ATTACHMENT 2

TO P-93045

PROPOSED CHANGE

4.0 DESIGN FEATURES

4.1 Site

The Fort St. Vrain Nuclear Generating Station is located approximately 35 miles north of Denver and 3.5 miles northwest of the town of Platteville, in Weld County, Colorado.

The site consists of 2798 acres. The EXCLUSION AREA BOUNDARY encloses the decommissioning Emergency Planning Zone, as shown on Figure 4-1.

Points where radioactive gaseous and liquid effluents are released are shown on Figure 4-1.

4.2 Reactor Building

The Reactor Building houses the prestressed concrete reactor vessel (PCRV), fuel handling area, fuel storage wells, fuel shipment preparation facilities, decontamination and radioactive liquid and gas waste processing equipment, and most reactor plant process and service systems.

Decommissioning will not involve any major modifications to the Reactor Building structural steel without verification of the seismic qualification, as described in Section 2.2.1 of the Decommissioning Plan.

4.3 PCRV Water Leakage Prevention

The PCRV will be filled with water to provide shielding for workers during initial PCRV internal dismantlement activities. To prevent leakage from the PCRV, all penetrations which are below the PCRV water line and have had their instrumentation removed are sealed. Sealing is accomplished with either welded cover plates, welded caps, or blind flanges.

Blind flanges for the seven outlet coolant thermocouple penetrations may be removed, one at a time, during underwater removal of the thermocouple assemblies. During this time, PCRV shield water leakage will be prevented by redundant seals on the thermocouple removal tools.

There are two independent trains in the PCRV water cleanup and clarification system, to allow for maintenance and repair. Each train has sufficient valves and drains to allow isolation as required.

ATTACHMENT 3

TO P-93045

NO SIGNIFICANT HAZARDS
CONSIDERATION EVALUATION

DECOMMISSIONING OF THE FORT ST. VRAIN
NUCLEAR GENERATING STATION

10 CFR 50.92 EVALUATION

INTRODUCTION

The Decommissioning Order approving the Decommissioning Plan (DP) and authorizing the decommissioning of the Fort St.Vrain Nuclear Generating Station was issued by the NRC on November 23, 1992 (Reference 1). As identified in the Decommissioning Order, decommissioning of the FSV is authorized in accordance with the DP subject to several conditions. Condition (c) states that "If the licensee desires (1) a change in the TS or (2) to make a change in the facility or procedures described in the Decommissioning Plan or to conduct tests or experiments not described in the Decommissioning Plan, which involve an unreviewed safety question or a change in the TS, it shall submit an application for amendment of its license pursuant to 10 CFR 50.90 or request approval of a revision to the Decommissioning Plan."

Pursuant to 10 CFR 50.92, each application for amendment to an operating license must be reviewed to determine if the proposed change involves a significant hazards consideration. The Commission has provided standards for determining whether a significant hazards consideration exists [10CFR50.92(c)]. A proposed amendment to an operating license for a facility involves no significant hazards consideration if the change to the facility in accordance with the proposed amendment would not:

1) involve a significant increase in the probability or consequences of an accident previously evaluated, or

2) create the possibility of a new or different kind of accident from any accident previously evaluated, or

3) involve a significant reduction in a margin of safety.

This amendment addresses the revision to Decommissioning Technical Specifications (DTS) Section 4.0, Design Features, Subsection 4.3, PCRV Water Leakage Prevention. The proposed revision would allow temporary removal of PCRV penetration blind flanges during underwater removal of the outlet coolant thermocouples. The outlet coolant thermocouples are being removed underwater to minimize worker radiation exposures. Blind flanges for each of the seven outlet coolant thermocouple penetrations will be removed, one at a time, during underwater removal of the thermocouple assemblies. During this time, PCRV shield water leakage will be prevented by redundant seals on the thermocouple removal tools (See Figure 1). This push rod assembly allows the thermocouples to be removed underwater. It is estimated that each of the seven evolutions will take approximately eight hours.

