decrease in DNBR margin is expected and no safety concern exists. This conclusion may be extended to other TMI-1 limits that are based on the thermal performance of the fuel assembly (e.g. LOCA limits).

6.4.4.3 Local Hot Spots - For very small plug fragments, the formation of local hot spots can be postulated for a chip wedged against a fuel rod. A single chip, even in a hot region of a rod, is not likely to create enough flow blockage and consequent heat transfer reduction to cause cladding damage. Flow around the piece would be turbulent. Also, due to the rough configuration of a chip it would have only point contacts with the rod (rather than a flush seal effect) allowing flow between them and minimizing any decrease in heat transfer. Given the small number of loose plugs and low likelihood due to the Inconel 600 properties of chips breaking off, it is not feasible that enough small pieces would accumulate at a specific location to cause overheating damage due to localized hot spots.

6.4.4.4 LOCA Considerations - TMI-1 is LOCA-limited (minimum allowable kw/ft) at the two-foot level. This is based on the generic B&W ECCS analysis for the limiting cold leg large break (Ref. 15). As discussed above, expected actual plug blockages will not affect LOCA limits. However, even for a postulated case in which the plugs produce significant blockage in the limiting fuel assembly, no effect would be predicted. The analysis shows that for this event almost immediate flow reversal occurs during blowdown. Most plugs trapped in the lower end fittings would be flushed into the lower RV plenum where they would have no feasible effect on the outcome of the event.

6.4.5 Other FA Effects - As determined above, the small possibility exists of a plug or fragment finding its way to the upper plenum and thus to the top of the core. Several other very low probability damage effects are therefore considered.

6.4.5.1 Upper End Fitting - At the top of the core a plug could enter the upper end fitting (UEF). Entrance is very unlikely since flow at this location is turbulent and in an upward direction out of the UEF. In any event, plug residence time would likely be short and impact velocities relatively low. Minor collision or wear damage might be caused to UEF structural components such as the grillage, guide tube nuts and holddown spring spider. Such damage is not a safety concern.

6.4.5.2 Holddown Springs - In the remote instance that a plug or fragment remained in the UEF it could conceivably jam in a fuel assembly holddown spring in such a manner as to cause stress concentration leading to fatigue failure. Failure of FA holddown springs has been evaluated in a previous GPUN submittal to the NRC (Ref. 13). That study, based on evaluations performed by B&W, concluded that no significant safety concerns exist for TMI-1 operation with broken holddown springs.

Under the most extreme thermal-hydraulic core conditions for liftoff (500°F, 4-pump startup) no TMI-1 assemblies were calculated to lift. Any piece of a broken spring that escaped from the upper end fitting and large enough to do significant damage to any RCS components would be detected by the Vibration and Loose Parts Monitoring System. Reactivity effects of hypothetically lifted assemblies would be minimal. No excessive vibrations would occur due to lifting of an assembly and lateral repositions would be restricted by adjacent assemblies (or core baffles). Any mechanical damage to a lifted assembly would, therefore, be limited to minor wear phenomena. These conclusions were found to be satisfactory in an NRC Safety Evaluation (Ref. 16).

6.5 CONTROL RODS

An important concern with loose parts in the RV is restriction of control rod motion, particulary regarding insertion capability and scram times during a reactor trip. Given the small population of oose plugs and the unlikely formation of smaller fragments, the chance of CRA movement being affected is very low. Nevertheless, potential CRA effects are addressed.

Low probability pathways for the plugs to the upper plenum have been defined. It should be noted that even if a plug reached the top of the core the pattern of flow exiting the FA upper end fitting is away from the inserted CRAs and the periphery of the upper plenum CRA guide tube opening. Intact plugs, therefore, are not expected to interfere with CRAs.

Small fragments formed in the lower core regions would be trapped in the FA spacer grids. It may be postulated that very small fragments could be carried all the way up through the assembly and enter the CRA guide tube in the upper plenum. It would also be possible, though very unlikely, for small pieces to be carried to the top of the RV and enter the Control Rod Drive Mechanism clearances at the bushing and seal areas around the CRA leadscrews. These gaps would only allow pieces 1/8" thick or smaller to enter. As described previously, formation of such fragments is not expected to occur.

6.5.1 Control Elements - The control rod (CRA) and axial power shaping rod (APSRA) assemblies (Fig. 11) are inserted into the FA guide tube openings in the UEF. During normal operation in the all-rods-out mode portions of the thin-wall stainless steel rod cladding are exposed to coolant flow within and above the UEF plenum. A loose plug in this region might collide with the rods causing some damage. Collision is very unlikely since flow exiting the UEF is away from the control rods. Given the relatively low plug impact energy in the turbulent flow, severe degradation of the cladding would not be expected even if collision occurred.

In the event the cladding of a CRA was breached some Ag-In-Cd poison material might leach into the coolant. Any poison isotopes in the coolant would likely be detected by normal chemical sampling analysis. Such damage to a CRA is not a safety concern since only a small amount of poison would be expected to be lost and reactivity effects would be insignificant. This is not a concern for APSRAs since the poison region is always in the core in the bottom three feet of the rods and is sealed from the upper portion of the cladding tube.

In an extreme case plug impact might cause distortion of the cladding to the extent that rod insertion would be restricted. This is not a safety consideration for APSRAs since during normal operation they are positioned within the insertion limits established in the safety analyses and have no scram function. Restriction of CRA insertion is addressed in the following section.

6.5.2 Delayed CRA Insertion

Small entrained pieces passing into the control rod guide assembly would probably be ejected at the bottom of the guide where 60% of the flow exits through holes in the guide wall (Fig. 12). If a piece stayed in the guide it would likely be carried out the top with the remaining 40% of the flow. There is a small possibility that fragments could become trapped within the guide.

Design of the brazement is such that the "C" and split-tubes limit access to the CR clearance gap (Fig. 13). Turbulent flow would also tend to minimize pieces settling between the rods and the guides. CR diametral clearances are 0.12" and 0.09", respectively, for the "C" and split-tubes. It is unlikely that pieces small enough to be carried through a fuel assembly would cause significant interference in these gaps.

Nevertheless, it can be postulated that trapped pieces might create increased drag forces on the CRA during scram thereby increasing drop times.

At the top of the RV there is a remote chance that a small piece could interfere with a leadscrew on a CRDM.

The CRA drop time used in safety analyses is conservatively slower than that experienced in field performance tests. Field tests on drop times required prior to each cycle restart show results in the range of 1.1 to 1.3 seconds to 3/4 insertion, versus a conservative value of 1.66 seconds to 3/4 insertion which is typical of safety analysis calculations for 177 Mark B fuel assembly cores.

The accident analyses performed for the Safety Analyses Report for any 177 FA core includes a study of the sensitivity of results to a delay in control rod insertion. This is normally performed for the overheating transients of Rod Group Withdrawal at HZP and Rated Power Conditions. These accidents can result in overpower and overpressure conditions in the primary system. Figure 14 from the TMI-1 FSAR shows the sensitivity results for trip delay times for the initial core. RCS peak pressures increase only 20 psi up to about 0.65 sec delay compared to the nominal 0.3 sec delay. Also, these values are conservative since current rod group worths and Doppler and moderator coefficients are enveloped by the FSAR analysis values.

Accounting for weight, buoyancy, and CRDM frictional forces, the net downward force of a CRA (with attached leadscrew) is 130 lbs. The 0.36 sec minimum conservatism in the 1.66 sec drop time can be interpreted as due to an increased drag force equivalent to 50 lbs. Taking credit for the increased trip delay of 0.35 sec (above the 0.3 sec nominal) as an increased drop time above the 1.66 sec analysis value converts to a total drag force of 75 lbs. According to B&W, studies have shown that delay times up to 1.0 sec result in peak pressure increases of only 40 psi. This translates into a total drag force of about 90 lbs. Given the small size of fragments likely to reach the CR guide and enter the gaps interference forces of this magnitude are very unlikely to occur.

The safety analyses were done for a full rod group. The effects of a single delayed CRA would be much smaller. Therefore, any postulated small increase of drag force on a CRA is not a safety concern.

6.5.3 Jammed CRA

In the extreme case, loose pieces may wedge against a CRA or leadscrew to the extent of total jamming thus preventing insertion. Safety analyses are performed to assure sufficient CRA negative reactivity is available for adequate core shutdown margin of at least 1.0% $\Delta K/K$. The analyses include the assumption that the maximum worth rod is stuck out of the core (Ref. 17). Thus a jammed CRA is not a new safety concern. A jammed APSRA is also not a safety issue since they do not scram and are not taken credit for in the shutdown analysis.

As a further precaution Tech Specs require control elements to be exercised every two weeks to verify movement of each element.

6.6 INCORE INSTRUMENTATION

There is a small potential for plugs or fragments to enter the 4" diameter opening at the bottom of the incore instrument guide tube assembly (GTA) (Fig. 4). There is about a 1" annular gap between the GTA and the top of the incore nozzle. Entrance is unlikely since flow in the GTA is lower than in the RV plenum and entrained pieces would follow the stronger flow paths.

If a piece entered, it might wedge between the narrowing GTA ID and the Inconel sheath of the incore string. Minimum diametral gap is about 0.33". Another possible wedging location for a small fragment is between the string and the incore nozzle ID (0.25" diametral gap). This is even less likely since there is no flow into the nozzle and guide piping. A wedged fragment could cause jamming of the string. This would be a problem only during refueling when the incores are withdrawn.

In a remote case wedging could cause fretting damage to the sheath. As a worst-case, the string might be partially or totally severed causing functional loss of the detectors and exposing the inner air spaces in the detector annulus and central tube to RCS pressure (Fig. 15). Detector loss would be monitored by the plant process computer and appropriate actions taken per Tech Specs in the unlikely event that such loss violated the minimum requirements for number and arrangement of detectors. Tech Specs and plant procedures also require functional testing of the neutron detectors once a month.

Exposure of the air spaces changes the RCS pressure boundary. This would be controlled by the high pressure closure seals that cap the detector lead wires and the central air tube. As described in

Reference 18, the strings and closures were specifically designed for potential use as core pressure measurement devices. Also, a severed string will remain within the guide structures and not interfere with other strings or components. Thus, even a severed incore string is not a safety concern.

6.7 VENT VALVES

The eight vent valves are located in the annulus above the inlet nozzles. The function of the valve is to prevent a pressure imbalance after an inlet pipe rupture and thus to enhance core cooling by allowing steam flow directly to the break. The valve design was examined to determine whether a loose plug could effect this function.

