Georgia Institute of Technology
NEELY NUCLEAR RESEARCH CENTER
900 ATLANTIC DRIVE
ATLANTA, GEORGIA 30332-0425
USA
(404) 894-3600

February 27, 1996

U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N.W.
Atlanta, GA 30323

Reference: Annual Report Docket 50-160; License R-97

Gentlemen:

Pursuant to Section 6.7.a of the Technical Specifications for the Georgia Institute of Technology Research Reactor (License R-97), the following annual report is submitted. The reporting period is January 1, 1995, through December 31, 1995 (calendar year 1995). The designation of the sections below follow the title and order of Section 6.7.a of our Technical Specifications.

1. **OPERATIONS SUMMARY**

   a. Changes in Facility Design

   There were three facility design changes during calendar year 1995. The changes were approved by the Nuclear Safeguards Committee. All design changes are described in Appendix A.

   b. Performance Characteristics

   During the reporting period, the reactor was operated at power levels up to 5.0 MW using a 18-element core. A five element fuel exchange to enhance self protection was performed. The fuel performance, with regard to its ability to contain and isolate fission products, continues to be satisfactory with no known problems. Minor repairs were made on some of the equipment (see Table 1).

040077
9603040314 960227
PDR ADOCK 05000160
R PDR

IE47
1/1

Telex: 542507 GTRIOCAATL
Fax: 404-853-9325 (Verify 404-894-3600)
A Unit of the University System of Georgia
An Equal Education and Employment Opportunity Institution

c. Changes in Operating Procedures

The list of new and/or revised procedures which were approved by the Nuclear Safeguard Committee during calendar year 1995 were as follows:

Proc. #	Title
4902	Corrective Maintenance
3800	Liquid Waste Disposal
7272	Log N Period Amplifier Calibration
7280	MAP-1 Recorder Calibration
7281	Temperature Recorder Calibration - Thermocouple
9013	Calibration and testing of Moving Air Particulate Monitor
9018	Charcoal Cartridge Analysis
9160	Calibration of the LB5100-W Counting System
1500	Irradiated Fuel Discharge to Storage Pool
1501	Lower Top Shield Plug Removal from Spent Fuel
1505	Preparation and Off-Site Shipment of Irradiated Fuel
1506	Physical Protection of Irradiated Fuel in Transit
1507	Emergency Threats to Irradiated Fuel in Transit
1508	Inspection, Testing and Operating Procedure for 6-M Drums
1510	BMI-1 Maintenance, Inspections and Tests
1511	BMI-1 Cask Operating Procedure
1512	Irradiated Fuel Shipment by NAC-LWT Cask

Proc. #	Title
9400	Environmental Monitoring
9501	Control & Accountability of Radioactive Sources

There were two procedures canceled:

4900	System Work Sheet
4901	Preventive/Corrective Maintenance on Safety Related Equipment

d. Results of Surveillance Tests and Inspections

The surveillance tests and inspection of the facility required by the Technical Specifications were performed. Documentation of each of the tests and inspections are available at the site for review.

e. Changes, Test and Experiments Approved by USNRC

There were no changes, tests or experiments that required the approval of the USNRC pursuant to 10 CFR 50.59(a).

f. Current Staff and Nuclear Safeguards Committee Membership

Dr. R. A. Karam, Director, Nuclear Research Center and Reactor Engineer
Mr. Dixon Parker, Reactor Supervisor
Dr. R. D. Ice, Manager of the Office of Radiation Safety
Mr. B. D. Statham, Electronic Engineer (approximately half time)
Mr. Neil Copeland, Senior Reactor Operator
Mr. Johannes Strydom, Senior Safety Engineering Assistant
Mr. Edgar Jawdeh, Health Physics
Ms. Debbie McGeorge
Mrs. Arlene R. Smith

In addition, the NNRC employed the following graduate students on part time basis:

Peter Newby, Senior Reactor Operator
Jeremy Sweezy, Senior Reactor Operator
Dwayne Blaylock, Senior Reactor Operator

Chris Comfort, Reactor Operator Trainee
Ralph Demeglio, Reactor Operator
Nick Jenkins, Reactor Operator
Shane Klima, Reactoı Operator Trainee
Katherin Norton, Reactor Operator Trainee
Tina Weatherman, Reactor Operator Trainee