BACKGROUND

In 10 CFR 50.36, Technical Specifications, Section 50.36 (c)(4), Design Features, states:

> Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1) [Safety limits,...], (2) [Limiting conditions for operation], and (3) [Surveillance requirements] of this section.

The FSV DTS Section 4.0, Design Features, Subsection 4.3, PCRV Water Leakage Prevention, states:

> The PCRV will be filled with water to provide shielding for workers during initial PCRV internal dismantlement activities. To prevent leakage from the PCRV, all penetrations which are below the PCRV water line and have had their instrumentation removed are sealed. Sealing is accomplished with either welded cover plates, welded caps, or blind flanges.

The PCRV has seven outlet coolant thermocouple assemblies that enter the PCRV and core via seven separate penetrations. These thermocouples provided a measurement of the temperature of the coolant helium at the outlet of each of the 37 fuel regions when FSV was operating. The FSV Updated Final Safety Analysis Report (Reference 2), Section 7.3.3.1, Outlet Coolant Temperature Measurement, stated:

> Assembly Construction. There are seven outlet coolant thermocouple assemblies that penetrate the reactor core at the core support block elevation. The seven assemblies are similar in construction except for the number of fuel regions which they cover. The longest assembly penetrates the core diametrically covering 7 regions. The remainder cover 6, 5, and 4 regions.
>
> The temperature sensors are rugged 1/4 in. O.D. Inconel-sheathed, MgO-insulated Geminol thermocouples. For each region there are four thermocouples embedded in a graphite spacer. The thermal diffusivity and mass of the graphite tends to average the temperature profile caused by uneven mixing of the coolant in the plenum and to improve the time response of the system. The spacer is located in the center of a core support block plenum. A graphite sleeve in the plenum isolates the graphite spacer and thermocouples from vibrational damage caused by coolant flow.

The thermocouples were originally intended to be removed in sections from the exterior of the PCRV through their respective penetrations, provided the dose rates were acceptable. DP Table 3.3-3 states that the thermocouples would be 50 R/hr on contact. During initial attempts to remove the thermocouple assemblies, it was determined that the radiation levels were too high

(approximately 200 R/hr) to remove them exterior to the PCRV. At this time approximately 10 feet of thermocouple assembly was cut and removed. The penetrations were blind flanged with the remainder of the thermocouples left at the approximate retracted postion of 10 feet (See Figure 2). Since the thermocouples were left in the partially retracted position, this allows unobstructed removal of some of the core support blocks, but it precludes removal of other core support blocks where the thermocouples essentially pin or dowel the blocks together.

PSC has already installed an outlet coolant thermocouple push rod assembly, including a blind flange as shown on Figure 1, on each of the seven thermocouple penetrations. Initially, the core support blocks that do not have thermocouples located inside them will be removed. Then, the thermocouple penetration covers (blind flanges) are removed, one at a time. A smooth threaded rod is attached to the push rod assembly and the thermocouples are pushed in the inward direction (inward relative to the PCRV exterior) until the thermocouples are exposed in the cavity created by the removed core support floor blocks. The push rod is then withdrawn, unthreaded, removed, and the blind flange is re-installed. The thermocouples may then be grappled and pulled further into the underwater cavity, where they can be cut as necessary and removed for disposal.

EVALUATIONS

DP Section 2.3.3.6.1, Preparation for Flooding the PCRV, states:

> Before flooding the vessel, all PCRV penetrations that are below the PCRV waterli: and have had their instrumentation removed (including instrument penetration internal components and other items such as the thermocouples routed through the core support blocks) will be sealed. These penetrations will be sealed by either one or a combination of the following: cutting and capping just outside the PCRV or by installation of bolted and gasketed blind flanges.

The penetration blind flange and redundant shaft seals are designed to withstand the static load generated by the head of water pressure acting on them. During use of the push rod assemblies, the wiper-type shaft seals are expected to prevent shield water leakage out of the penetrations.

