GENERAL ELECTRIC

NUCLEAR ENERGY

ENGINEERING

DIVISION

GENERAL ELECTRIC COMPANY, P.O. BOX 460, PLEASANTON, CALIFORNIA 94566

July 9, 1979

Mr. Robert W. Reid, Chief
Operating Reactors Branch #1
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D. C., 20555

Subject: STRUCTURAL MODIFICATIONS FOR THE GENERAL ELECTRIC TEST REACTOR - DOCKET 50-73

Reference: Letter, R. W. Reid to R. W. Darmitzel, dated June 27, 1979

Dear Mr. Reid:

Enclosed are General Electric Company's responses to the questions contained in the referenced letter. Answers are provided for all questions discussed during the meeting of June 18, 1979. Questions #6 and #7 were not raised during the meeting and will require additional time in which to respond and will be forwarded with the response to the additional 28 questions just received.

If we can be of further assistance in this matter, please let me know.

Very truly yours,

RW Darmitzel

R. W. Darmitzel
Manager
Irradiation Processing Operation

vcc

Enclosure

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GENERAL ELECTRIC

AFFIRMATION

The General Electric Company hereby submits the attached response regarding Structural Modifications for the General Electric Test Reactor - Docket 50-73.

To the best of my knowledge and belief, the information contained herein is accurate.

By: R W Darmitzel

R. W. Darmitzel, Manager
Irradiation Processing Operation

OFFICIAL SEAL
VIRGINIA C. CASQUEIRO
NOTARY PUBLIC - CALIFORNIA
ALAMEDA COUNTY
My comm. expires MAR 8, 1981

Submitted and sworn before me this ninth day of July, 1979.

Virginia C Casqueiro, Notary Public in and for the County of Alameda, State of California.

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External Distribution

Response to R. W. Reid's Letter re: Structural Modifications

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Advisory Committee on Reactor Safeguards

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RESPONSES TO NRC REQUEST FOR INFORMATION BASED ON GETR SITE VISIT HELD 18 JUNE 1979

REQUEST NO. 1

Provide the details of the evaluations of the effects of the impact of the stack, or any credible portion of the stack, and all of the cooling tower hardware on the fuel flooding system supply lines.

Response to Request No. 1

The fuel flooding system (FFS) line and 4-inch diameter schedule 80 stainless steel protective shield pipe are buried 8 to 12 inches beneath the ground in the vicinity of the cooling tower and exhaust stack. The FFS line will not be damaged by either the cooling tower or the exhaust stack. Information concerning the potential effects of the cooling tower failing and falling on the shield pipe are given in EDAC Report 117-217.08. The effects of the potential impact of the stack are given below.

The 8 to 12-inch soil cover over the steel protective shield pipe was provided to cushion the fall of any object, including the exhaust stack. Thus the only potential condition that could affect the FFS line is if the shield pipe is exposed during a postulated surface rupture offset and then is subsequently impacted by the stack. Because of the large diameter to thickness ratio of the stack relative to the steel protective shield pipe, the stack would likely collapse on impact with the shield pipe. In addition, the loose soil (produced by the postulated surface rupture offset) beneath the shield pipe would cushion the fall of the stack and would help absorb the kinetic energy of the stack.

321 266

REQUEST NO. 2

Regarding the screw jacks providing vertical support for the primary heat exchanger:

(a) Justify that it is not necessary to provide a locking mechanism (e.g. lock nuts).

(b) Discuss how impact loads were considered in the design of these supports to resist earthquake loadings since they are capable of resisting only compressive forces and vertical deflections due to upward loading may create gaps between the heat exchanger and the jacks.

(c) Discuss in detail the installation procedures for these jacks, including the significance of any precompression and how the magnitude of this precompression was determined.

Response to Request No. 2(a)

Each screw jack will be locked in final position (see Response 2(c)) by threading a stainless steel band clamp around the lever socket (actuating device) and around the screw jack body. The clamp will then be pulled tight and crimped, thus preventing further movement.

Response to Request No. 2(b)

Based on the results of a recent stiffness test of one heat exchanger screw jack, a revised dynamic analysis of the heat exchanger was conducted, and it was determined that the screw jacks will always be in compression for the criterion earthquake loading.

The previous calculations of the vertical dynamic response of the heat exchanger were based on conservatively low estimates of the vertical stiffness of the heat exchanger support structure because the actual vertical stiffness values were not known at the time of the analysis. Table 5-2 in EDAC Report 117-217.06 gives a net tension force value of 27 kips in the vertical direction due to the combined gravity and earthquake loads. This result was based on a vertical natural frequency of 19.1 Hz. Using the experimental stiffness results for the screw jack and the stiffness properties of the tension columns the vertical natural frequency was found to be greater than

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Response to Request No. 2(b) - continued

33 Hz. Using 33 Hz the uplift forces were found to be less than the weight of the heat exchanger which insures that gaps will not form between the heat exchanger and the screw jacks, and thus no impact loads will occur.

Response to Request No. 2(c)

One of the screw jacks was recently removed for vertification of its stiffness characteristics (see 2(b) above). The three screw jacks will be reinstalled according to the following procedure:

1. Loosen the 1-1/4 inch nuts on the three tension columns.
2. Remove the shims from between the tops of the tension columns and the tube sheet flange. If the shims are not loose, raise the screw jacks in 15^{o} to 20^{o} rotational increments of the pinion gear (~0.01 inch vertical increments) until the tension columns shims are loose.
3. Loosen and remove the five remaining 1-1/4 inch studs and nuts between the tube sheet flange and the original heat exchanger support structure.

   At this point 100% of the heat exchanger deadload is supported by the three screw jacks.
4. Lock each screw jack in position by looping a stainless steel band clamp around the screw jack lever socket and around the screw jack body. Pull the band clamp tight and crimp.
5. Reinstall the shims on the three tension columns and tighten the nuts. Either stake or provide jam nuts for the studs.

The installation will be performed in accordance with documented procedures and the existing quality assurance plan.

REQUEST NO. 3

Indicate any systems inside the containment building which will have to be moved to accommodate the installation of the fuel flooding system and describe in detail the nature of the required modifications.

Response to Request No. 3

Following is a description of the routing of the fuel flooding system (FFS) inside the containment building and the equipment which must be modified to accommodate this new system. One supply line (the north line) enters penetration 19 on the northwest of the containment building. This penetration is located about three feet above grade level and will only contain the FFS supply line. The existing primary pressurizer vent and containment building leak test lines (which currently occupy this penetration) will be relocated elsewhere.

NOTE - Penetration 18 is located about 18" directly below penetration 19 and contains a 1-1/2 inch resin transfer line, a 1-inch nitrogen gas line, a 1-inch unused spare line and several smaller lines. This piping and associated valves are located well below the FFS line and will not interact with the line under normal or accident conditions. Some mechanical maintenance equipment is presently stored in this area. This equipment will be moved prior to reactor operation and storage in the area will be restricted.

The north FFS line from penetration 19 to the reactor biological shield wall (about 3 feet away) is a flexible stainless steel pipe with stainless steel braid covering. There is no overhead equipment in this region which represents a missile threat to the FFS line.

At the biological shield, the north FFS line is routed vertically to the third floor. In all exposed regions, the FFS line consists of a flexible hose encased in steel shielding to eliminate missile hazard. There is no equipment in exposed regions on the first, second or third floor which requires modification. The entire assembly is mounted directly against the

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massive biological shield. Where the FFS line penetrates the second and third floors, it is routed through oversized holes. In the floor regions, the line consists of a rigid stainless steel pipe. On the third floor, the north FFS line is routed both to the canal and the pool. The line which traverses the third floor to the canal consists of a combination of rigid and flexible stainless steel pipe encased in a steel shield. The line to the pool also consists of a combination of rigid and flexible stainless steel pipe, but is buried in a trench (in the third floor) covered by a metal plate. Considerable modification on the third floor (i.e., the addition of the Missile Impact System) has been done to assure protection of the FFS piping and other safety systems in this region. The new third floor Missile Impact System is discussed in General Electric's submittal dated July 20, 1978 (Updated Responses to the NRC Order to Show Cause). Other than the modifications described therein, no other third floor equipment must be modified to assure protection of the FFS supply lines or other safety related equipment.