It is unlikely that any plugs entrained in the inlet flow will reach the valve elevation. There are deflectors on the core support shield that direct inlet flow downward into the downcomer. Thus, any flow in the valve region is low velocity and turbulent. The valve body and hinge components are massive relative to the plugs. Impact effects would be minimal. A dimensional study in the TMI-1 FSAR (Section 3) shows that the rotational clearances within the hinge are less than 0.03"; too small for a plug to enter. There is a larger gap (about 3/8") between the hinge outer journal and the valve body in which a deformed plug might wedge resulting in restricted valve motion. The probability of this occurring is extremely small given the flow conditions in the valve region and the small plug population. Even in the postulated case of a vent valve sticking closed, the effect on peak clad temperature during a LOCA has been demonstrated to be minimal as shown in Figure 16 taken from the TMI-1 FSAR.

Also, Tech Spec surveillance requirements include visual inspection and testing of vent valve movement during each refueling outage.

6.8 RCS CONNECTIONS

Based on flow conditions during normal operation any loose plugs and fragments would likely be carried past the various orifices for connections to the RCS. Also, the estimated low impact energies of the plugs are not expected to cause significant damage to any instruments protruding into the flow paths. Nevertheless, studies were performed to determine whether effects due to plugs could raise safety concerns related to these connecting systems components. Results are tabulated in Table 2. Any of the potential damage effects identified are considered to be of low to extremely low likelihood of occurrence. Therefore, loose plugs effects on RCS connections do not raise any safety issues not covered by existing analyses, limits, procedures, or system redundancies.

One connection to which particular attention was paid is the decay heat removal dropline. As discussed above, it is unlikely that a plug will enter the dropline at all during normal operation since it would likely be entrained in the hotleg flow and carried past the static dropline. As described in Table 2, damage to the DH system when in use is also unlikely and controllable by redundancy of valves and pumps. In the very unlikely event that both DH pumps failed, heat removal by OTSG steaming is available. Also, during a LOCA the dropline from the hotleg is not used. In the LPI injection mode the DH pump takes suction from the BWST. Thus, even if a plug had reached the first isolation valve it would not be transported to a DH pump. If recirculation from the RB sump were necessary, sump screens would prevent a plug that had escaped to the sump from being carried to a DH pump. Therefore, loose plugs do not create a safety concern for DH system functions.

7.0 CONCLUSIONS

Based on the evaluations described above it is concluded that operation of TMI-1 with six loose OTSG plugs represents a very small operational risk and does not raise any new safety concerns.

The evaluations examined potential effects of plug impact, wedging, wear, and flow blockage. Results demonstrated that likely effects on all fuel, control rod, and other RCS components and connecting systems will be limited to minor impacts and flow blockage. Even unlikely effects were determined to create no significant operating problems or new safety concerns.

These conclusions would not change for loose plug populations an order of magnitude larger than six. OTSG plug repairs will assure that an increase in the number of loose plugs is very unlikely.

Specifically, the plugs cannot attain impact energies large enough to damage pressure boundaries, fuel, instrumentation, or other RV and RCS components. The small size and population of plugs cannot cause flow blockages that would degrade DNBR or fuel assembly holddown margins. Nor is there any likelihood of an intact plug affecting control rod movement or causing any new safety concerns regarding the functions of connecting systems to the RCS. Small fragments that could cause fuel rod fretting or restrict control rod motion are unlikely to form due to the high strength, ductility, and toughness of the Inconel 600 plug material. In the very unlikely event of fuel damage or control rod interference existing plant radiation monitoring limits and safety analyses assure safe shutdown.

In summary, no credible effects due to the loose plugs were identified that could place the plant in a configuration that would exceed existing safety analyses envelopes or prevent safe reactor shutdown and core heat removal capability. The likelihood of any significant degradation to the proper functioning of the fuel and all other primary system components is extremely low. No Tech Spec safety margins have been reduced. The plugs do not introduce any significant increase in the probability of occurrence of events previously analyzed in the safety analyses nor any increase in the consequences of those events. No potential for an accident or malfunction different than those previously analyzed has been created. Thus no unreviewed safety questions have been created. These judgements are based on:

- the small size and mass of the plugs,
- the small loose plug population
- the expected randomness of plug flow paths in the RV
- the minor consequences of all credible effects

It is further concluded that TMI-1 can be operated safely with a small population of loose OTSG plugs in the RCS with reasonable assurance of no significant increases of undue risk to the health and safety of the public or plant personnel.

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16. NRC Letter, Docket No. 50-289, J. F. Stolz (NRC) to H. D. Hukill, 4/20/81.
17. TMI-1 Cycle 5 Reload Report, December 1978.
18. B&W Topical Report, BAW-10123, "Nuclear Applications Software Package", February 1978.

TABLE 1

RV/RCS LOOSE PLUG EFFECTS

COMPONENT	POTENTIAL EFFECT	LIKELIHOOD OF OCCURRENCE	REPORT SECTION
o Primary Pressure Boundary	. Impact Breach	None	6.1
	. Wedging Breach	None	6.1
o RC Pumps	. Impeller Jamming	None	6.2
	. Minor Impact Damage	Low	6.2
	. Significant Impact Damage	Extremely Low	6.2
	. Seal Wear	Extremely Low	6.2
o RV Internals	. Minor Impact Damage	Medium	6.3
	. Significant Impact Damage	Extremely Low	6.3
	. Wear Damage	Very Low	6.3
o Fuel Assemblies - Lower End Fitting	. Minor Impact Damage	Medium	6.4.1
	. Significant Impact Damage	Very Low	6.4.1
	. Minor Wear Damage	Low	6.4.2
	. Major Wear Damage	Very Low	6.4.2
- Fuel Rod	. Fretting Damage	Very Low	6.4.3
- FA Blockage	. Minor Flow Reduction	High	6.4.4
	. Major Flow Reduction	None	6.4.4
	. FA Lift	None	6.4.4.1
	. Reduced Fuel Cooling	Very Low	6.4.4.2
	. Local Fuel Hot Spots	Very Low	6.4.4.3
	. Effect on LOCA	None	6.4.4.4
- Upper End Fitting	. Impact Damage	Extremely Low	6.4.5.1
	. Wear Damage	Extremely Low	6.4.5.1
- FA Holddown Springs	. Wedging/Fatigue Failure	Extremely Low	6.4.5.2
o Control Rods	. Impact Damage	Extremely Low	6.5.1
	. Delayed CRA Insertion	Extremely Low	6.5.2
	. Jammed CRA	Extremely Low	6.5.3
o Incore Instrumentation	. Wedging	Low	6.6
	. Fretting Damage	Very Low	6.6
o Vent Valves	. Impact Damage	Extremely Low	6.7
	. Jamming Closed	Extremely Low	6.7

TABLE 2

RCS CONNECTIONS AND PLUG TRANSPORT EFFECTS

Connection/ Component	Orifice Diameter	Potential Effects of Plug Transport	Likelihood of Occurrence	Operational/Safety Concerns
• RCS HOTLEG				
- Pressure Taps	1"	Loss of a pressure tap due to blockage	Very Low	None - Each hotleg has redundant pressure taps.
- RTDs	---	Loss of indication due to impact	Low	None - Each hotleg has redundant temperature detectors.
- Flow Meters	---	Loss of indication due to impact	Very Low	None - Internal Elements are of steel construction 1" thick.
		Blockage of flow through flow tube	Extremely Low	None - Throat diameter is 34.7".
		Loss of indication due to blockage of pressure tap	Very Low	None - Each hotleg has 4 dynamic pressure taps and 4 static pressure taps. Blockage of dynamic tap would give a low flow indication to RPS. Two such signals are required to initiate a reactor trip. Blockage of a pressure tap would not interfere with the normal reactor protection logic.
- Decay Heat Dropline	12"	Blockage of valves DH-V1, 2, 3	Low	May cause a gate valve to remain partly open - not a concern since 3 valves are in series and isolation capability is retained.
		Impact on DH pumps	Low	Probability of damage to DH pumps is small due to low fluid velocity and low pump speed. In the event of pump damage, redundant pump is available.
- PZR Surge Line	10"	None	Very Low	None - plug may pass through surge line and settle in bottom of PZR.

TABLE 2 (Cont'd)

RCS CONNECTIONS AND PLUG TRANSPORT EFFECTS - page 2

Connection/ Component	Orifice Diameter	Potential Effects of Plug Transport	Likelihood of Occurrence	Operational/Safety Concerns
• RCS COLD LEG				
- RTDs	---	Loss of indication due to impact	Low	None - Each cold leg has redundant temperature detectors.
- PZR Spray Line	2.5	Blockage of RC-V1	Very Low	May cause valve to remain partly open resulting in increased spray flow. This could cause some depressurization of the RCS which can be compensated for by the Control Room operators.
- PZR Spray Bypass	0.75"	Blockage of bypass flow	Extremely low	None - Spray line temperature is monitored in Control Room.
- Core Flood Inlet	10"	None - flow would push plug into RPV	Very Low	None
- HPI/Makeup Conn.	2.5"	Partial blockage of HPI nozzle	Very Low	None - If HPI is initiated, plug will be flushed into RCS cold leg.
- Letdown Conn.	2.5"	Blockage of globe valve leading to reduction in letdown flow.	Very Low	None
• HIGH POINT VENTS				
- RPV Vent	0.466"	Blockage of vent path	Extremely Low	None - Alternate vent paths available.
- Hot Leg Vents	1"	Blockage of vent path	Extremely Low	None - Alternate vent paths available.
- PZR Vent	1"	Blockage of vent path	Extremely Low	None - Alternate vent paths available.
• CRDMs	---	Interference with lead screw resulting in delay in rod insertion or failure of rod to drop	Extremely Low	FSAR Analysis has considered one stuck control rod assembly (of maximum worth) and concluded that adequate shutdown martin is assured (See Sect. 6.5.)

