This SAR is in Support of FCN GA4316A and is enclosed as Page_____ of _____ of that FCN even though the SAR is a separate Uncontrolled Document.

GENERAL ATOMIC COMPANY
FORT ST. VRAIN NUCLEAR GENERATING STATION
SAFETY ANALYSIS REPORT

1. INITIATING DOCUMENT: FCN FSV-GA-4244

		YES	NO
2.	CATEGORY: PLANT CHANGE	X	
	CLASS I	X	
	SAFE SHUTDOWN COOLING		X
3.	FAILURE MODES AFFECTED		X
4.	SAFETY RELATED COMPONENT, SYSTEM OR STRUCTURE CHANGE	X	
5.	SAFETY SIGNIFICANT CHANGE	X	
6.	UNREVIEWED SAFETY QUESTION		X
7.	TECH SPECIFICATION CHANGE	X	
8.	FSAR CHANGE		X

DOCUMENT CHANGE ONLY [ ]
MAINTENANCE [ ]
TEST [X]

STATE IN ITEM 10 THE BASIS FOR THE BOXES CHECKED

9. APPLICABLE FSAR OR TECH SPEC SECTIONS REVIEWED: Tech. Spec. Sections 2.14 6.1, LCO 4.1.2, LCO 4.1.3 and FSAR Section 3.5.3.1

10. BASIS FOR SAFETY EVALUATION: (Add additional Sheets if Required): This change is Class I and safety related because temporary modifications are being made to a Class I system. Safe shutdown cooling is unaffected by this temporary change since the operability of the orifice valve is unaffected. This change does not alter failure modes, nor does it degrade system function. However, inasmuch as this change involves control rod drives, it is prudent to consider it as being safety significant and to have it reviewed accordingly. This change does not constitute an unreviewed safety question. There is no increase in the probability or consequences of a radiological accident from that evaluated in the FSAR, and the margins of safety as defined in the bases for the technical specifications are maintained. No FSAR change is required A Tech. Spec. change may be required as a result of not having 37 pairs of control rods. See attached pages for more detailed support of this SAR.

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11. IS SAN DIEGO SAFETY ANALYSIS/LICENSING REVIEW REQUIRED? YES [X] NO [ ]

12. HAS SAN DIEGO SAFETY ANALYSIS/LICENSING REVIEW BEEN PERFORMED? YES [X] NO [ ]

13. K. E. Asmussen 10-19-78 — INITIATOR/DATE
A. J. Kennedy 10/24/78 — LICENSING/DATE

14. GAC ENGR. REVIEW/DISPOSITION: I concur with this safety evaluation presented in in support of the temporary installation of two instrumented control and orificing assemblies.

ENG'R/DATE [illegible] (10/20/78)
R. Losenberg 10/22/78

FORM: SAR 4/77

SAFETY ANALYSIS REPORT
FOR
INSTRUMENTED CONTROL AND ORIFICING ASSEMBLIES

1. FOREWORD

Core temperature and nuclear channel fluctuations have been observed at the Fort St. Vrain nuclear generating station. While the fluctuations are irregular and complex, the average system parameters are not significantly affected. As part of a comprehensive diagnostic program to investigate and characterize the fluctuations, installation of in-core instrumentation is being considered. Installation of two in-core instrumentation packages can be accomplished by modifying two control rod drive assemblies. The safety evaluation of this proposed temporary use of instrumented control rod drive assemblies during cycle 1 and/or cycle 2 is presented below.

2. PLANNED MODIFICATION

It is planned to make temporary modifications to two control and orificing assemblies by removing one control rod from each assembly and adding an instrument package which occupies the space vacated by the removed control rod. However, while the instrument package does occupy the space vacated by the removed control rod, the instrument package is not connected to the corresponding control rod cable. Instead, the instrument package is isolated from and independent of normal operation of this remaining control rod.

The axial positioning of the instrument package is independent of the operation and positioning of the single control rod. It is intended that each instrument package will be manually inserted into and withdrawn from the core channel normally occupied by the removed control rod. Consequently, these operations are to be done at refueling conditions. Thus, at refueling conditions, with all control rods fully inserted, the instrument packages will be lowered into the selected regions of the core and secured. The instrument package will then remain stationary during subsequent reactor operation and testing.