The FFS line entering the containment building on the southwest side is similar to the redundant north line described above. The south line enters penetration E-15 located about 9 feet above grade level. This penetration contains two 2-inch and one 2-1/2 inch lightweight electrical conduits which do not represent a hazard to the FFS line. The FFS line from the penetration to the biological shield wall (approximately 4 feet away) is a flexible stainless steel pipe with stainless steel braid covering. An electrical terminal cabinet near this area will receive additional concrete anchors to assure it will remain in position in an earthquake. At the biological shield wall the south FFS line is routed vertically to the third floor, and then to the pool and canal, in the same manner as the north line. No equipment (other than that discussed above) requires additional modification to assure protection of the south FFS line.

REQUEST NO. 4

> Provide the deflection patterns and the deflections of the containment building under the maximum seismic loadings, including the consideration of buckling.

Response to Request No. 4

The possible deflection patterns for the containment building (shell) due to maximum seismic loads are discussed in two parts: deflections caused by vibratory ground motion and deflections caused by postulated surface rupture offset beneath the reactor building.

Vibratory Ground Motion--A linear elastic analysis of the reactor building, including the containment shell indicated that the containment shell stresses exceed a conservative estimate of the critical buickling stress at the first floor level. This analysis indicated that local buckling deflections may occur in the containment shell in the region of the first floor for the criterion earthquake loading.

Since the deflections of the containment shell are limited by the concrete structure located two inches inside the containment shell and the ring stiffeners, global buckling deflections of the shell during vibratory ground motion would be prevented.

Surface Rupture Offset--An analysis for postulated surface rupture offset beneath the reactor building was performed and reported in EDAC Report 117-217.02. It was found that soil passive pressures caused by a rupture offset beneath the reactor building may cause the exterior basement wall to be loaded beyond its flexural yield capacity. Deflection at the mid-height of the basement wall could be one meter or less. If the basement wall deflects inward the containment shell will deflect the same amount. The associated downward movement of the containment shell in the region of the surface rupture offset would be less than one foot.

Response to Request No. 4 - continued

A second possible deflection pattern of the containment shell caused by surface rupture offset would be due to the reactor building basement slab spanning the gap caused by a vertical thrust component of the hypothetical fault occurring beneath the east side of the reactor building. It was reported in EDAC Report 117-217.02 that for this load case the basement would yield upward pushing the containment shell vertically. This may cause the shell on the east side of the reactor building to buckle. However, only local buckling deflections would occur because of the close proximity of the adjacent concrete structure (i.e. basement wall, floor slabs, and concrete columns).

321 272

REQUEST NO. 5

Demonstrate that all modes of containment building failure will not impact the fuel flooding system penetrations.

Response to Request No. 5

The fuel flooding system (FFS) penetrations are located between the first and second floors in the region where the concrete core walls of the reactor building are located. As discussed in Response to Request No. 4 only local buckling deflections will occur because the containment shell is constrained by the concrete structure and the stiffener rings. Effects of the shell deflections are discussed in two parts: potential penetration deformation caused by vibratory ground motion and caused by surface rupture offset.

Potential Penetration Deformation Caused by Vibratory Ground Motion--An analysis of the containment shell for vibratory ground motion indicated that buckling could occur in the lower half of the region of the shell between the first and second floor levels. Since the FFS penetration on the south side of the reactor building is located in the upper half of the region between the first and second floors, it would be unaffected. The FFS penetration on the north side of the reactor building would be subjected to buckling stresses in the shell. An approximate, but conservative, stress analysis of the north penetration was conducted and it was found that the penetration can withstand the local buckling stresses which would occur in this region (see EDAC 117-217.08). The penetration nozzle would respond as an out-of-plane stiffener to insure that if the containment shell buckles the penetration hole would remain plane and not warp. The small buckling deflections of the containment shell might cause the nozzle to rotate slightly to conform with the buckled shape of the shell. However, because the FFS line at the penetrations consists of a flexible hose, this type of deflection can be accommodated.

Potential Penetration Deformation Caused by Postulated Surface Rupture Offset--Deflection of the containment shell caused by failure of the basement

exterior wall will pull the containment shell downward in the region between the first and second floors. The maximum one foot vertical displacement of the containment shell associated with the failure of the basement wall would not affect the FFS lines because several feet of slack in the FFS lines is provided on each side of the penetrations. In addition, the penetrations are located clo: to the concrete core wall which will constrain the shell deflections.

The second mode of containment shell deflection referred to in Response to Request No. 4 caused by postulated surface rupture offset would only affect the FFS penetration on the north side of the reactor building. The type of buckling associated with this mode would be similar to buckling caused by vibratory ground motion, and therefore the FFS line would behave in a similar manner as discussed above for the effects of vibratory ground motion.

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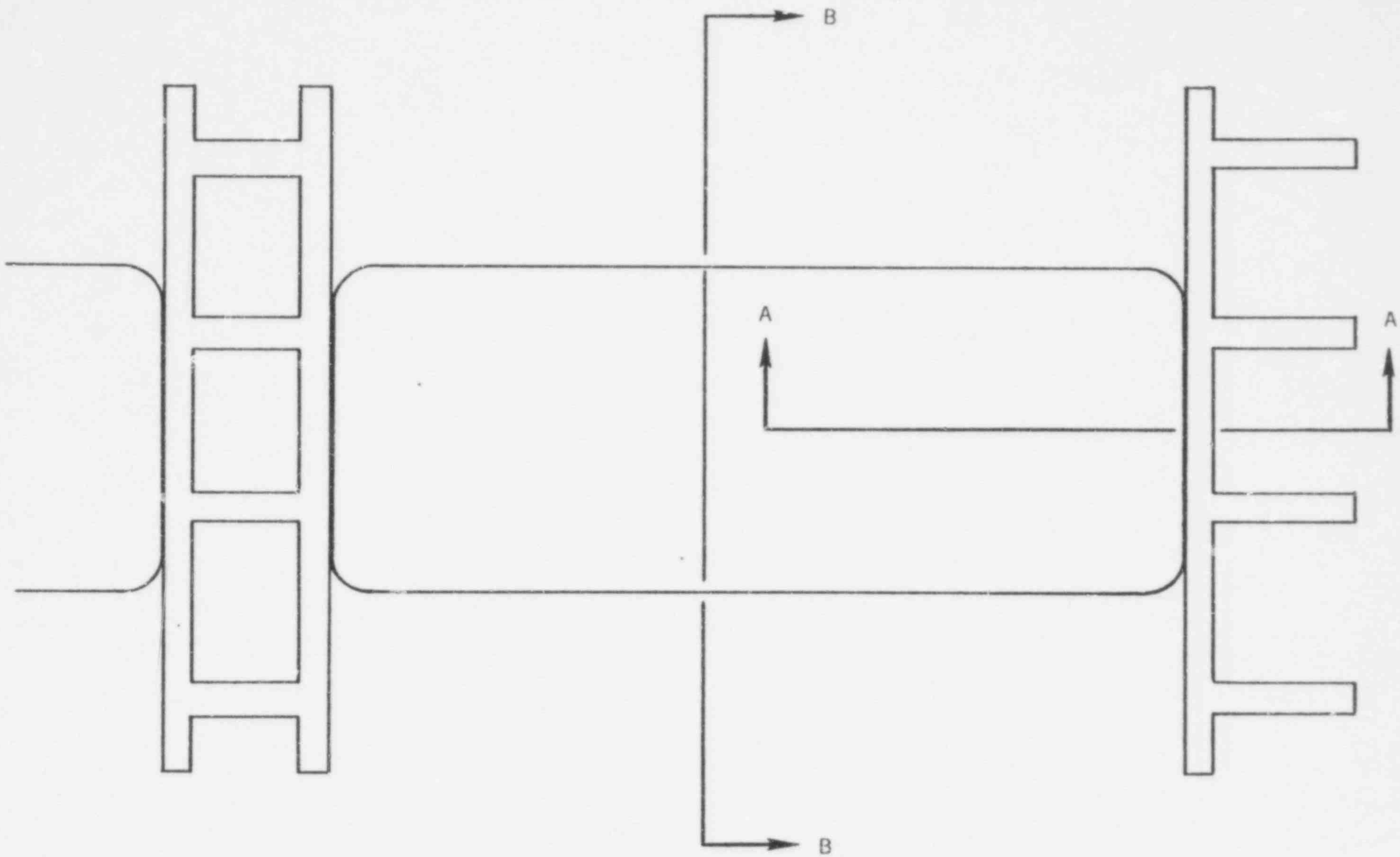
REQUEST NO. 8

Provide the height above finished grade of the tops of the walls surrounding the water bladders for the fuel flooding system.