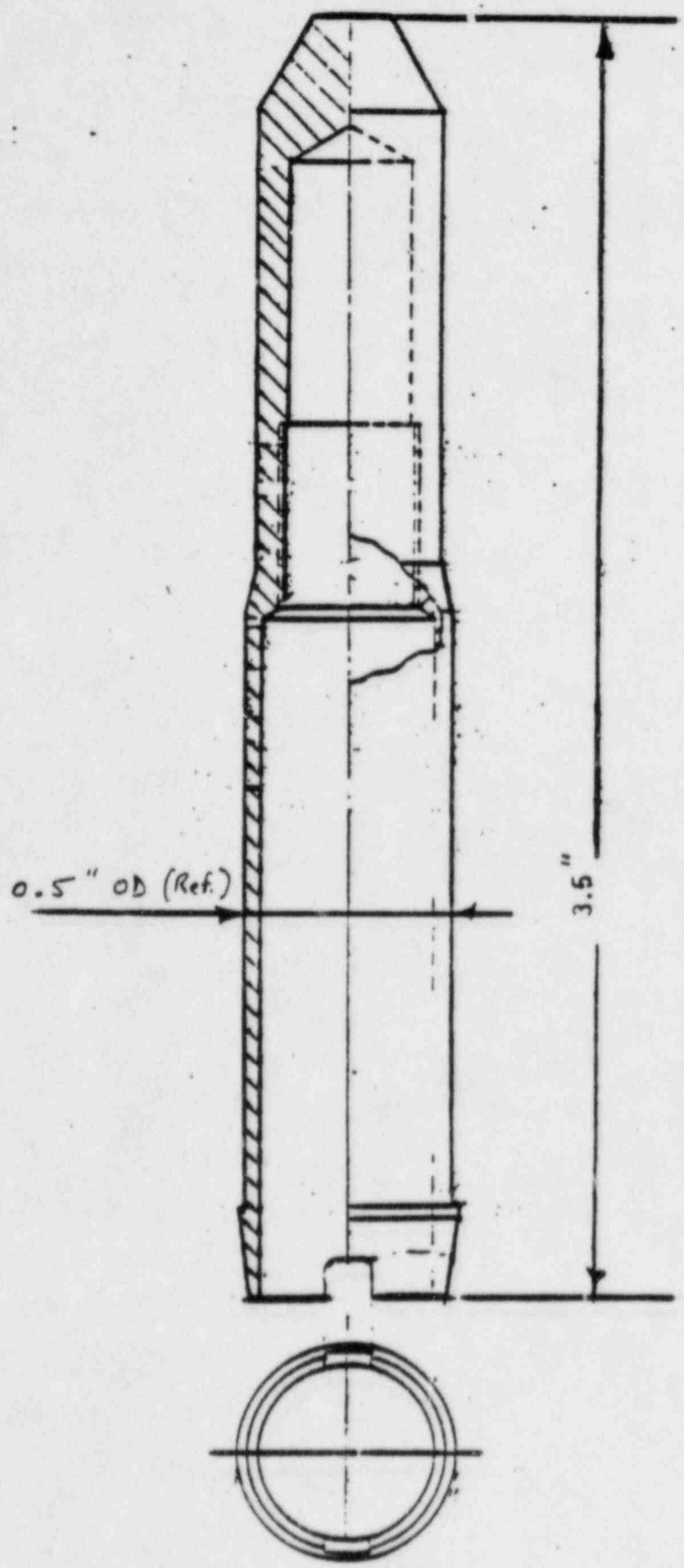
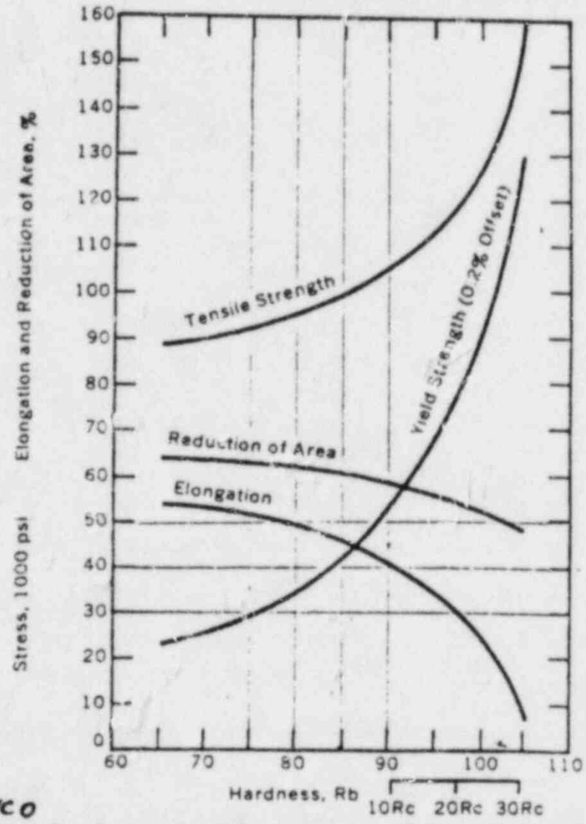


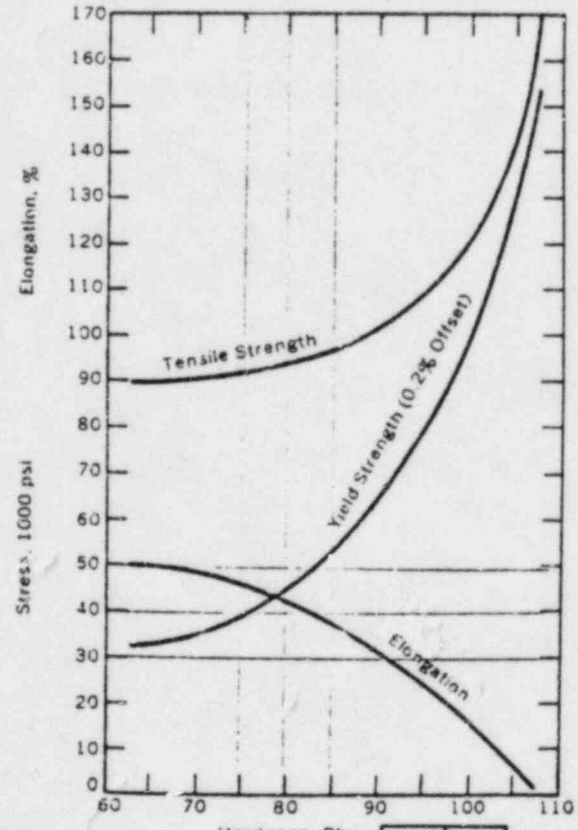
Figure 1

TMI-1 STEAM GENERATOR ROLLED PLUG - MODEL II



INCO

Figure 2 – Average tensile propertier of hot-rolled and cold-drawn rod.



INCO

Figure 3 – Average tensile properties of sheet and strip.

Figure 2A

Elongation of Inconel 600

[Inco Handbook T-7, Engineering Properties of Inconel alloy 600]

INCO

Table 14 — Room-Temperature Impact Strength of Rod

Condition	Impact Strength, ft-lb	
	Izod Notch	Charpy U-Notch
Cold-Drawn	>120	230
Cold-Drawn, Annealed	70-100	151
Hot-Rolled	>120	230
Hot-Rolled, Annealed	100-120	—

INCO

Table 15 — Impact Strength (Charpy V-Notch) at Elevated Temperatures

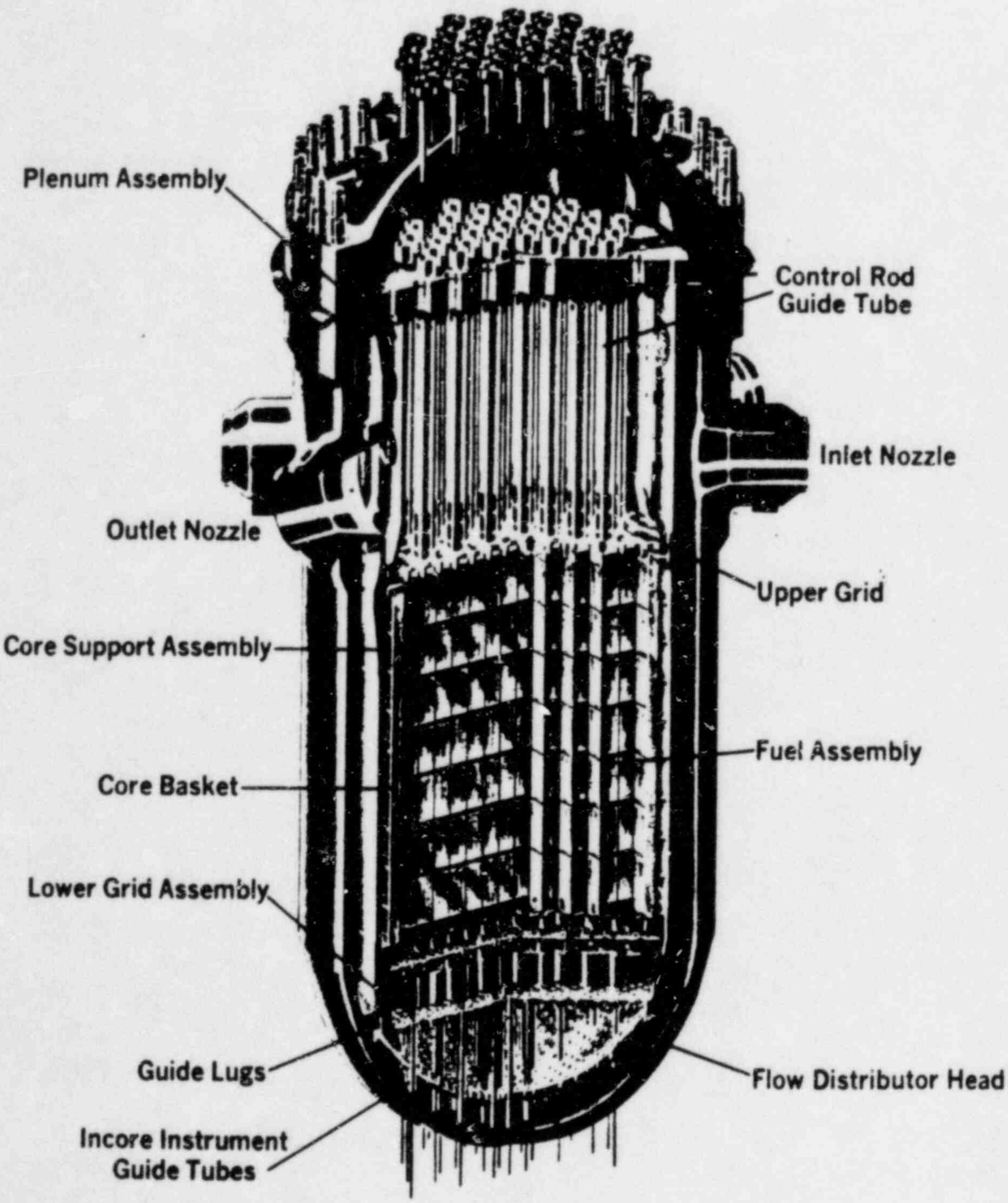
Temperature, °F	Impact Strength, ft-lb	
	Annealed	Cold-Drawn
70	180	114
800	187	84
1000	160	86
1200	160	104
1400	154	163