The in-core instrumentation will consist of the following (the purpose of each instrument and its identifying item number on Figure 1 is also indicated):

A. <u>three thermocouples</u> - to monitor helium temperature at three axial locations; item nos. 29, 30, and 31 on Figure 1

B. <u>a self-powered neutron detector</u> (SPND) with a compensating cable - to monitor the local/region flux level at the core midplane; item no. 27 on Figure 1

C. <u>a fission couple</u> - to monitor local/region flux level near the bottom of the core; item no. 24 on Figure 1

D. <u>a fission chamber</u> - to monitor region flux level; located in bottom block of upper reflector; item no. 28 on Figure 1

E. <u>a microphone</u> - to detect changes in turbulence, i.e., flow; item nos. 25 and 43 on Figure 1. Item no. 25 (KAMAN microphone) is on C&O assembly S/N 20 and item no. 43 (GULTON microphone) is on C&O assembly S/N 43.

These instruments will be securely attached at various axial locations to a support rod. The axis of the support rod will be maintained at a position near the centerline of the control rod channel by means of several spiderlike assemblies at various axial locations (Fig. 1). The basic materials of construction are magnesium oxide, stainless steel, inconel and chromel-alumel. Joints will be welded.

In addition to the above noted instrumentation, the two control and orificing assemblies are provided with pressure transmitters located in the mechanism compartment. They are plumbed in a manner to measure the pressure differential across their respective orifice valves.

One of the control and orificing assemblies is provided with a linear variable differential transducer (LVDT). It is also located in the mechanism compartment and is mounted at the upper end of the orifice valve drive mechanism. This device will provide information relative to the vertical movement (and/or growth) of the region's central column over which the control and orificing assembly is mounted.

Other temporary modifications were required in order to pass the temporary in-core instrumentation leads through the primary and secondary closures. These included making two temporary ports in the primary closure piece of the control and orificing assembly, and the fabrication of temporary secondary closure plates containing four ports (see Fig. 2, in particular see views A and D).

After exiting the temporary secondary closure plate, the temporary in-core instrumentation leads will pass between hold down plates through a notch provided to accommodate their passage (see Fig. 2, view A).

There are many pairs of refueling regions for which it is feasible to install the instrumented control and orificing assemblies. However, the following three pairs of refueling regions are considered to be prime candidate locations for the installation of the two instrumented control and orificing assemblies: 35 and 36, 4 and 35, and 4 and 36 (Fig. 3). The effect on the power distribution, shutdown margin, rod withdrawal sequence, and maximum worth rod of installing two instrumented control and orificing assemblies in each of these three pairs of regions has been investigated for cycles 1 and 2.

The relative axial positions of the various instruments, when installed in the reactor, are shown on Figure 4. During transport of the instrumented control and orificing assembly and prior to their insertion into the core, the instruments are withdrawn into the control rod drive assembly much like a withdrawn control rod.

Following completion of all measurements requiring the use of the instrumented control and orificing assemblies, all temporary instrumentation is to be removed (at the hot servide facility) and disposed of. The drives will then be returned to their original operational condition; i.e., the removed control rod will be reinstalled and all nonfunctional ports required for the temporary instrumentation will be sealed.

The instrumentation and the modifications to the plant to accommodate the instrumentation are shown on Figures 1 and 2.

3. SAFETY ANALYSIS

Specific information for the basis of the safety evaluation follows:

A. Appropriately designed seals permit passage of the in-core instrumentation leads through the primary and secondary pressure boundaries while maintaining the system's pressure integrity. All pressure retaining members are designed, fabricated, and tested in accordance with the requirements of Section 3 of the ASME Boiler and Pressure Vessel Code for Class 1 components for the primary closure parts and Class 2 for the secondary closure parts and demonstrated to be acceptable in accordance with the existing pressure retention and leakage criteria.