Response to Request No. 8

Figure #1 (three views) provides the orientation and dimensions of the FFS water supply reservoirs with respect to grade level and adjacent walls.

321 275

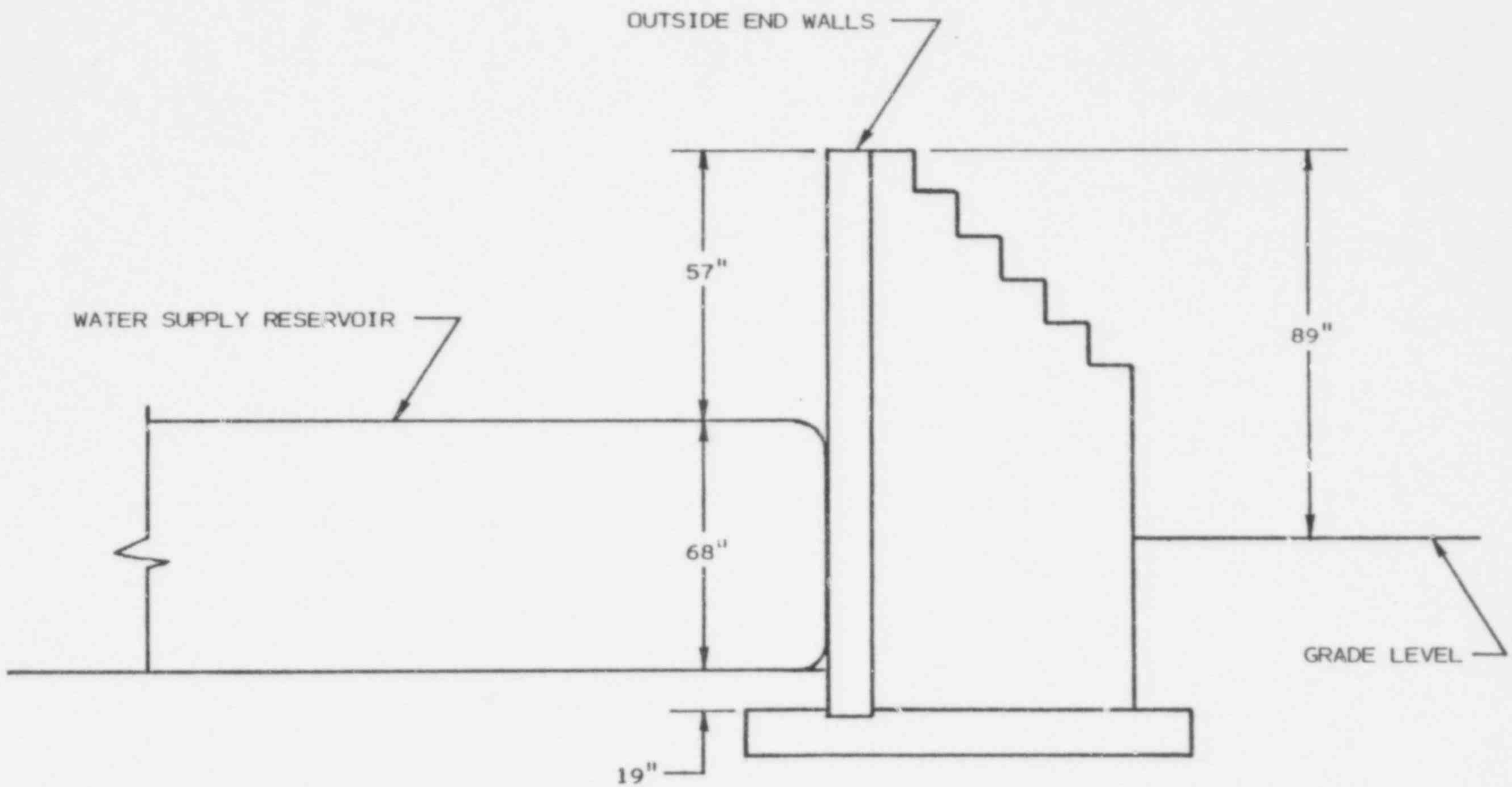


FFS WATER SUPPLY RESERVOIRS

PLAN VIEW

FIGURE 1.

321 276

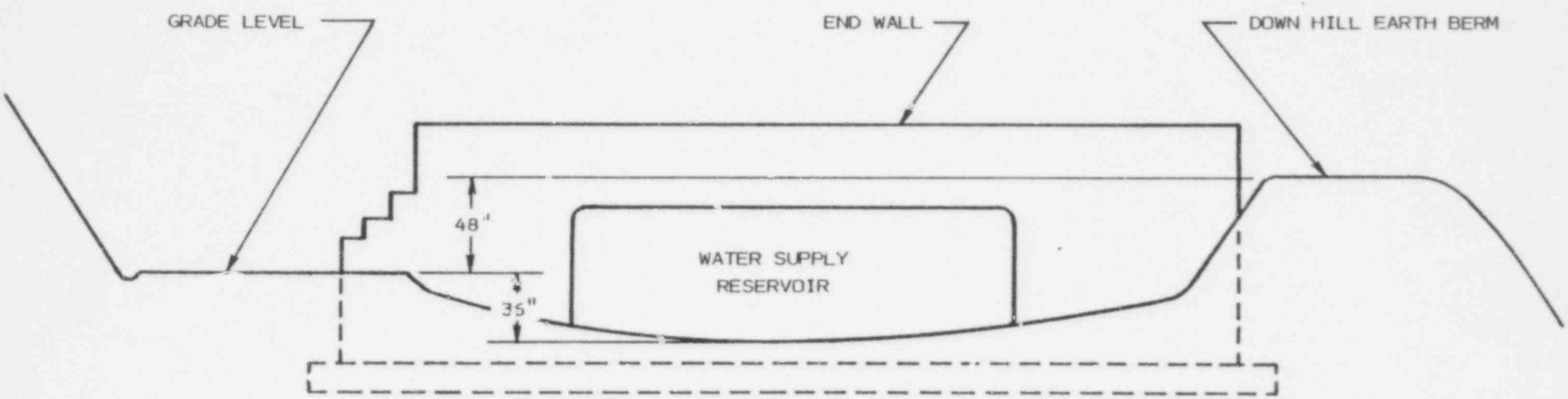


FFS WATER SUPPLY RESERVOIRS

SECTION A-A

FIGURE 1.

321 277



FFS WATER SUPPLY RESERVOIRS

SECTION B-B

FIGURE 1.

321 278

REQUEST NO. 9

> Provide complete details of the automatic level sensing system for the fuel flooding system. Include a discussion of the confidence that can be placed on the functioning of this detection system and the basis for this confidence.