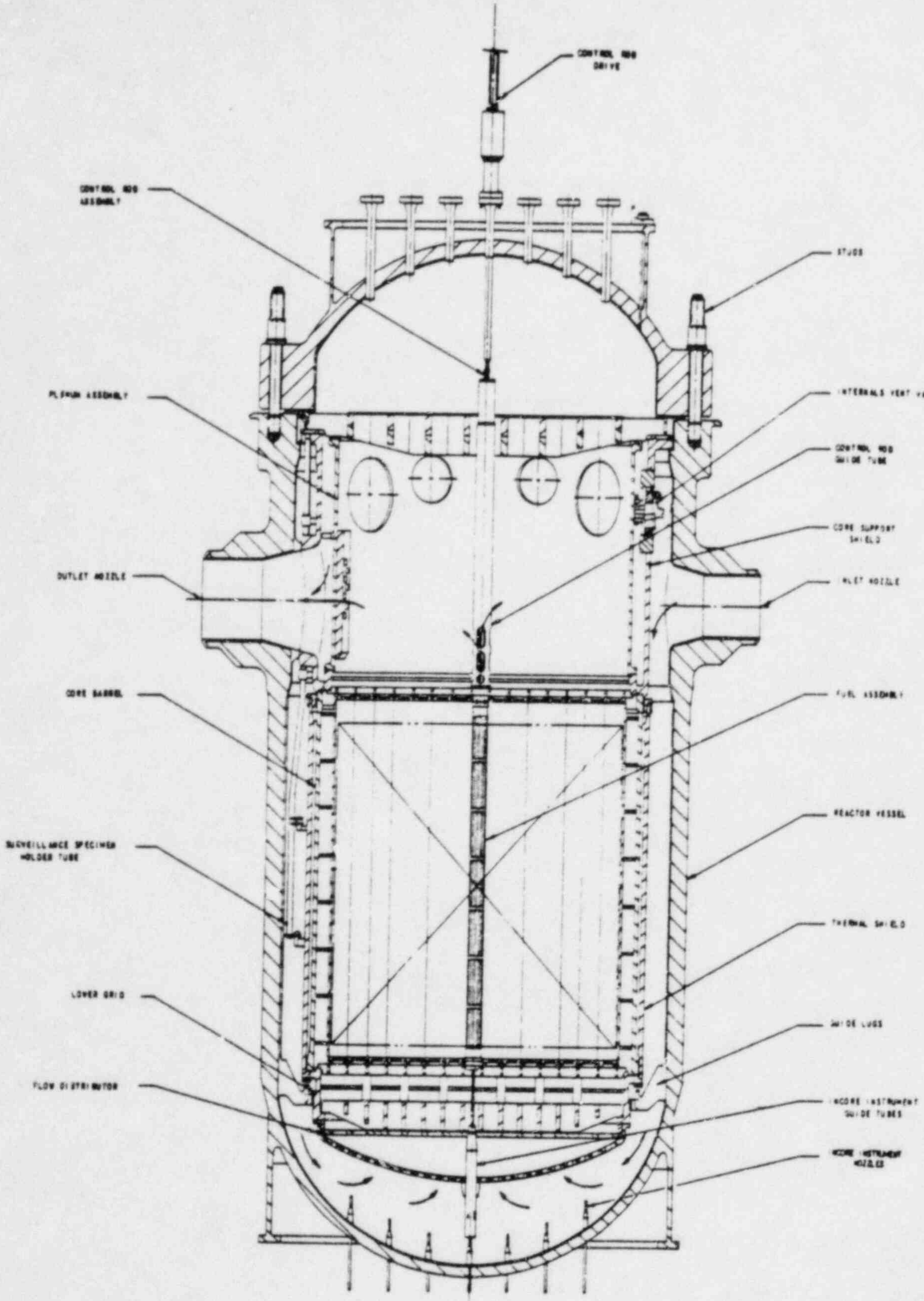
Figure 2B

Impact Strength of Inconel 600

[Inco Handbook, T-7
Engineering Properties of
Inconel Alloy 600]

FIGURE -3 REACTOR CORE





(GTA)

Figure 4

REACTOR VESSEL AND INTERNALS GENERAL ARRANGEMENT

THREE MILE ISLAND NUCLEAR STATION UNIT 1

Met Ed

JCP&L

FIGURE 3-46

(AM. 34 11-22-72)

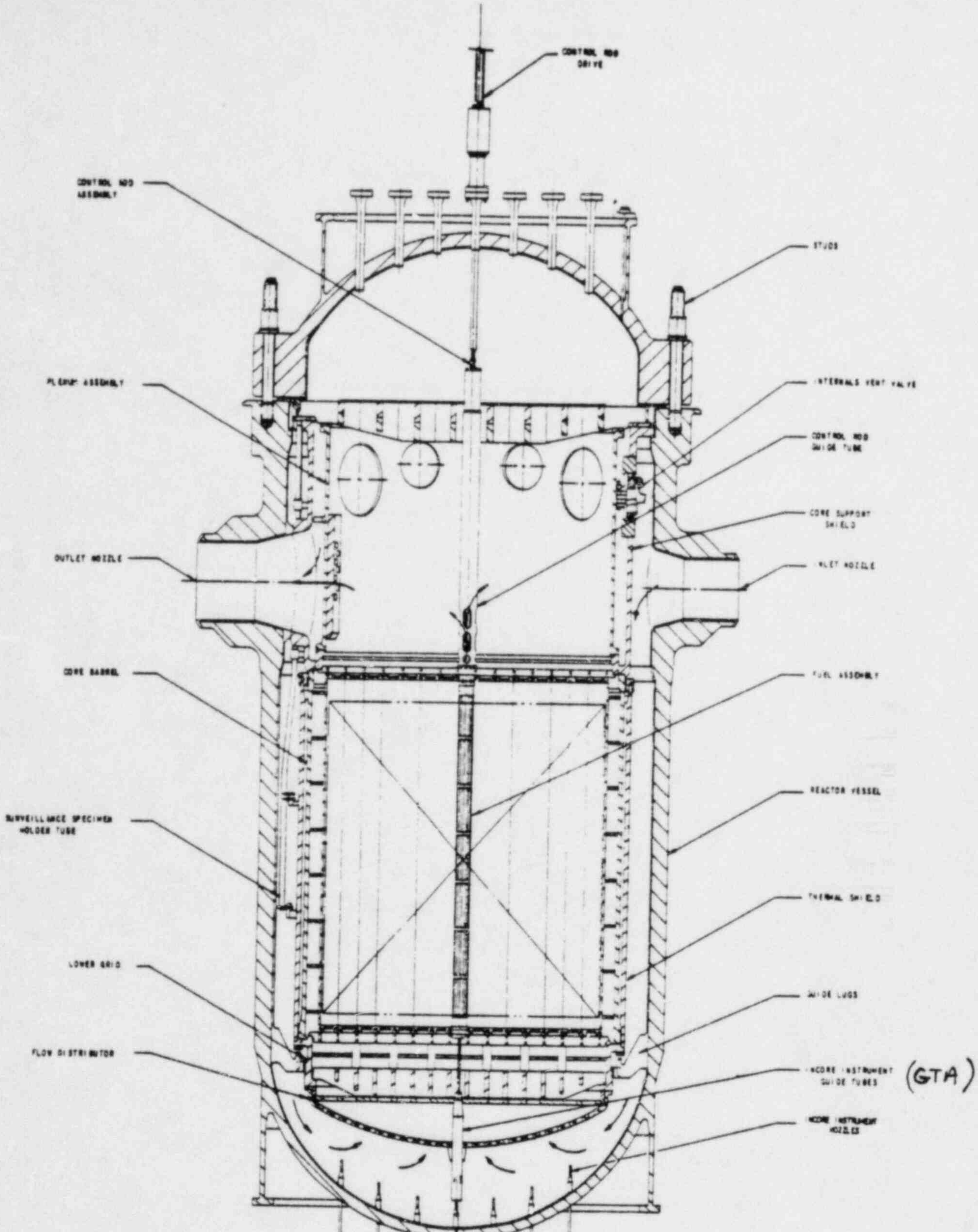


Figure 4

REACTOR VESSEL AND INTERNALS
GENERAL ARRANGEMENT

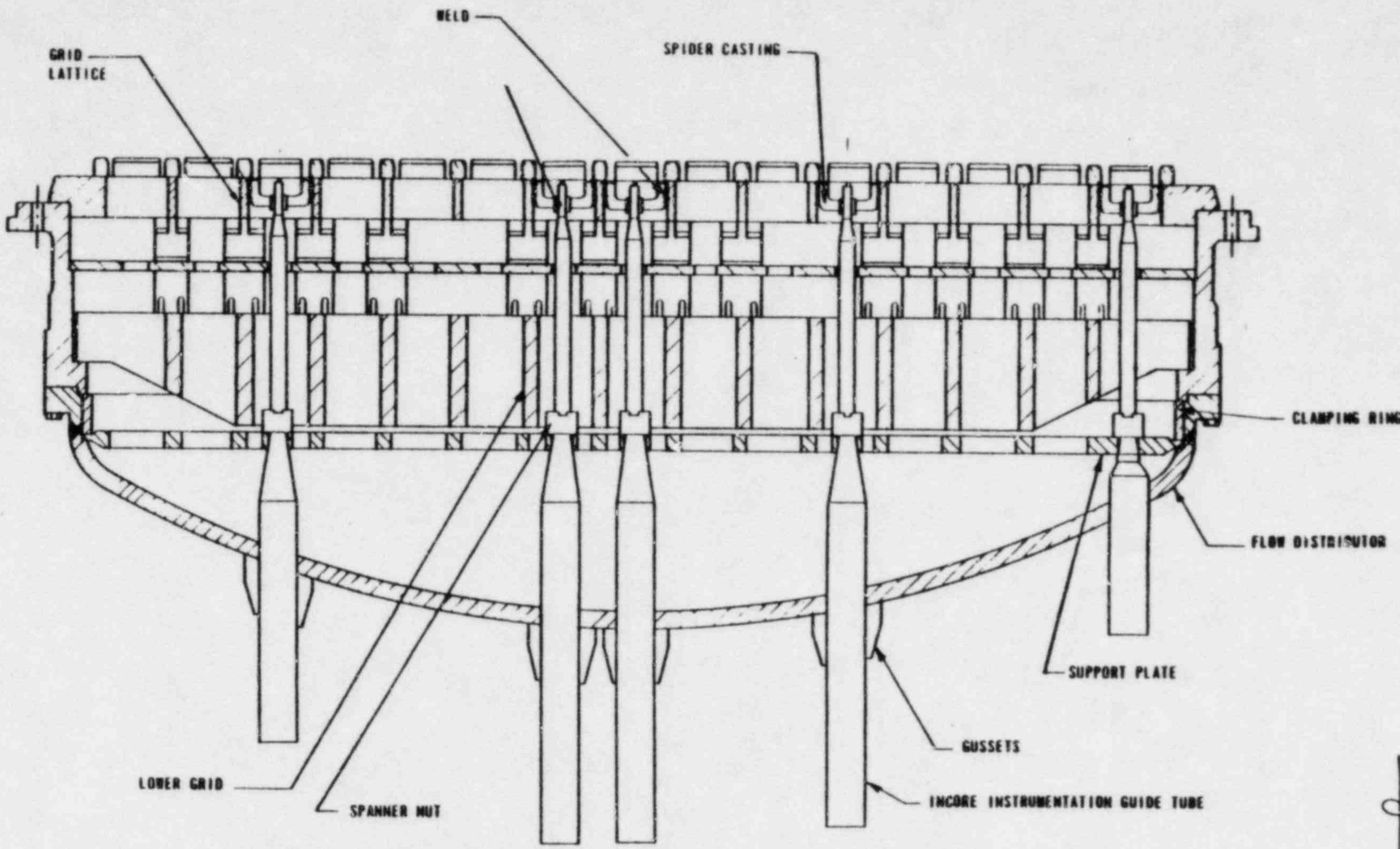
THREE MILE ISLAND NUCLEAR STATION UNIT 1