Any minor leakage past the shaft seals could be readily contained and collected. In the unlikely event that the redundant shaft seals would completely fail, the resultant leak rate would be less than two gallons/minute. In this event, workers operating the push rod assembly could re-install the blind flange to isolate leakage past the failed seals, since the total force on the flange would be approximately 16 pounds-force. All radiological provisions will be in place prior to performance of these evolutions (i.e., water leakage collection, spray shields, water proof protective clothing). Worker doses from this leakage would still be much less than the doses that would be received from removing the 200 R/hr thermocouples through the penetrations. This small amount of leakage due to failure of the redundant shaft seals is well within the make-up capability of the shield water system.

CONCLUSIONS

Based on the information presented above, the following conclusions can be reached with respect to 10 CFR 50.92:

1. Revising DTS Design Feature 4.3 to allow temporary removal of the seven core outlet thermocouple penetration covers, one at a time, does not significantly increase the probability or consequences of an accident previously evaluated in the DP. The design calculations indicate that the push rod assembly redundant shaft seals are adequate for the pressure conditions. The wiper-type shaft seals are expected to prevent shield water leakage during use of the push rod assemblies, and any minor leakage that might result could be readily collected and contained. However, should the seals completely fail, the resulting maximum flow rate of less than two gallons/minute is well within that which could safely be contained. With the resultant force on the penetration cover of approximately 16 pounds-force, if the redundant seals did fail, the blind flange covers could be replaced. The accident analysis described in DP Section 3.4.7, Loss of PCRV Shielding Water Accident, assumes that the entire contents of the PCRV shield water system (conservatively assumed 423,500 gallons) is emptied into the reactor building due to a pipe rupture. In addition, the activity concentration is assumed to be the maximum allowed by DTS LC 3.4 of 62.4 μCi/cc. The loss of shield water accident analysis clearly bounds any potential leakage past the push rod assembly.

2. Revising DTS Design Feature 4.3 to allow temporary removal of the seven core outlet thermocouple penetration covers does not create the possibility of a new or different kind of accident from any accident previously evaluated in the DP. The potential loss of shield water due to the push rod assembly redundant shaft seals failing is of the same type/kind of accident as the accident analysis described in DP Section 3.4.7, Loss of PCRV Shielding Water Accident. The loss of shield water accident assumes that the entire contents of the PCRV shield water is released due to a pipe rupture which would be at a much greater volume flow rate than two gallons/minute.