Response to Request No. 9

Each redundant reservoir location contains two interconnected reservoir tanks. The interconnecting piping (located in protected "bag wells" at each reservoir location) contains a standpipe with six ultralow differential pressure sensor switches. These switches will be set to trip on the following conditions:

1. High level - a level set arbitrarily higher than the normal tank level and lower than the maximum recommended by the manufacturer.
2. Normal level - the normal level for the FFS.
3. Normal low level - an acceptable level lower than the normal level.
4. Low level - an acceptable level which provide a warning to refill the supply reservoirs.
5. Compliance level - the lowest level providing the design basis water capacity.
6. Half level - a level to be used during supply reservoir filling.

The level sensing system collects, checks, transmits, receives and displays bag (reservoir) level data. The physical layout of the fuel flooding system (FFS) surveillance instrumentation is shown in Figure 1. The elementary diagram is shown in Figure 2. Data on level switch status goes to the "Data Collector" located at each bag well and is transmitted through a similar "Data Collector" (located at the "penetration wells" adjacent to containment building) to the "Control Room Status Panel" in the reactor control room.

A controller in the bag well "Data Collector" parallel loads a latching shift register with information from the bag level switches and other FFS instrumentation. The controller then shifts the shift registor mode to "Shift" and the data flows to the "Penetration Well Data Collector". The

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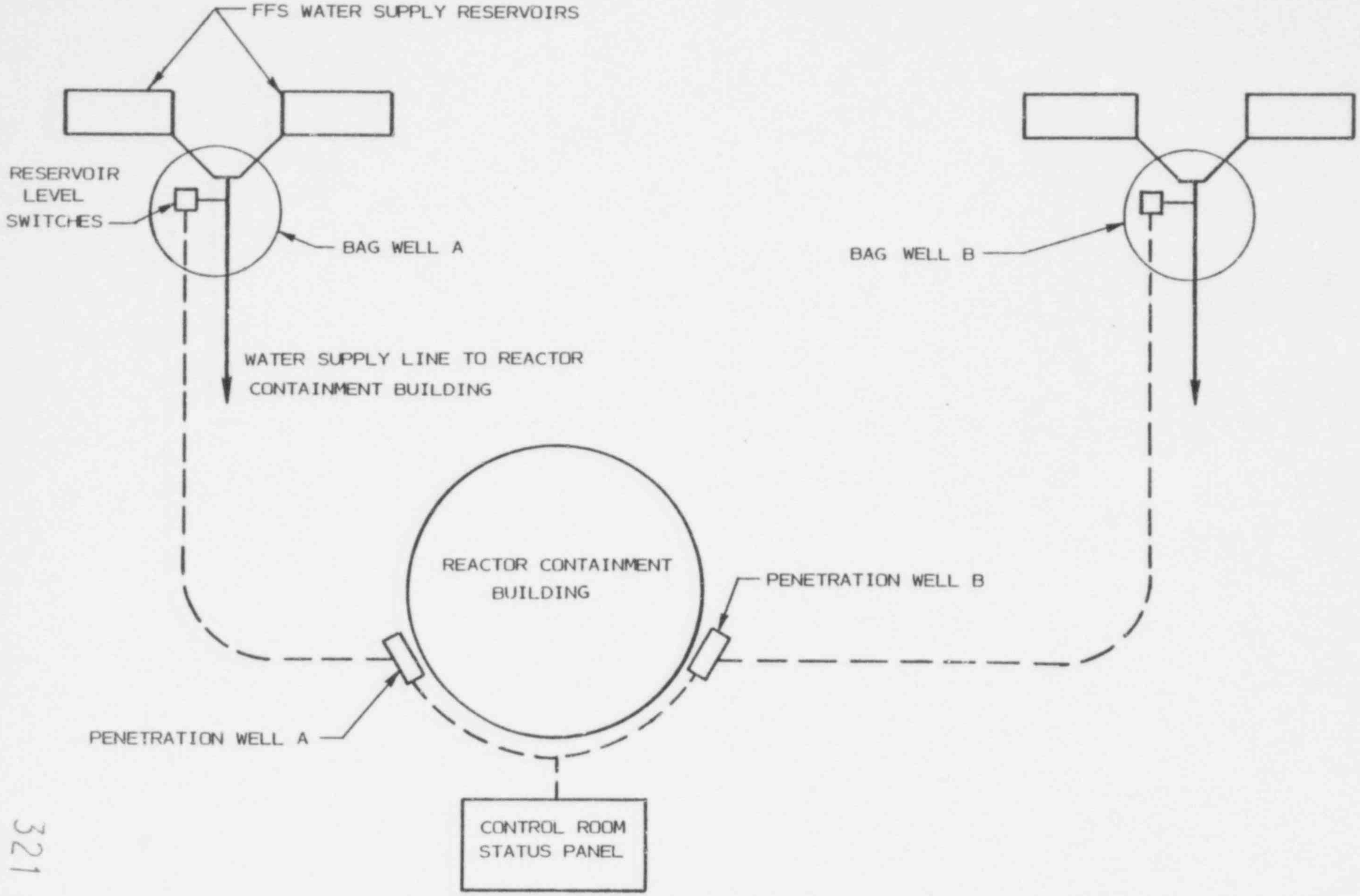
Response to Request No. 9 - continued

penetration well "Data Collector" collects additional local FFS instrumentation information. The controller in the penetration well "Data Collector" senses when the information arrives from the bag well, parallel loads the local information into the shift register, then shifts out both sets of information at approximately 10,000 cycles per second, once per minute. The data in the form of a digital word then goes to the "Control Room Status Panel" where a controller compares the incoming word with the displayed word. When the incoming word is different from the displayed word four times, the Control Room controller latches in the new word and displays it. There is also a data detector that checks that at least one valid word comes in every 10 minutes. A "Control Room Status Panel" diagnostic unit performs an on-line operational check that checks the line, the word generators and the word receivers. A console alarm is activated if any problem is detected so an Instrument repairman can be called.

The level switches are set to open on decreasing level, each switch set for a different level as described above. The normal water level is maintained three switches above the compliance level. Failure of one switch would not result in a non-compliance situation. The system circuit evaluates the incoming signal once every ten minutes. A control room alarm would be activated if the incoming signal were not received because of "Data Collector" or connecting cable failures. An alarm is also activated if the switch contact positions are out of phase. For instance, if the low level switch contacts open (indicating reservoir water below that level) when the normal low level switch contacts are closed (indicating reservoir water above that level) an alarm is activated.