Met-Ed

FIGURE 3-46
(AM. 34 11-22-72)

Figure 5

Flow Distributor and Lower Grid Assembly



Babcock & Wilcox

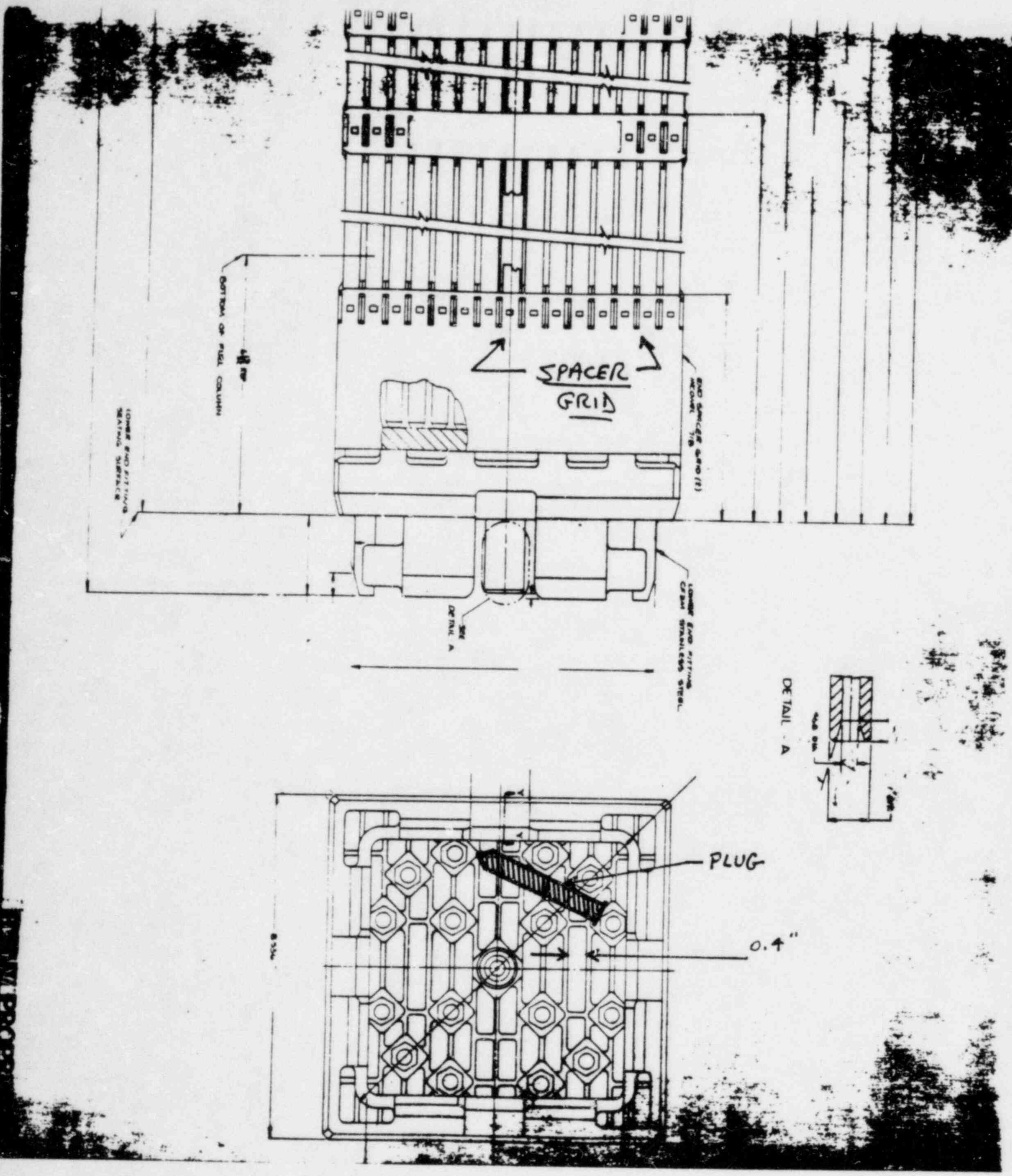
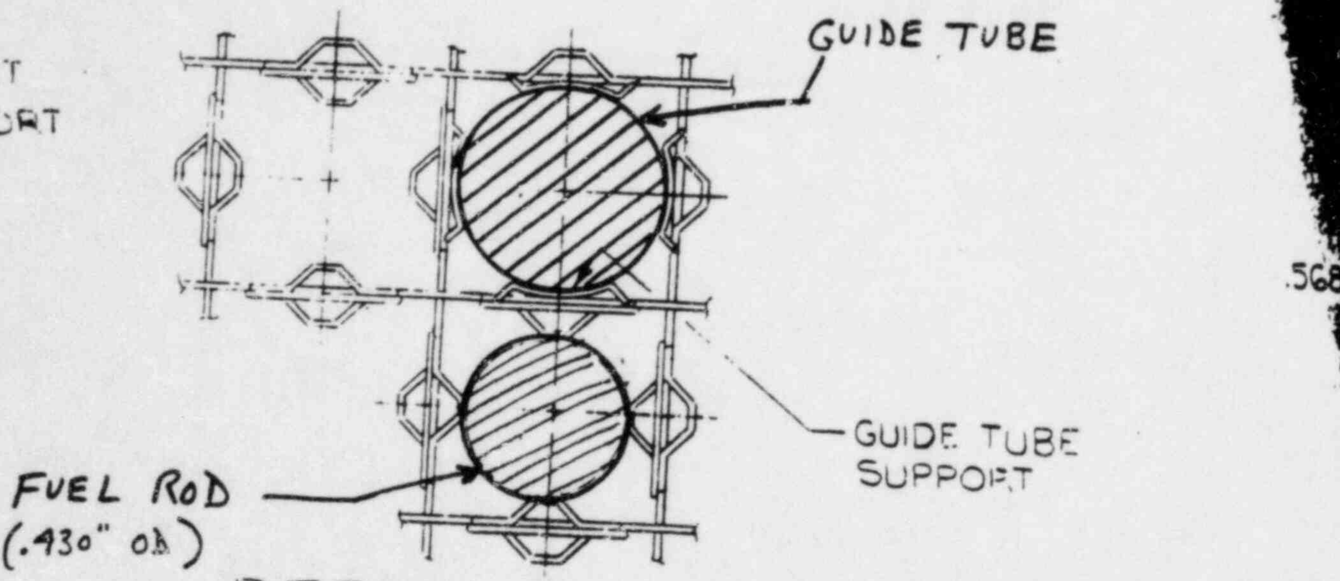


Figure 6

Mark B Fuel Assembly
Lower End Fitting

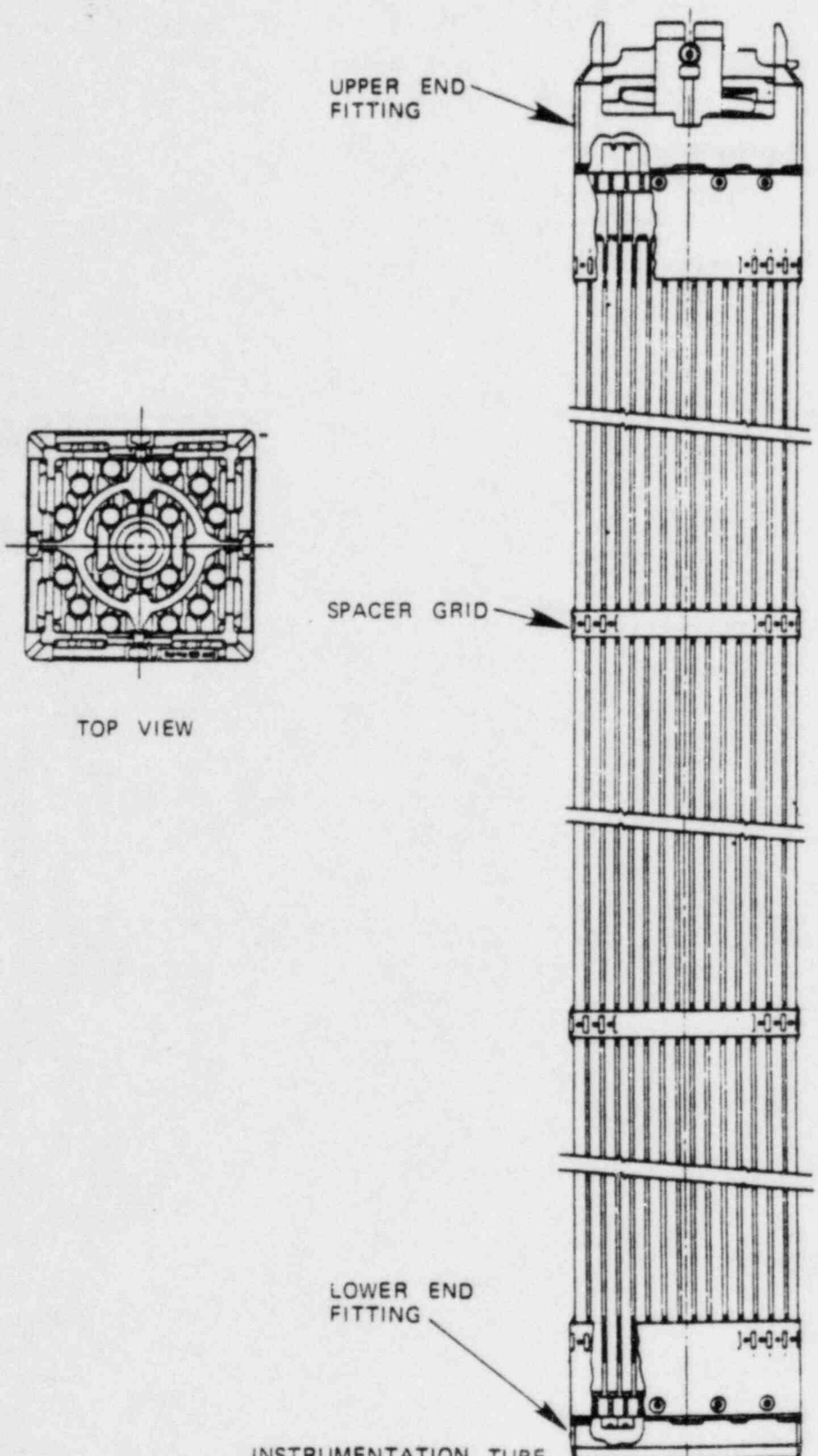
FUEL ROD ASSEMBLY
SCALE : 24" = 1'-0"