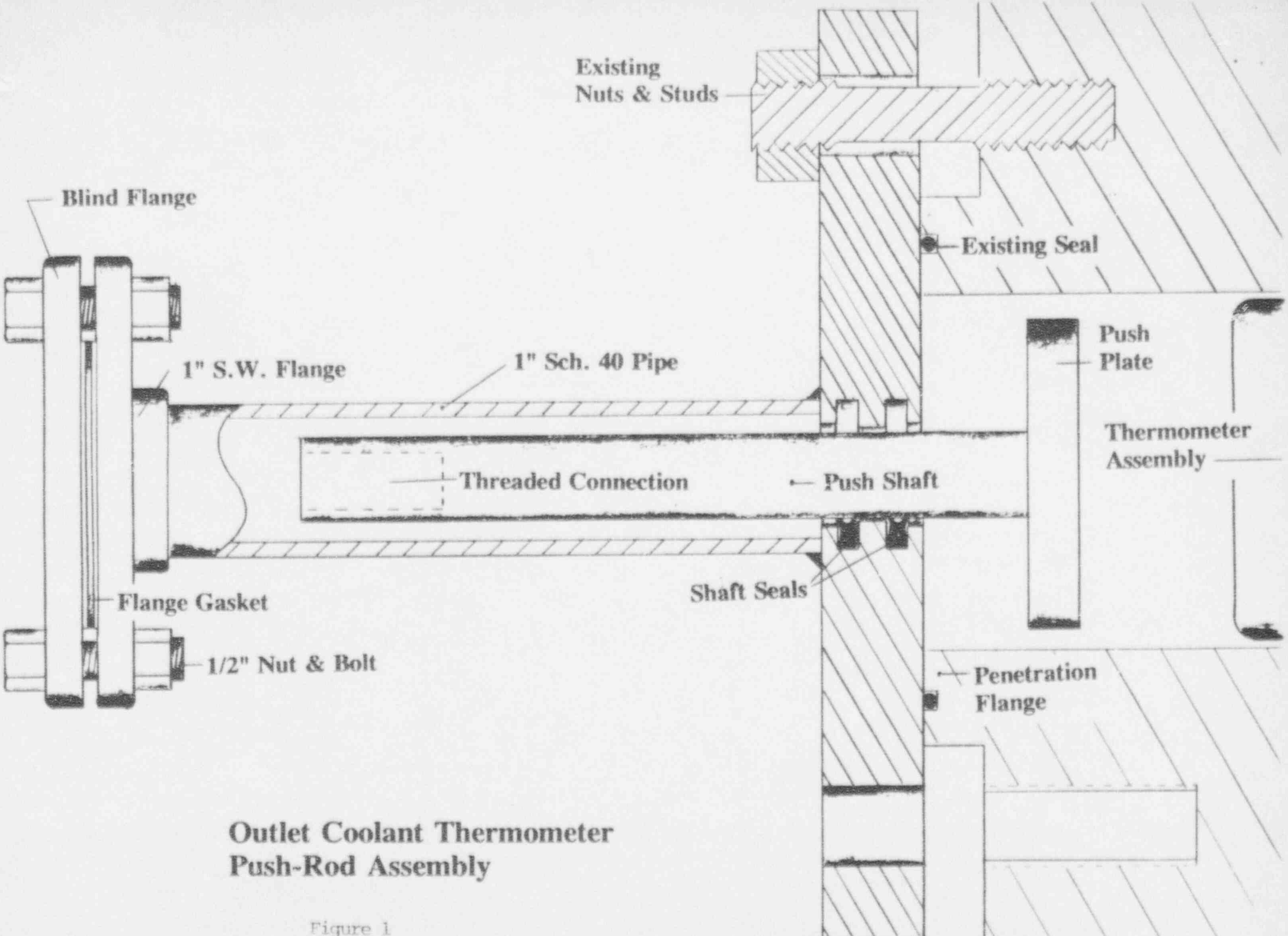
3. Revising DTS Design Feature 4.3 to allow temporary removal of the seven core outlet thermocouple penetration covers does not involve a significant reduction in a margin of safety. The Loss of Shielding Water Accident described in DP Section 3.4.7 assumes that the entire contents of the PCRV shield water is released due to a pipe rupture. This accident compared to the postulated failure of the push rod assembly redundant shaft seals is not a reduction in the margin of safety. The activity concentration in the loss of shielding water accident is limited by DTS LC 3.4 which is independent of this evaluation.

Based upon the preceding evaluation, it has been determined that revising DTS Design Feature 4.3 to allow temporary removal of the seven core outlet thremocouple penetration covers does not involve a significant increase in the probability or consequences of an accident or malfunction previously evaluated, create the possibility of a new or different kind of accident or malfunction from any previously evaluated, or involve a significant reduction in a margin of safety in the basis of any Technical Specification. Therefore, it is concluded that the licensing amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92 (c).

REFERENCES

1. NRC letter from Erickson to Crawford, dated November 23, 1993; Subject:"Order to Authorize Decommissioning of Fort St. Vrain and Amendment No. 85 to Possession Only License No. DPR-34 (TAC No. M82592)."