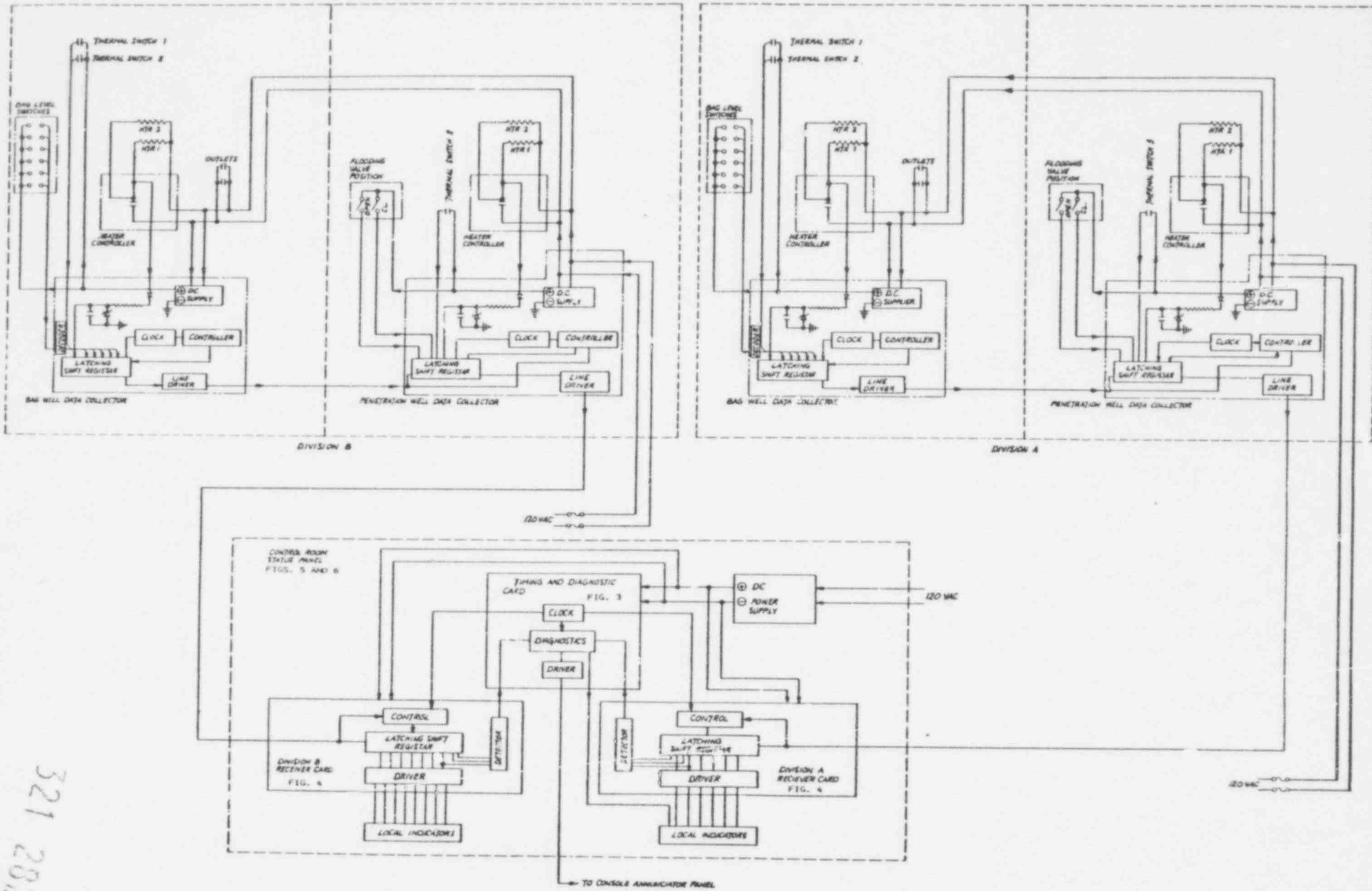
The fuel flooding system supply reservoir level instrumentation has been designed to provide reliable water level indication. Postulated instrumentation failures would either be detected during periodic testing or would activate a trouble alarm. There is high confidence that the system will perform as intended.



FFS SURVEILLANCE INSTRUMENTATION

FIGURE 1.

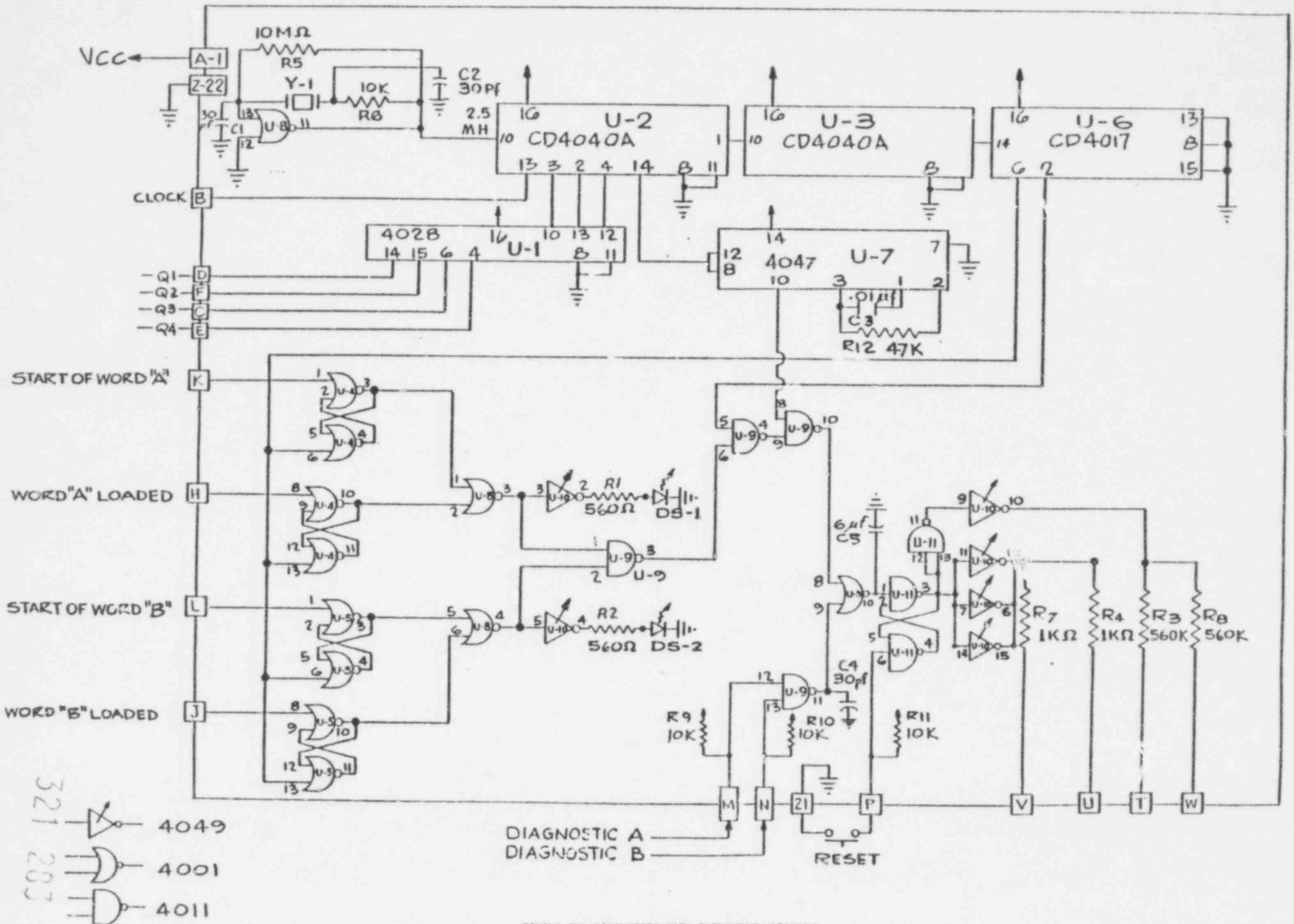
321 281



FFS SURVEILLANCE ELEMENTARY DIAGRAM

FIGURE 2.