DETAIL D
SCALE : 24 = 1'-0"

FIG. 7 - FUEL ASSEMBLY SPACER GRID CELLS

FIGURE 8 FUEL ASSEMBLY



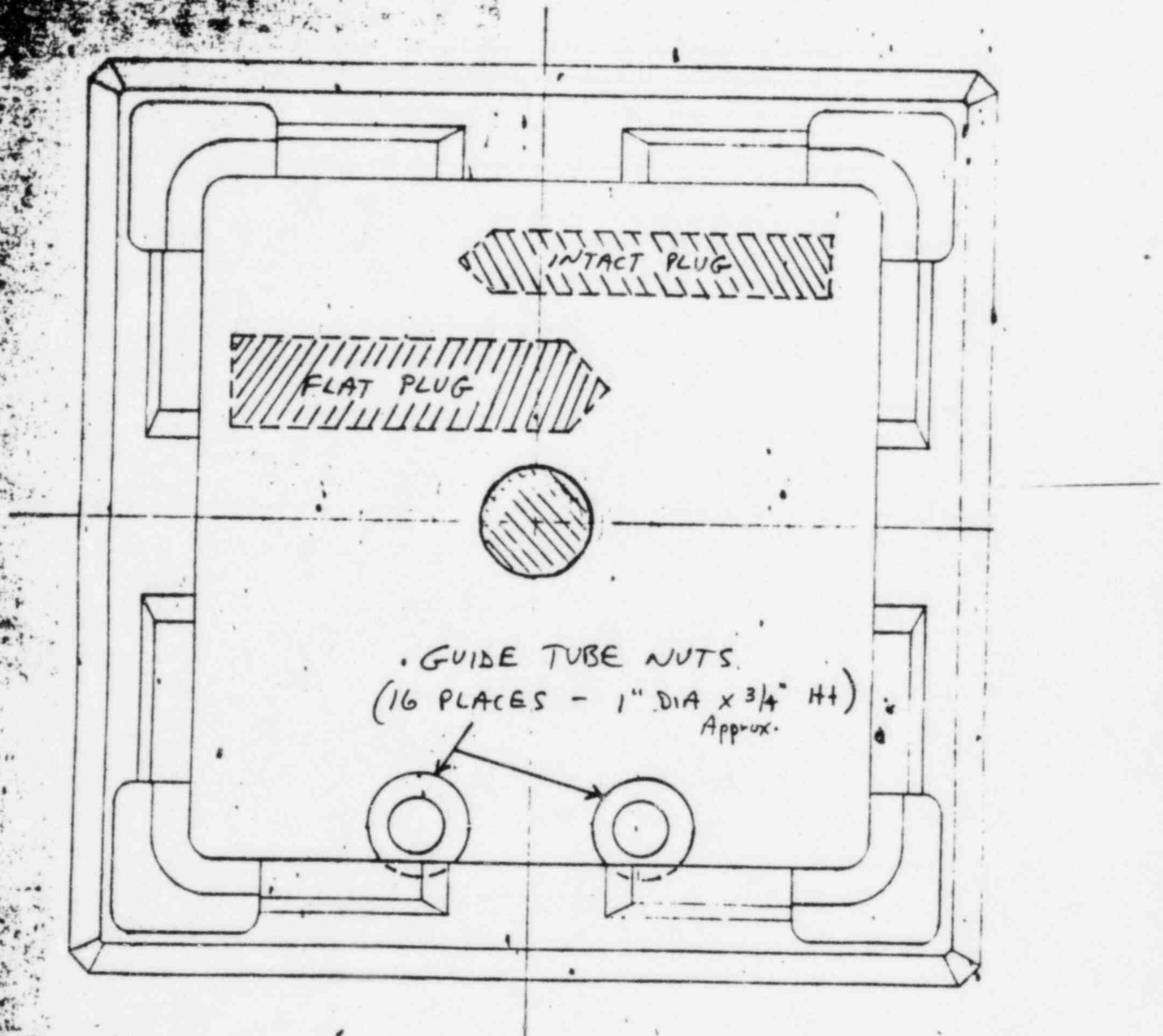

INTACT PLUG
FLAT PLUG
GUIDE TUBE NUTS
(16 PLACES - 1" DIA x 3/4" Ht)
Approx.

Figure 9

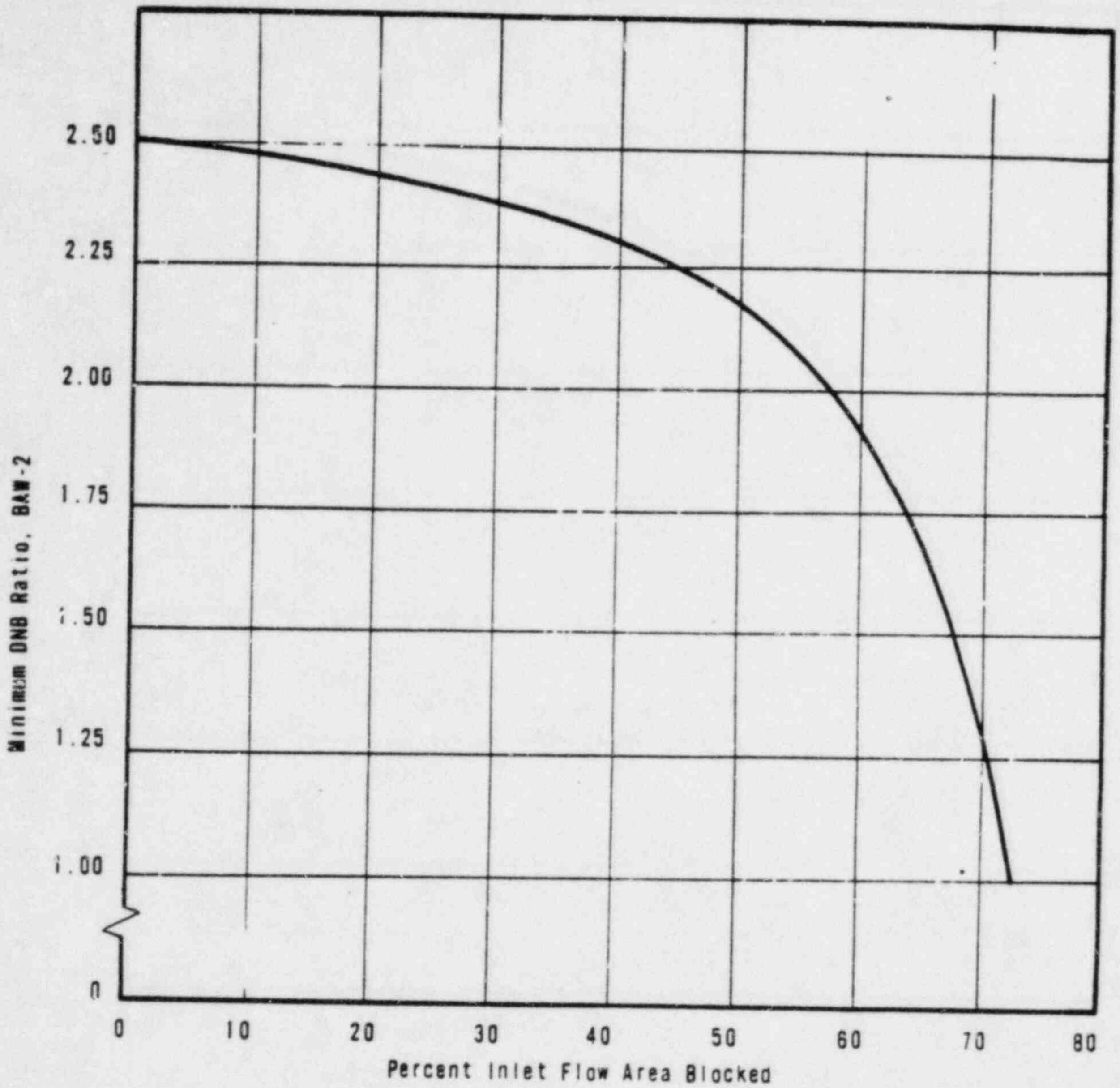


Figure 10

CONSUMERS POWER COMPANY
MIDLAND PLANT UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Minimum DNB Ratio Versus
Percent Inlet Flow Area Blocked