2. Fort St. Vrain Updated Final Safety Analysis Report, Revision 9, Public Service Company of Colorado.



Outlet Coolant Thermometer
Push-Rod Assembly

Figure 1

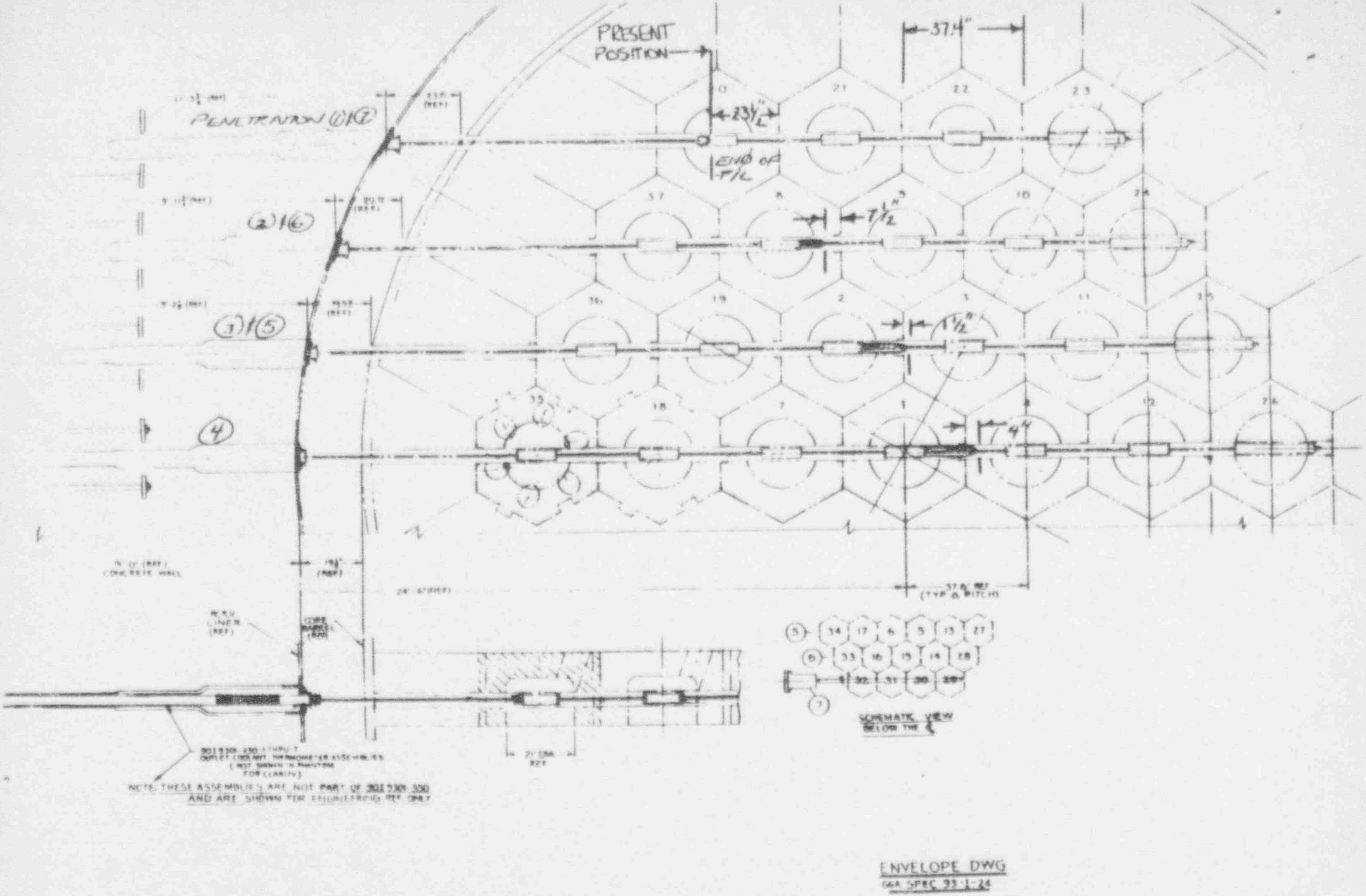

PRESENT POSITION
37.4"
PENETRATION (1) & (7)
23 1/2"
END OF F/L
(2) & (6)
7 1/2"
(3) & (5)
1 1/2"
(4)
4"
CONCRETE WALL
R.P.V. LINER (REF)
CORE BARREL (REF)
24'-6" (REF)
37.6" REF (TYP @ PITCH)
SCHEMATIC VIEW BELOW THE ℄
NOTE: THESE ASSEMBLIES ARE NOT PART OF 301330-300 AND ARE SHOWN FOR ENGINEERING REF ONLY
ENVELOPE DWG
GGA SPEC 93-1-24

Figure 2