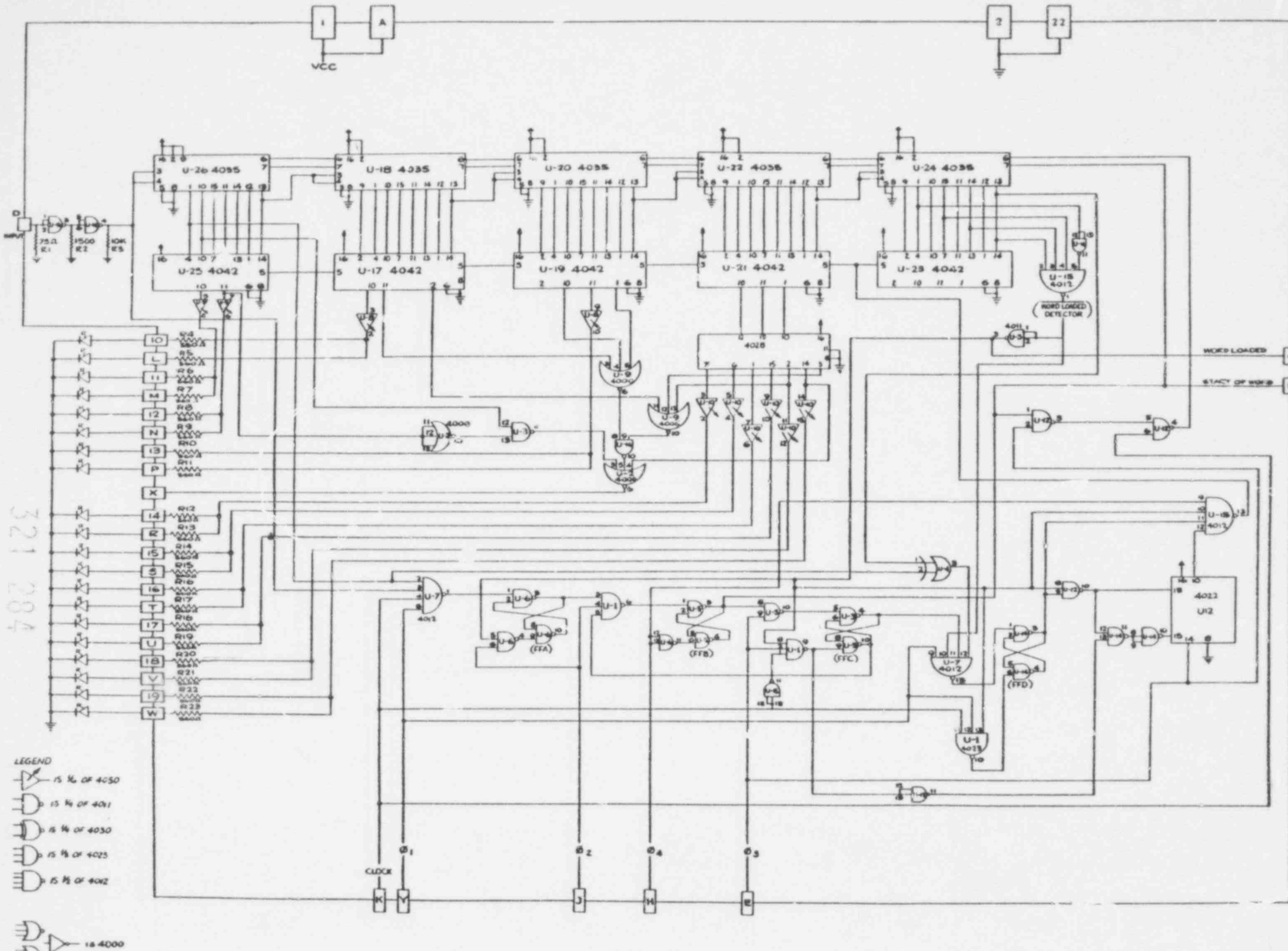
321 282



321 283

TIMING AND DIAGNOSTIC CARD ELEMENTARY DIAGRAM

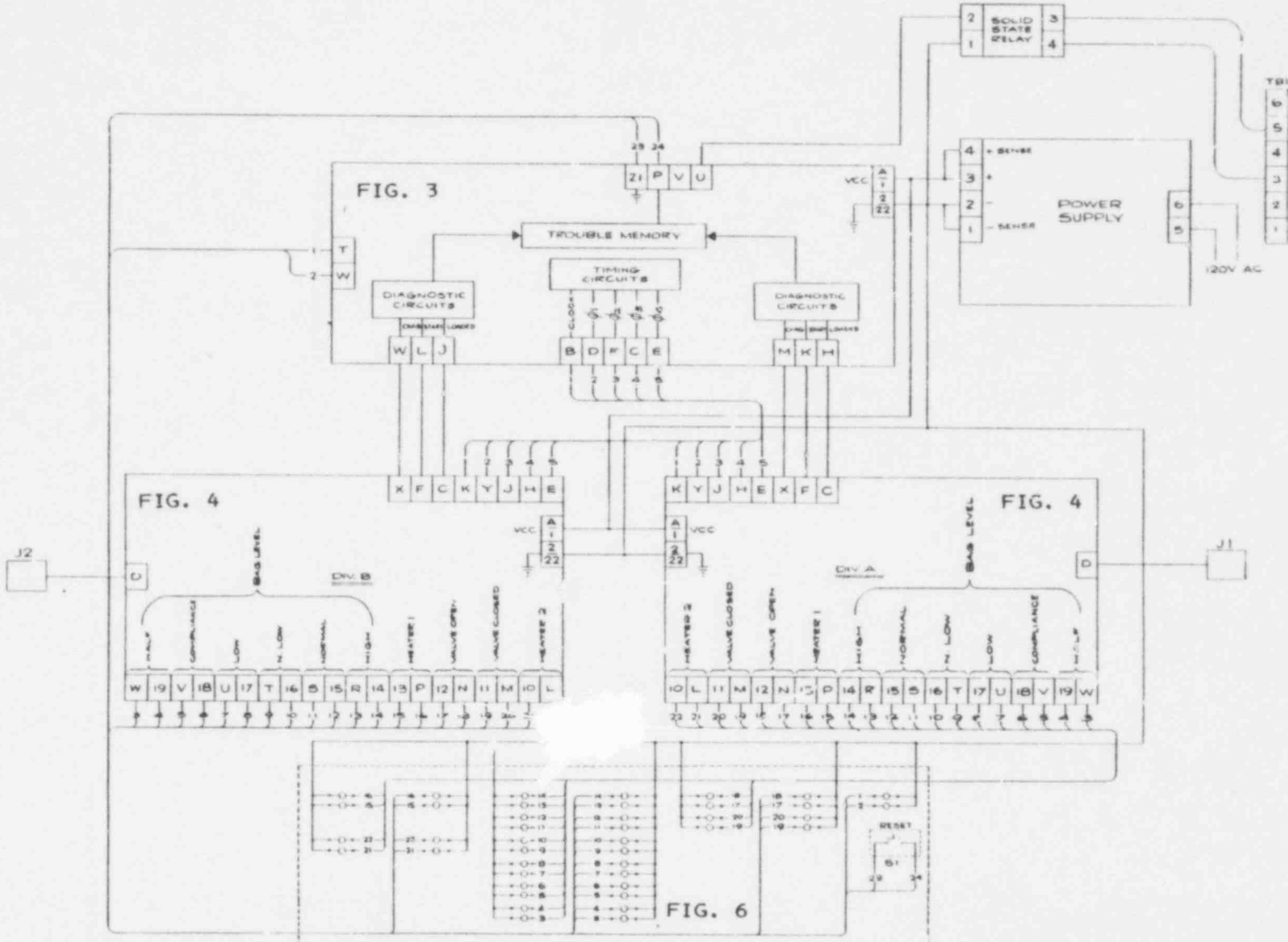
FIGURE 3.



321 284

RECEIVER CARD ELEMENTARY DIAGRAM

FIGURE 4.



STATUS PANEL ELEMENTARY DIAGRAM

FIGURE 5.