FSAR Figure 4.2-19

FIGURE 11 CONTROL ROD ASSEMBLY

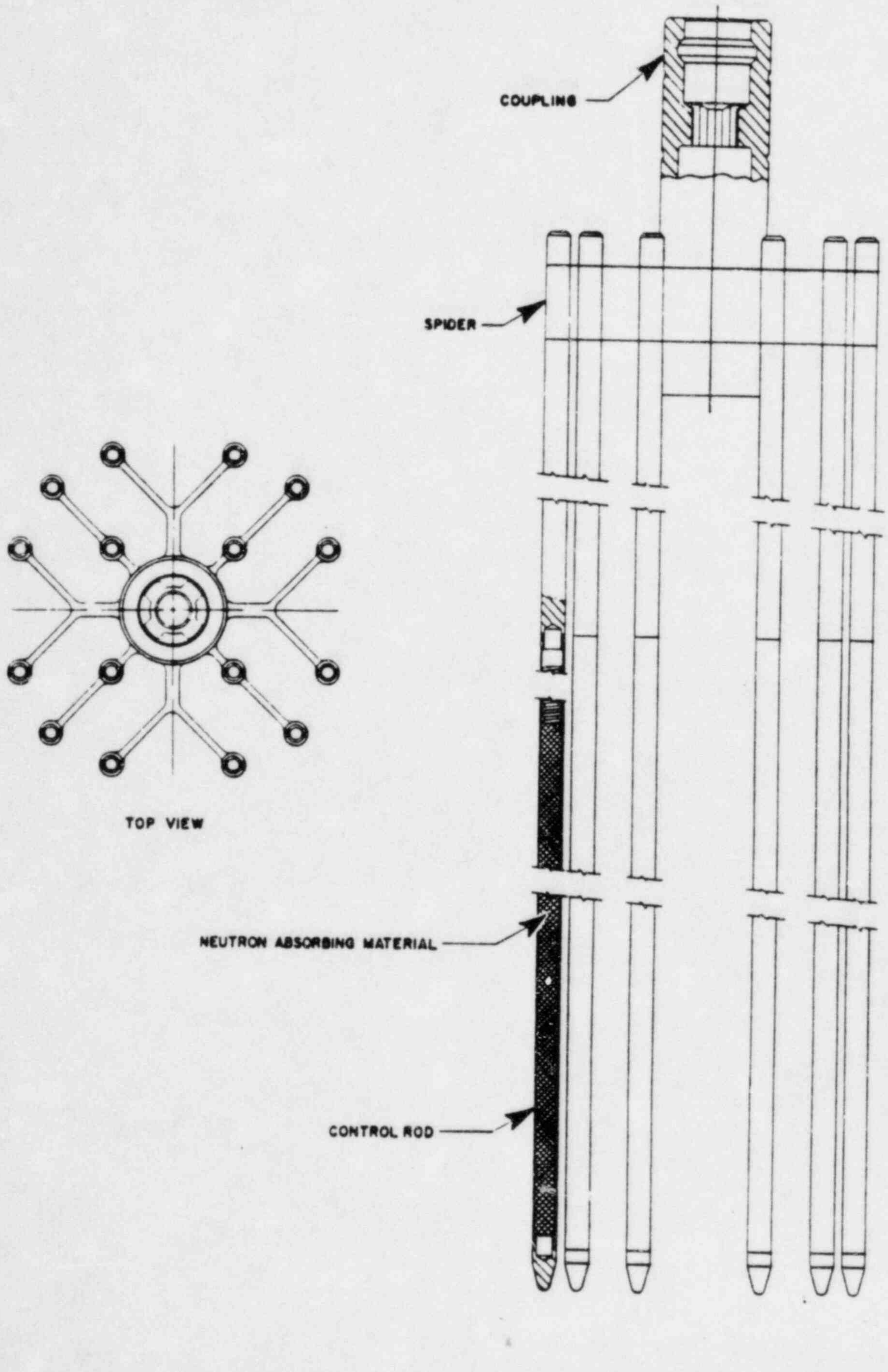


Figure 4-5
Upper Guide Tube Structure
Flow Paths

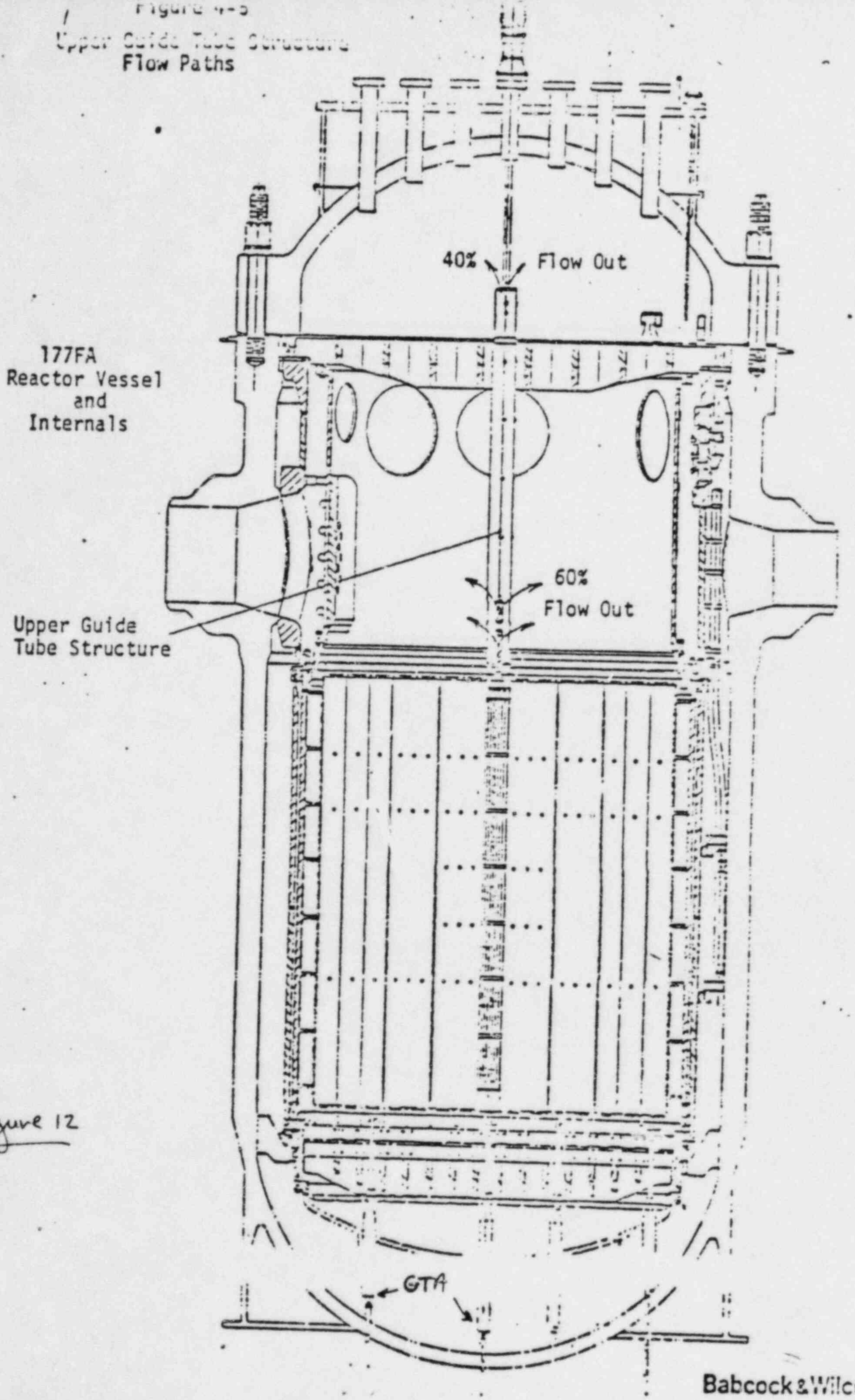


Figure 12

Babcock & Wilcox

Cross Sectional View of
Column Weldment

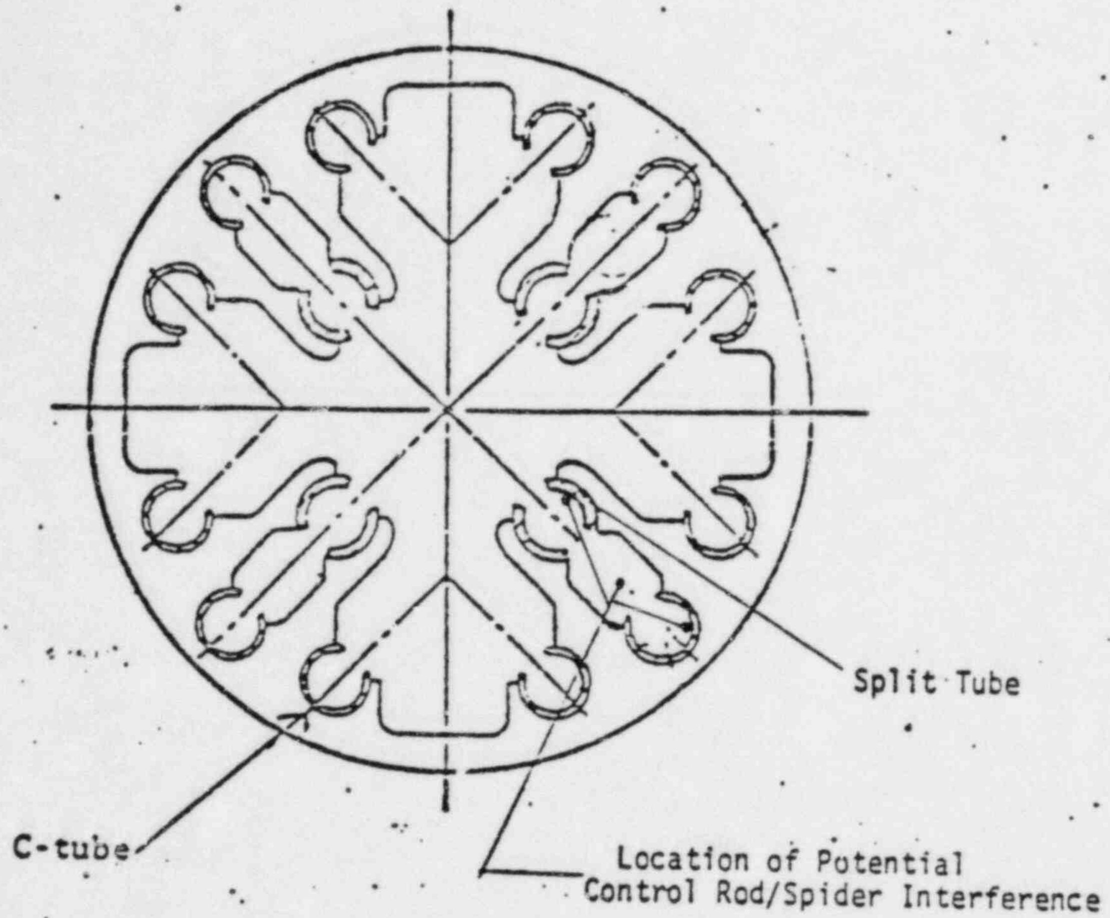


Figure 13

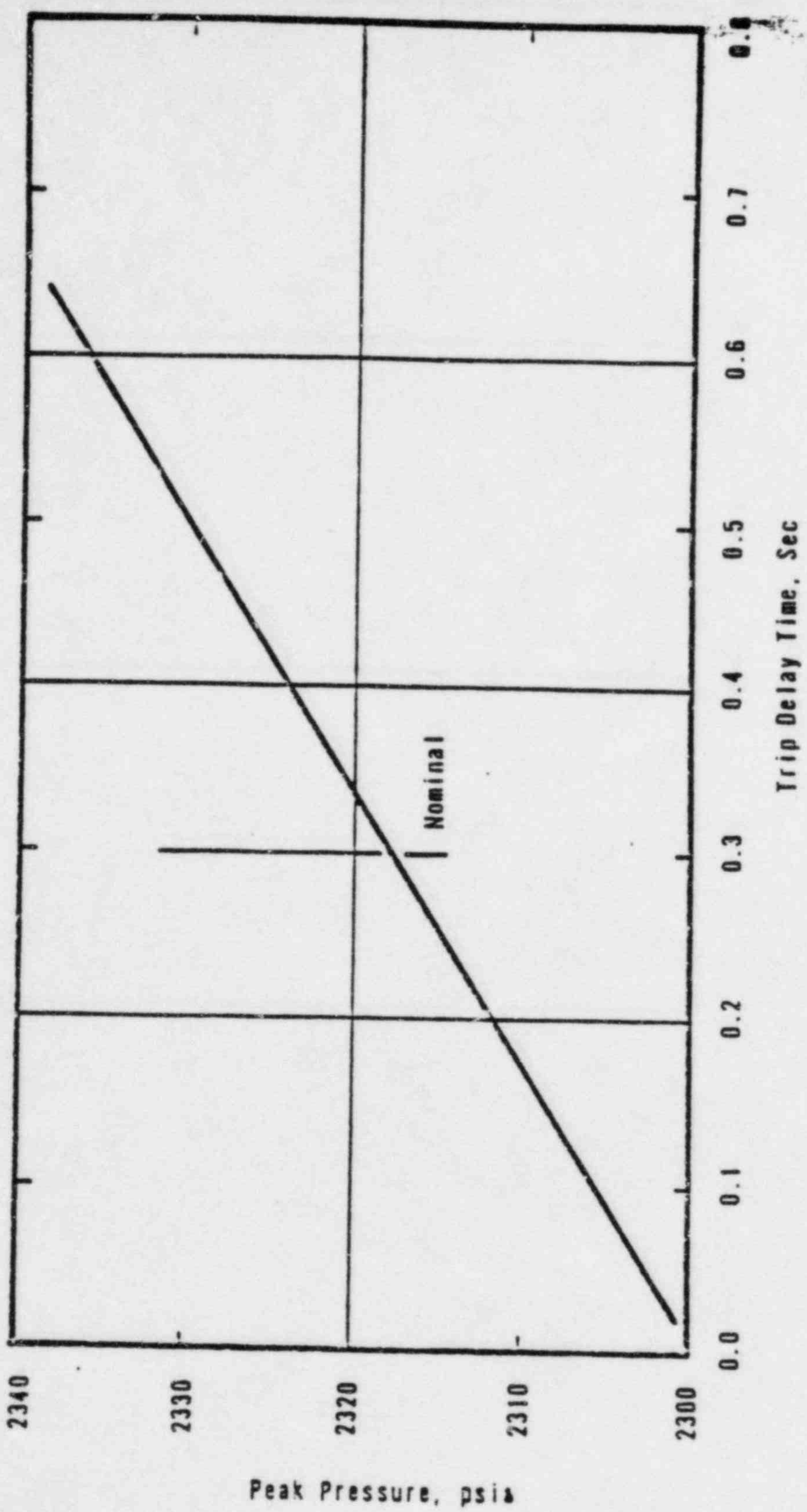


Figure 14

PEAK PRESSURE VS TRIP DELAY TIME FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5% $\Delta k/k$ ROD GROUP