B. The operability of the reserve shutdown system and the orifice valve and its drive are unaffected by the modifications.

C. The presence of the instrument package in the control rod channel will have an insignificant effect on the normal helium flow through the channel. More specifically, calculations indicate that the flows through the channel containing the instrument package, through the corresponding channel with the control rod withdrawn, and through the corresponding channel with the control rod inserted are essentially the same (within 10%). These conclusions result from a comparison of the relative flow resistances for these cases. The values of flow resistance for the instrument package and for the control rod were determined by means of full scale flow tests.

The flow resistance due to the instrument package in the control rod channel is slightly less than that of a fully inserted control rod. However, the resulting difference (increase) in helium mass flow is not significant.

D. The technical specifications require that the reactor must be capable of achieving a shutdown margin of 1% (0.01 $\Delta\rho$ subcritical) at a core average temperature of $\leq 80°F$ with no xenon and the maximum worth control rod pair withdrawn (LCO 4.1.2). The following three pairs of refueling regions were considered as potential locations for the installation of the two instrumented control and orificing assemblies: 35 and 36, 4 and 35, and 4 and 36. The corresponding shutdown margin was calculated (using the physics design code GAUGE; FSAR Section 3.5 Ref. 3) for each of these three different configurations involving pairs of instrumented control and orificing assemblies. To show that the shutdown margin is satisfactory for these cases with the maximum worth rod withdrawn, the summary in Table 1 gives the calculated shutdown margin at the middle of cycle 1 and at the beginning (BOC), middle (MOC), and end (EOC) of cycle 2. The middle of cycle and beginning of cycle conditions were found to yield the minimum shutdown margins for cycles 1 and 2, respectively. In each case, the shutdown margin was found to be greater than 2% (0.02 $\Delta\rho$).

S.b. $\geq 80°F$

It should be noted that the calculated shutdown margins of >2% include the following conservative assumptions. First it was assumed ...

Table 1

Shutdown Margins - Cycles 1 and 2

Regions Assumed to be Unrodded			Shutdown Margin (%) Δρ			
Regions Containing Instrumented C&O Assemblies	Maximum Worth Rod Pair		Cycle 1	Cycle 2		
	Cycle 1	Cycle 2	MOC*	BOC	MOC	EOC
35 & 36	37	18	2.4	2.1	2.1	2.1
4 & 35	34	34	4.2	3.3	3.3	3.3
4 & 36	37	37	3.0	3.0	3.5	4.4

* MOC conditions yield minimum shutdown margin during cycle 1.

that all the Pa-233 had decayed to U-233. Secondly, since the scram time of a single rod on a modified control and orificing assembly is anticipated to be approximately double what it is with two rods, it was conservatively assumed that these single rods remain withdrawn during a scram. Thus, in calculating the >2% shutdown margin, it was assumed that three regions were unrodded; namely, the regions containing the two instrumented control and orificing assemblies plus the region associated with the maximum worth control rod pair.

E. The criteria serving as the basis for establishing the control rod withdrawal sequence are given in Technical Specification LCO 4.1.3. Calculations with the physics design code GAUGE (FSAR Section 3.5 Ref. 3) have demonstrated the acceptability of using the standard rod withdrawal sequence when the instrumented control and orificing assemblies are installed in any of the proposed configurations for cycle 1 or cycle 2. The power distributions are well within the allowable limits set by LCO 4.1.3 in all cases. These calculations modeled explicitly the fact that one control rod consisted of boronated graphite compacts containing 40 w/o natural boron while the second control rod consisted of compacts containing 30 w/c natural boron (see item H below).

The power levels at which the candidate regions for instrumented control and orificing assemblies become unrodded during the standard rod withdrawal sequences of cycles 1 and 2 are summarized in Table 2. Once the single rod of an instrumented control and orificing assembly has been withdrawn, it, of course, has no effect on the power distribution. When rodded, however, the presence of a single control rod in a region instead of a pair of rods does tend to increase the region's peaking factor (RPF).