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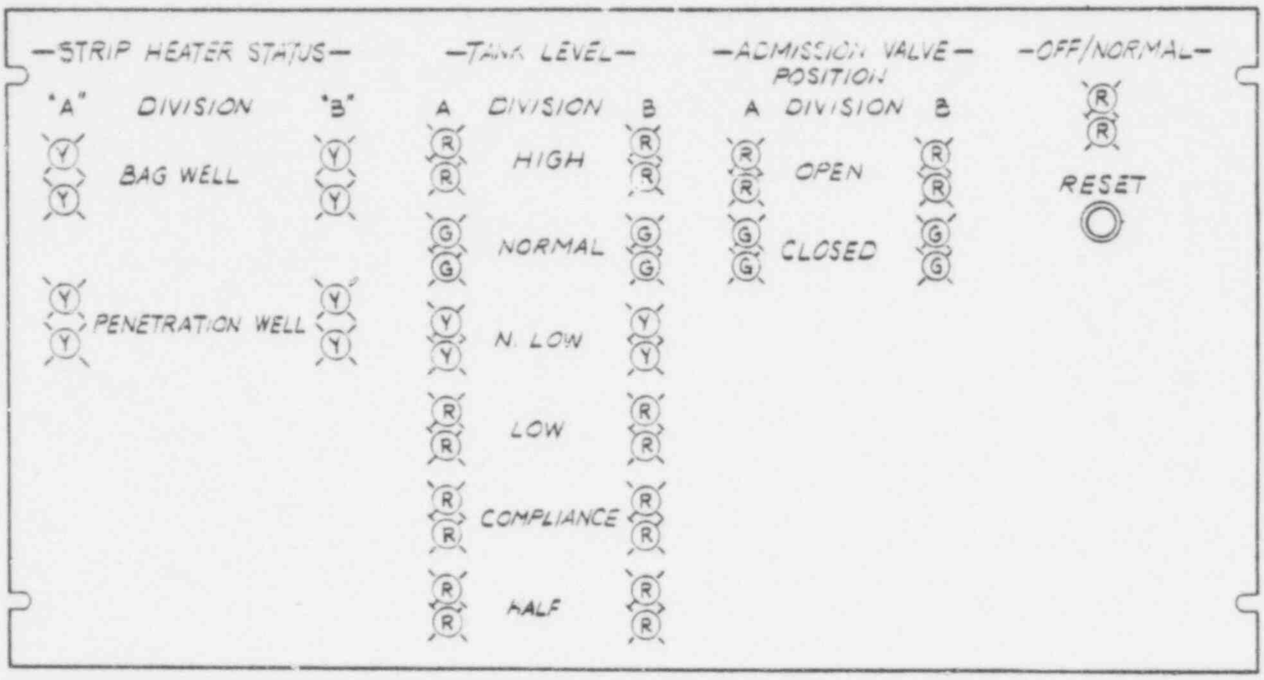

STATUS PANEL

FIGURE 6.

321 206

REQUEST NO. 10

Discuss the surveillance program which will be implemented on the components of the fuel flooding system to assure that they will have the required strength to function subsequent to a seismic event. Focus especially on deterioration of the fuel flooding system hoses and bladders.

Response to Request No. 10

The surveillance program for the fuel flooding system (FFS) components consists of the following tests and inspections:

1. Automatic Valve Operability Test - The FFS automatic valves will be manually operated once per reactor operating cycle (average five weeks) and visually inspected for proper operation and any anomolous condition.
2. Automatic Valve Preventive Maintenance - The FFS automatic valves will be rebuilt on a 10-year frequency.
3. Reservoir Water Sample - Reservoir water samples will be analyzed when the tanks are initially filled, one month after filling, six months after filling, and annually thereafter. Acceptance criteria is not yet established, but will be based on trends rather than quantitative criteria.
4. Anti-Siphon Valve Test - Each FFS anti-siphon valve will be tested annually to verify that a siphon break is accomplished in the respective division piping. A siphon will be initiated and the siphon break confirmed by visual observation. Untested valves will be temporarily plugged so that each redundant valve will be tested.
5. System Flow Test - The FFS automatic valves will be manually tripped and the water flow to the pool and canal for each division will be measured for proper values. The flow test will be performed quarterly the first year and annually thereafter.
6. System Visual Inspection - During the flow test described in #5 above, the FFS line will be visually inspected for leaks from the automatic valve to the pool and canal.

   The FFS lines in the reactor pool will be visually inspected for good condition as part of the Supervisor's Final Core and Pool Inspection Checklist. This inspection will be performed before every reactor cold startup.

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Response to Request No. 10 - continued

A monthly visual inspection will be performed on the remainder of the FFS. The inspection will include general condition, proper connections, water leakage, and other anomalous conditions which could potentially affect the system. Areas to be inspected include the reservoirs, reservoir end walls, reservoir hoses and valve pit, level instrumentation, hose trench, containment building valve panel, penetration, pipe line to the pool and canal, anti-siphon valve, throttle valves and shutoff valves.

7. Standpipe Pneumatic Test - The reactor emergency cooling valve standpipes will be capped biennially and pneumatically pressure tested with the reactor emergency cooling valves closed. This test assures that the standpipes do not leak and primary water would be maintained above the core in the unlikely event that the pool inadvertently drains.
8. Standpipe Inspection - The standpipes and connected FFS hose in the pool will be visually inspected annually to verify good condition and proper connections.
9. Sample sections of fuel flooding system supply line hose will be buried in a similar manner as the supply line. These samples will be inspected annually and tested biennially. The test will consist of pressurizing the hose to the design pressure and applying an axial load until a leak developes which causes the internal pressure to decay. Acceptance criteria have not been established but will be based on the load required to pull the hose out of the trench a total of five meters (600 pounds force). Previous tests demonstrate that the axial failure load (i.e., load which causes onset of leakage) exceeds the pull out load by a factor of six.
10. Sample sections of the water reservoir material will be exposed to the same environmental conditions as the reservoirs. These samples will be inspected and tested for tensile strength annually. Acceptance criteria have not been established but will be based on the stresses in the reservoirs experienced during the postulated seismic event. It has been determined that the tensile strength exceeds the postulated seismic event stresses by more than a factor of three.

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REQUEST NO. 11

Verify that short threaded bolts on the primary piping restraints will be replaced prior to any restart of GETR.

Response to Request No. 11

The project engineer responsible for modifications to the primary piping restraints is (and has been) aware of the short threaded bolts. It is planned to replace the existing bolt assembly (with a new assembly which will provide full thread engagement) before the plant restarts. Installation will be performed in accordance with documented procedures and the existing quality assurance plan. When these (and other) modifications are completed, they will be carefully examined by General Electric and consultant personnel to assure conformance with engineering requirements and analysis assumptions.

321 209

REQUEST NO. 12

> Attachment 1 indicates that during the installation of thirteen (13) out of fifty-six anchor bolts that rebar was encountered and drilled through. Additionally, at some other places it was noted that when rebar was encountered in drilling, the supports wer relocated and holes redrilled without the plugging of the initially drilled holes. Indicate the locations where holes were left unplugged. Discuss the effects of the drilling through of the rebar on the strength of the structure. Also, discuss the potential for and the consequences of moisture contacting the rebar (e.g., corrosion) in the holes containing anchors and in the unplugged holes on the strength of the structure, the anchor bolts, and the overall support. Provide the bases for your conclusions.

Response to Request No. 12

Holes were left unplugged in the regions listed in Table 1. The unplugged holes in the concrete represent a negligible volume and thus do not affect the strength of the structure.

The reinforcing steel, which was partially cut during the installation of the piping and primary heat exchanger bracing anchors, is located in the region of the reactor building where the strength of the walls is conservatively based on only the capacity of the plain concrete. Thus, the wall reinforcing steel which was partially cut does not affect the assumed strength of the concrete used in the reactor building analysis.

Three reinforcing steel bars located in the equipment room floor slab were partially cut. These three bars represent a very small percentage of the total reinforcing steel. In addition, the cut bars are not located in a critical stress area. Thus, the strength of the structure is not affected.

The only remaining question involves the possible loss of anchor bolt load carrying capacity (in locations where the bolts <u>may</u> be in contact with rebar) as a result of potential corrosion effects. Following are reasons why this is not considered an item of concern:

a. For consequential corrosion to occur in these regions, an adequate supply of water and free oxygen must be present. From earlier tests

conducted on the rebar in the primary system equipment room (see General Electric submittal dated November 11, 1977 - Attachment 2, Appendix A), primary water does not, by itself, pose any corrosion threat. In these earlier tests, the rebar was examined at two locations where water seepage was evident. This examination was performed by actually removing the surface concrete and exposing sufficient rebar to allow a good visual examination and photo documentation. The photos in the November 11, 1977 submittal show the rebar to be in excellent condition.