THREE MILE ISLAND NUCLEAR STATION UNIT 1

Incore Instrument String Cross Section

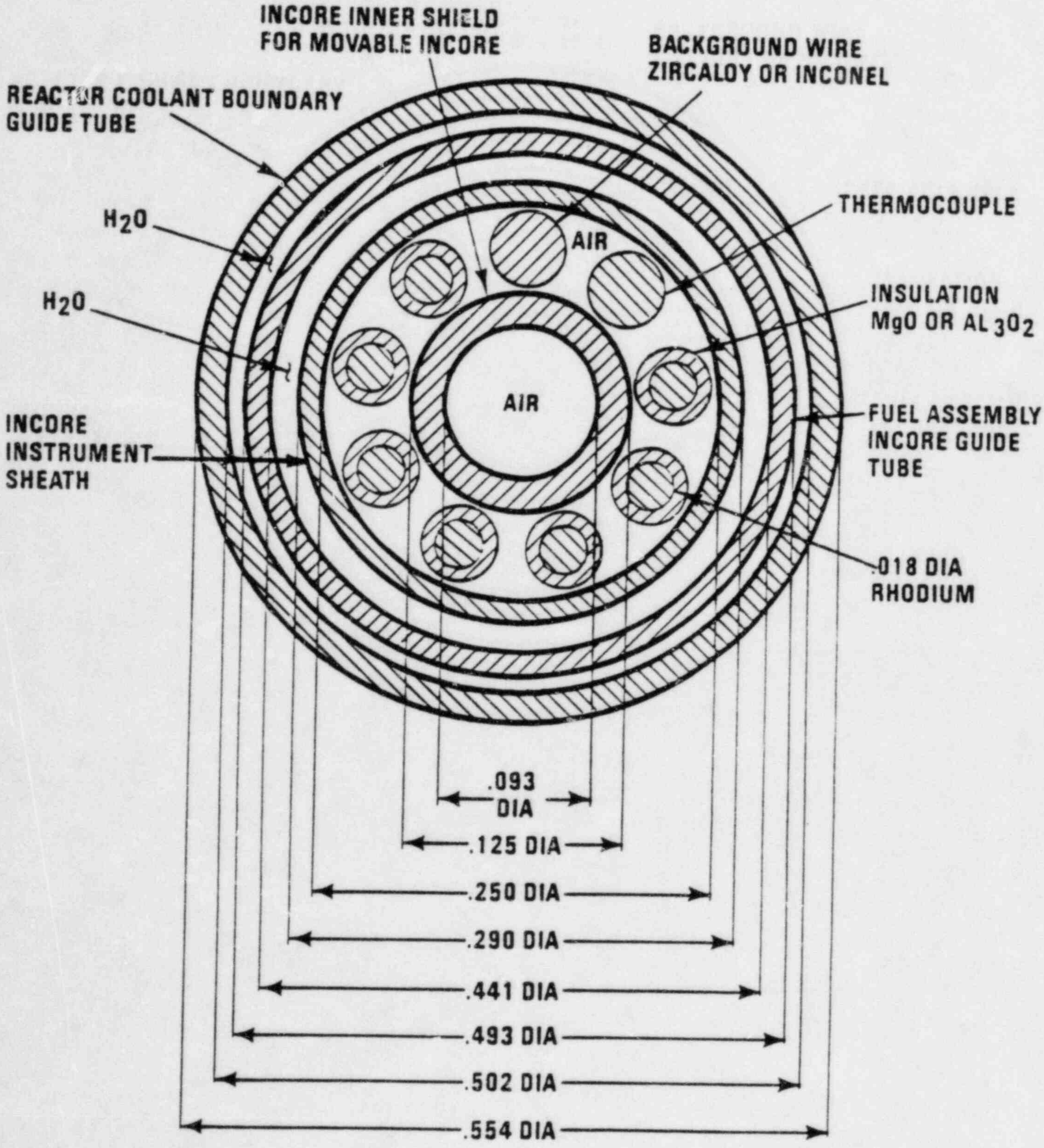


Figure 15

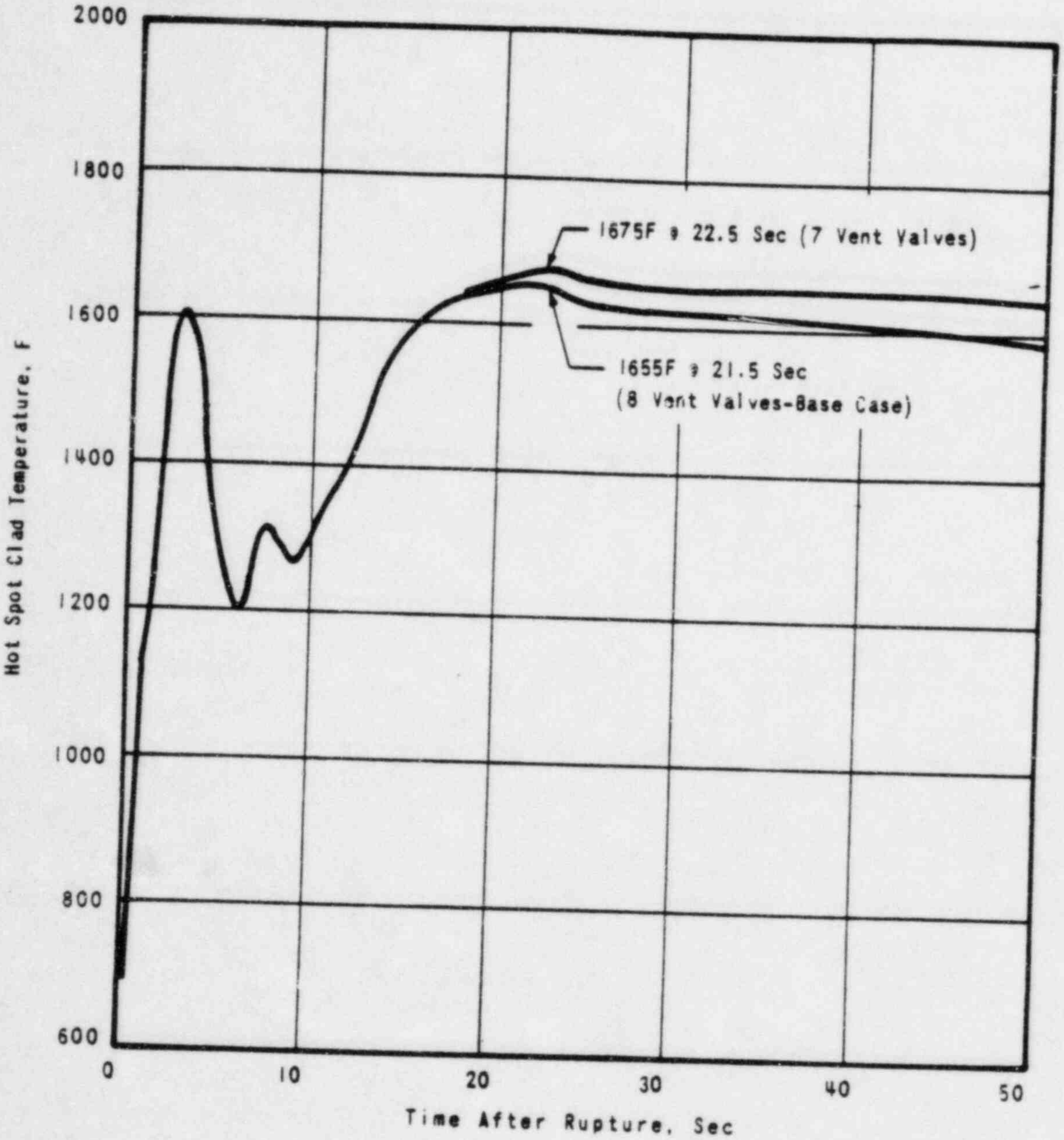


Figure 16

HOT SPOT CLAD TEMPERATURE VS TIME AND NUMBER OF INTERNALS VENT VALVES FOR A 28-IN. ID, DOUBLE-ENDED, COLD LEG RIPE RUPTURE

THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 14-47