This effect is most significant for the case of installing the instrumented control and orificing assemblies in the adjacent regions 35 and 36 during cycle 1. Furthermore, the effect is most pronounced when the corresponding rod groups containing these two regions are fully inserted (at $\lesssim$18% power). This results in about a 40-50% increase in the power in these two regions. However, these are low

Table 2

Region	Power Level at which Region is Unrodded* Cycle 1	Cycle 2
4	Subcritical	Subcritical/Critical
35	~20%	Subcritical
36	~50%	~25%

* Assuming standard rod withdrawal sequence.

power regions (RPFs < 0.5) and the change is easily accommodated by adjusting the orifice valves. The power in the regions immediately adjacent to regions 35 and 36 also increases by 6 to 12% with the remainder of the core regions decreasing slightly (by ≤7%) or not at all (see Fig. 5). Although the power distribution is somewhat different when instrumented CRDs are installed, the differences are well within the Technical Specification limits of LCO 4.1.3.

When the control rod in region 35 is withdrawn (at ~20% power), only region 36 has a single rod and the differences are less. Similarly, in the case of one instrumented CRD in one of the peripheral regions (35 or 36) and one in region 4, the power distribution changes are smaller than those shown for regions 35 and 36.

During cycle 2, the only significant case is when region 36 contains one of the instrumented control and orificing assemblies. The other two candidate regions for instrumented assemblies become unrodded while the reactor is still subcritical, or just critical (see Table 2). As mentioned above, this effect is most pronounced when the single rod is fully inserted (at power levels <~25%) and results in about a 32% increase in the RPF of region 36. Again, however, the change is in a low power region (RPF = 0.56), the resulting power distribution is well within the Technical Specification limits of LCO 4.1.3, and the changes are easily accommodated by orifice valve adjustments (see Figure 6).

F. The calculated maximum control rod pair reactivity worth was likewise investigated and determined to be in compliance with the appropriate criteria of LCO 4.1.3. Thus, the standard control rod withdrawal sequences of cycles 1 and 2 are acceptable and do not need to be modified.

G. Since the instrument package will extend the full length of the core and since it has a low neutron cross section for capture, the effect of its presence on the axial and radial power distributions is negligible.

H. The control rods on the standard control and orificing assembly of refueling region 35 consist of boronated graphite compacts containing 40 w/o natural boron. The single control rod of the instrumented control and orificing assembly to be installed in refueling region 35 will have boronated graphite compacts containing 30 w/o natural boron. There is no significant difference in the relative worth of these two types of control rods since they are both essentially black to thermal neutrons. The calculated worth of a single 30 w/o boron rod in region 35 is negligibly different from that of a 40 w/o boron rod, the calculated difference being less than 0.0001 ΔK.

In any event, as described above, the shutdown margin calculations demonstrated compliance with technical specification limits while assuming this rod to be stuck in the fully withdrawn position. Furthermore, power distribution calculations were performed which conservatively modeled a 30 w/o boron control rod in region 35 during the rise to power. The resulting power distributions were well within the technical specification limits of LCO 4.1.3 for both cycles 1 and 2.

I. The remaining control rod in each of the two instrumented drives may be inserted or withdrawn in the normal manner. However, the scram insertion time for these two rods is anticipated to be approximately doubled. This characteristic is neither deleterious nor considered as a degration of reactor performance since, as noted in the preceding, these rods are not required to establish the shutdown margin necessary for compliance with Technical Specification requirements.

NOTE: The slack cable assembly is adjusted to properly function with a single rod in the same manner as with a rod pair. The trip point will be appropriately reset.

J. Mechanical failure of an instrument package is analogous to the failure of a control rod. The bottom of the instrument package has been designed such that in the unlikely event of a failure the exit to the control rod channel would not be blocked to prevent the flow of coolant gas (see Fig. 7).

Nevertheless, conservative thermal and fuel performance analyses have been performed assuming the instrument assembly drops to the bottom of the channel and completely blocks the coolant flow while the plant is operating at full power. These analyses show that, even under these most severe conditions and if the failure of the instrument assembly went unnoticed for 48 hours or more, the instrument package would not melt and the impact on fuel performance would be negligible. Actually, if the instrument package should fall, the failure would result in anomalous readings which would give immediate indication of such a failure.