In any case, with the wedge anchors (and restraints) in place, there is no opportunity for consequential quantities of water and air (oxygen) to come into contact with the adjacent rebar. The wedge anchors fit very tightly in their holes at installation (i.e., no holes were over-drilled to make installation easier), and the restraint base plates (which cover the holes and adjacent concrete) eliminate any significant pathways that might otherwise exist.

b. To provide long-term assurance that the anchor bolts will continue to maintain their load carrying capacity, a surveillance program has been established wherein:

   1) The restraint base plates will be visually inspected periodically* to assure they are tight against the wall.
   2) Accessible carbon steel components will be visually inspected periodically* to assure that no significant corrosion has occurred.
   3) All accessible components will be visually inspected periodically* for evidence of wear.

   An additional inspection will be integrated into the surveillance procedure which calls for a periodic* check of the anchor bolt torque setting.

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* Initial inspection frequency will be 1) prior to the time the reactor vessel is reloaded with fuel, 2) after the first power run, 3) six months after startup, and 4) one year after startup. Initially 50% of the restraints will be examined in each inspection. After the initial period (i.e., after one year of operation), the inspection frequency and the number of restraints inspected each time may be altered based upon the previous results obtained.

TABLE 1

UNPLUGGED HOLE LOCATIONS

System/Structure	Location
Primary Pipe Restraints	2 holes near Restraint #1-2.
Primary Heat Exchanger Restraints	1 hole near lower support collar west wall bracket
Pool Heat Exchanger Restraints	2 holes near bottom south wall restraint pad
Pool Heat Exchanger Restraints	1 hole near upper south wall restraint pad
Pool Heat Exchanger Restraints	2 holes near upper west wall restraint pad

321 292

REQUEST NO. 13 (Not included in the NRC memorandum of June 26, but discussed during the site visit of June 18)

Provide information on all tests which have been conducted on the FFS supply line hoses.

Response to Request No. 13

Four documented tests have been performed on the FFS supply line hoses. These tests are the Hose Pull Test, the Hose Tensile Test and two Hose Rupture Tests. These tests are described below:

1. Hose Pull Test - The Hose Pull Test was conducted by burying a hose assembly (i.e., hose and fittings) in a trench approximately 80 foot long. The burial specifications corresponded to the specifications on the installation print.

   The hose assembly was then pul. . through the trench. (See Figure 1) Note that this procedure is a more conservative test of the strength of the hose than pulling the hose up out of the trench (which may occur during postulated surface rupture offset). The hose assembly was pulled a total distance of five meters. The assembly was then visually inspected and hydrostatically tested at the design pressure. There were no i.dications of any damage and the assembly passed the hydrostatic test with no indication of leakage.

2. Hose Burst Test at Ambient Temperature - At the conclusion of the Hose Pull Test (and hydrostatic test) described above, one section of hose with hose fittings was hydrostatically pressurized to failure under ambient temperature conditions. Failure was defined as any leak causing a pressure loss. Failure occurred at a pressure a factor of eight higher than the design pressure and a factor of 12 higher than the maximum normal operating pressure.

3. Hose Tensile Test - The Hose Tensile Test was performed by an independent testing laboratory. This test consisted of pressurizing a hose (with end fittings) which was previously used in the Pull Test

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<u>Response to Request No. 13</u> - continued

described above to the system maximum normal operating pressure and then applying a tensile load to failure. Failure was defined as any leak causing a pressure loss. F ilure did not occur until a force a factor of six greater than the force required to pull the hose through the trench (five meters) was reached.

4. Hose Burst Test at Elevated Temperature - This test was performed by an independent testing laboratory. This test consisted of preheating a hose (with end fittings) which was previously used in the Pull Test described above to 225^{o}F for 24 hours. The hose was filled with water prior to beginning the test. The assembly was then pressurized to failure. Failure was defined as any leak causing a pressure loss. Failure occurred at a pressure a factor of three greater than the design pressure and a factor of four greater than the system maximum normal operating pressure.

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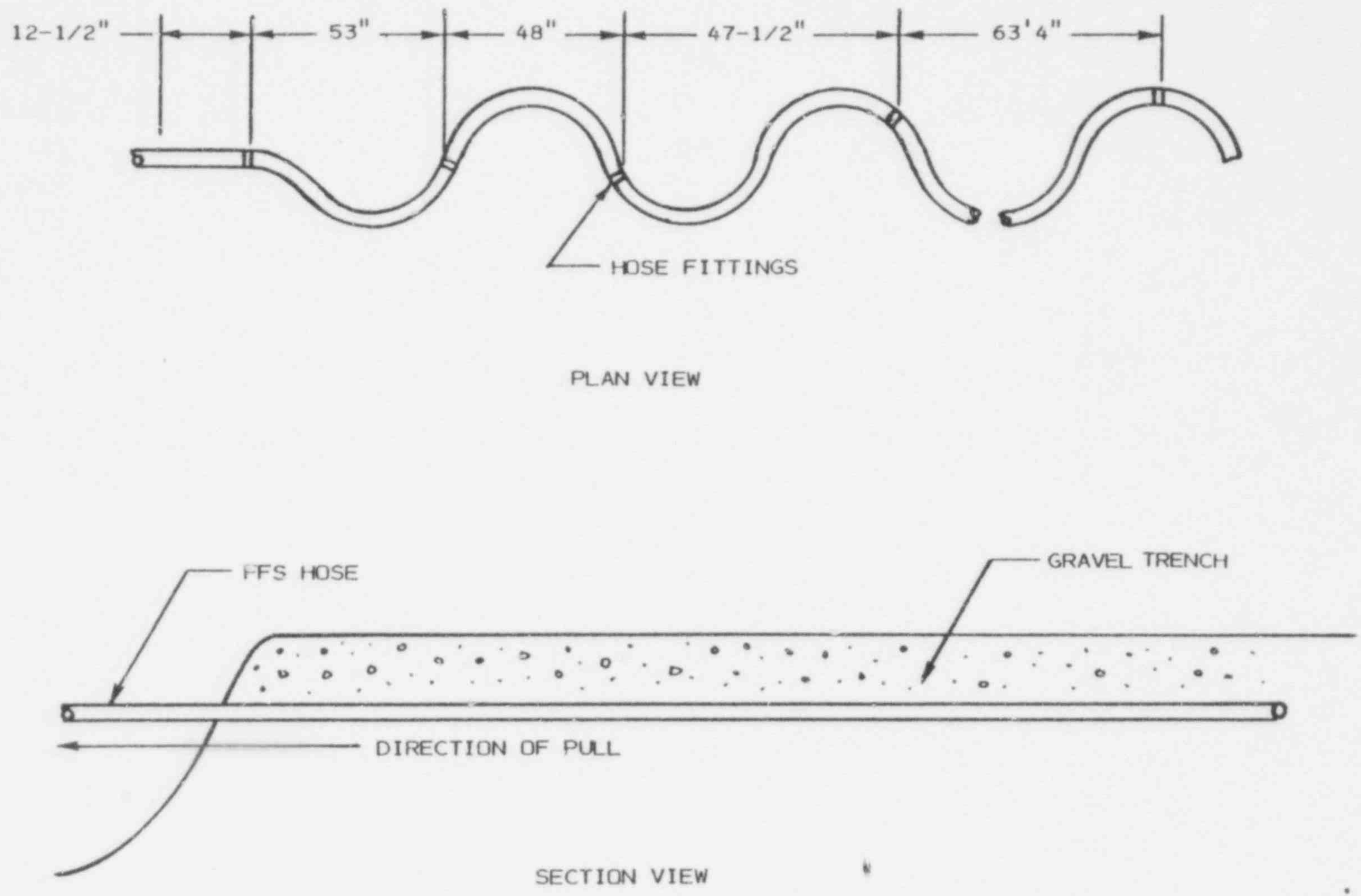

HOSE PULL TEST

FIGURE 1.

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