K. When installed, the bottom of the instrument package will be about five inches above the bottom of the control rod channel. The consequences of the instrument package, which weighs ~12 lbs, falling during full flow conditions and impacting on the bottom of the channel has been evaluated and determined to be insignificant. The reflector block upon which the instrument package would impact has been designed and tested to withstand the impacts of regular control rods, with shock absorbers, weighing ~120 lbs falling from the withdrawn position.

L. The probability of an instrument package failure has been minimized by careful design, the judicious choice of materials having high integrity at high temperatures, careful attention given to fabrication techniques (e.g., joints are welded), extensive inspection procedures to assure quality, pre-testing of the actual assembly and of mockups (see below), and by conservatively designing for large margins of safety. Additionally, the bottom of the instrument package has been designed to act as a catcher for small parts, should an unlikely occurrence of that nature happen.

M. The support structure for the instrumentation is fabricated from Inconel 600. The stress on the support rod due to the maximum differential pressure drop across the catcher plus the static weight of the entire assembly has been conservatively calculated and found to be less than 10% of the tensile yield stress of the material at the highest anticipated operating temperature. The highest temperatures occur at the bottom of the assembly. Actually, the portion of the support rod which experiences the highest stress is at the top of the assembly and is thus in a significantly cooler environment.

N. Tests performed to confirm and support the mechanical design and fabrication of the instrument assembly included the following:

a. The actual instrumented assemblies have been pull tested to 50% of the tensile yield stress, followed by careful reinspection. The load placed on the assemblies during this test is a factor of five greater than the maximum anticipated tensile force.

b. A mockup of the bottom three feet of the instrument support rod has been fabricated from production materials and tested at temperature to its failure point. This mockup represented all typical welds to the central support rod.

c. A full scale model of the entire instrument package has been made of appropriate structural material, inserted into a simulated control rod channel and flow tested. Air was used as the test fluid and was drawn through the control rod channel so as to simulate the range of Reynolds number expected in the reactor.

d. The ability to manually insert and withdraw the instrument assembly into the control rod channel has been demonstrated by testing. The control rod channel mockup for this test modeled the maximum possible amount of control block misalignment for each block.

4. IMPACT ON PLANT OPERATION

The presence of the instrumented control rod drives in the reactor will have essentially no impact on the normal operation of the plant. The only noticeable difference will be a somewhat different power distribution at power levels less than 50% and 25% of full power during cycles 1 and 2, respectively. This is easily accommodated by the normal procedure of adjusting the orifices as is done routinely during every rise to power. Operation of the pla.t will be within all Technical Specification and design limits.

As mentioned in Section 2, it is intended that each instrument package will be manually inserted into and withdrawn from the core. While it is expected that the instrumented control and orificing assemblies will spend a relatively short time in the reactor, a conservative estimate of the dose rate from the lead wire bundle (and support rod) assumed three month's exposure at 70% power. The resulting anticipated dose rate one foot above the lead wire bundle coiled in the external void space of the control and orificing assembly primary closure is anticipated to be 12 mr/hr. The expected dose rate to the hand holding the hotter portion of the lead wire bundle is 100 - 200 mr/hr.

In the highly unlikely event that one of the instrument packages should fail and fall (~5 inches) to the bottom of the control rod channel, the retrieval of the assembly would be accomplished much in the same manner as would the retrieval of a failed control rod. Such an operation would involve the use of the fuel handling machine and the core service manipulator and special tools (FSAR Sections 9.2.4, 9.2.5, and 9.2.6). The core service manipulator attachments and special tools include: a five-inch hand, an eight-inch hand, a hook, a fuel element grapple, a fuel element pickup probe, a control rod removal tool, a cable cutoff tool, a fuel element righting tool, and a core service vacuum tool. These devices were designed to perform non-routine salvage operations in the space within the Prestressed Concrete Reactor Vessel (PCRV) that is accessible through the refueling penetrations, and are capable of reaching to the core support floor.

5. CONCLUSION

This change does not constitute an unreviewed safety question and there is no increase in the probability of or the radiological consequences of a system failure. The change does not create the possibility of a radiological accident different from that evaluated in the FSAR, and the margins of safety as defined in the bases for the Technical Specifications are maintained.

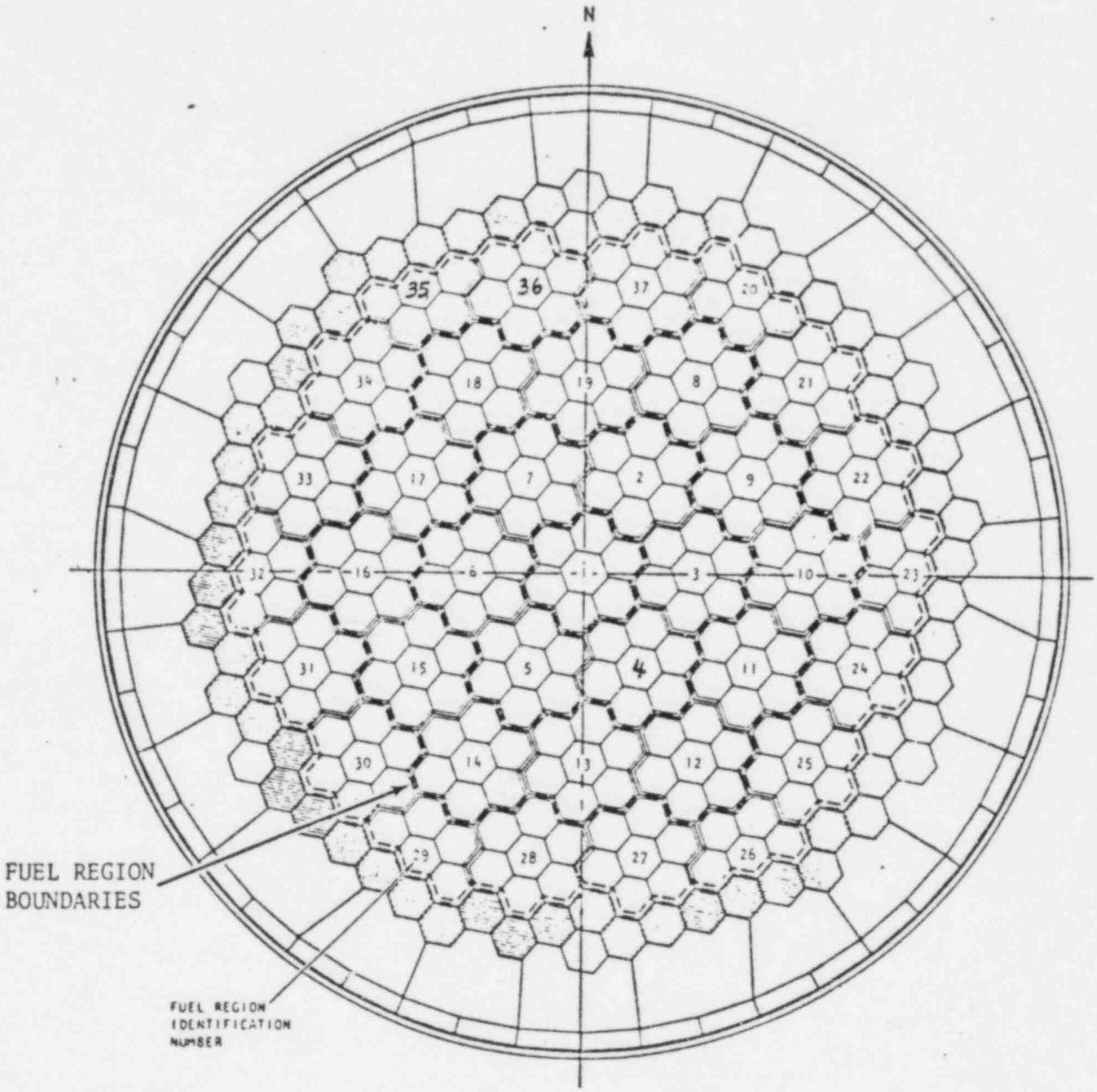
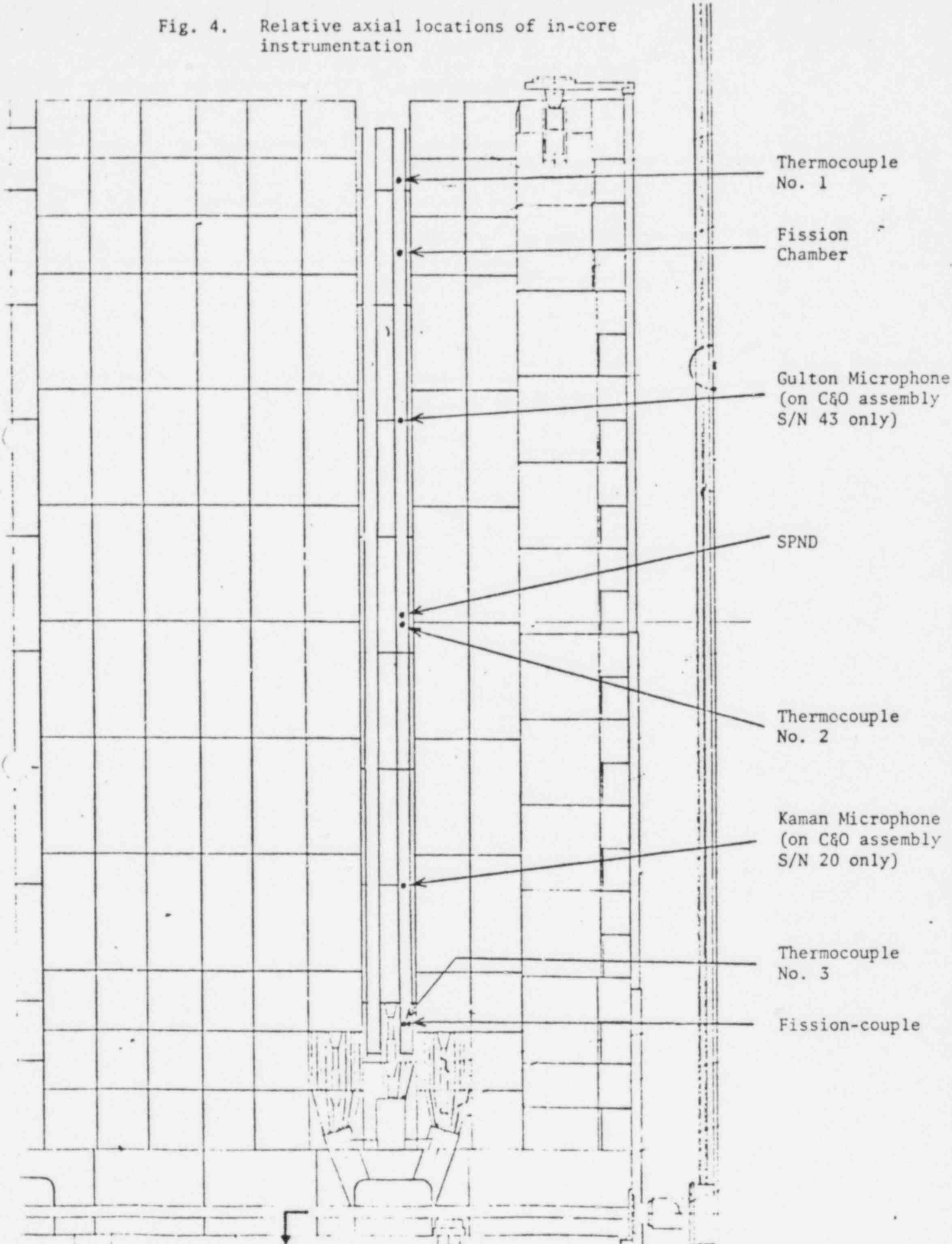


Fig. 3. Plan view of Fort St. Vrain reactor

Fig. 4. Relative axial locations of in-core instrumentation



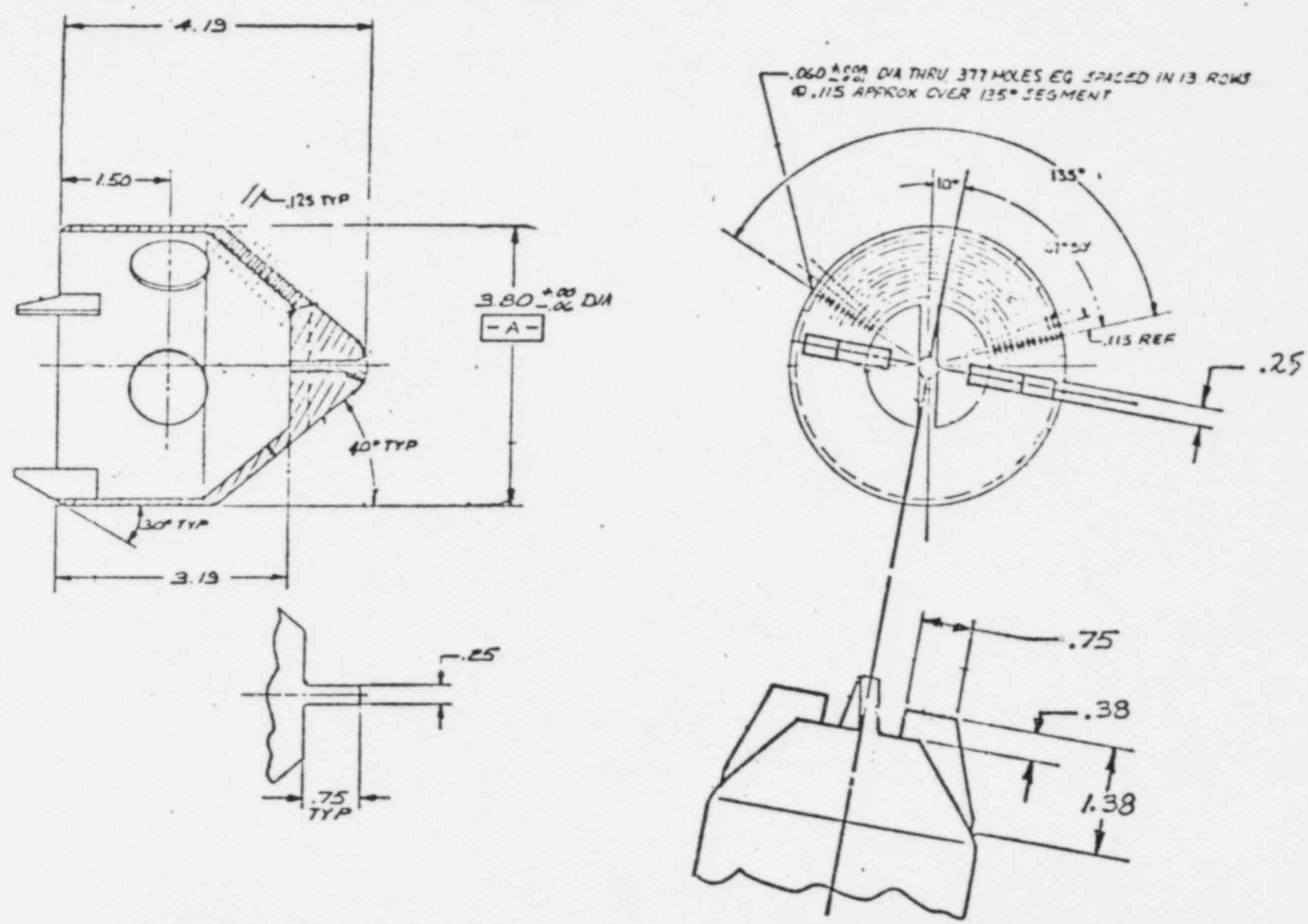


Fig. 7. Bottom of instrumented assembly - catcher