The current membership of the Nuclear Safeguards Committee is:

(1) Mr. Emsley Cobb, Chairman
Discipline: Reactor Operation and Reactor Safety

(2) Dr. Bernd Kahn
Discipline: Radiation Protection and Environmental Measurements

(3) Dr. Robert Braga
Discipline: Chemistry

(4) Dr. Prateen V. Desai, Secretary
Discipline: Thermal Hydraulics, Mechanical Systems

(5) Dr. Billy R. Livesay, Member
Discipline: Material Science, Physics

(6) Mr. Jack Vickery, Member
Discipline: Security

(7) Dr. Thomas G. Tornabene, Member
Discipline: Biology and Biochemistry

(8) Dr. S. M. Ghiaasiaan, Member
Discipline: Nuclear Engineering

(9) Mr. Len Gucwa, Member
Discipline: Reactor Safety

(10) Mr. Steve Ewald, Member
Discipline: Health Physics

(11) Dr. Peggy Girard, Member
Discipline: Biology and Biochemistry

(12) Mr. James O'Hara, Member
Discipline: Health Physics

2. POWER GENERATION

For the period January 1, 1995, through December 31, 1995, the total power generation of the GTRR was 244.98 MW hours. The reactor was operated a total of 151.6 hours: 21.9 hours at power levels equal to or less than 100 kW, 81.0 hours at power levels 100 kW to 1 MW, and 48.7 hours at power levels above 1 MW.

3. SHUTDOWNS

During this reporting period there were 9 unscheduled shutdowns. Table 1 gives details.

TABLE 1 UNSCHEDULED REACTOR SHUTDOWNS DURING 1995

Number	Date	Scram	Cause	Corrective Action
95-01	1/18/95	Manual Scram	Operators scrammed reactor in response to a D_2O leak alarm caused by a light water leak in the shield system.	Stop Leak added to shield system.
95-02	10/23/95	Negative Period Trip	Operator induced control rod motion caused negative period trip during shutdown.	Instructed operators to move rods more slowly during shutdown procedures.

Number	Date	Scram	Cause	Corrective Action
95-03	10/25/95	Power Trip	The reactor power was 5 MW. The reactor was in auto control. The reg rod was near high limit. Operator instructed trainee to move shim blade to reset reg rod position. The trainee incorrectly shut off auto controller and the shim motion caused trip.	All operators were trained in high power operations. Operators were reminded to carefully watch all trainee actions.
95-04	10/25/95	Control Air Low Pressure Trip	During calibration of the Kanne chamber, electrical noise caused building isolation. All building isolation valves shut, but one air solenoid valve stuck and caused air loss and low pressure in the air system.	All solenoids were serviced and tested. All functioned properly. Problem has not reoccurred.

Number	Date	Scram	Cause	Corrective Action
95-05	10/26/95	Manual Shutdown	Operators shut down the reactor in response to an inoperable regulating rod.	Regulating Rod Drive was inspected. The angie gear was found broken. An exact replacement was found and installed. The reg rod was tested and functioned properly.
95-06	11/1/95	Low Bismuth Coolant Flow Trip	Movement of the biomed shutter caused material to block the bismuth water collection system.	Operators instructed to watch bismuth tank level carefully during operation of the biomed facility.

Number	Date	Scram	Cause	Corrective Action
95-07	11/8/95	Low Ion Chamber Voltage	Flux amp #2 trouble light was on, indicating a problem with the flux amp or the cable.	Flux amp reset immediately. Inspection of the cable and testing of the flux amp and power supply showed no problems. The problem has not reoccurred.
95-08	11/9/95	Negative Period Trip	When an operator was showing a trainee the use of the auto control system the reactor power increased. Operator induced control rod motion to bring the power back down caused the trip.	Training session to emphasize the auto controller use and the proper response to the high power.

Number	Date	Scram	Cause	Corrective Action
95-09	11/17/95	Door Open Trip	While exiting the emergency airlock, the doors open trip occurred.	Operator reprimanded and others were informed of error.

4. UNSCHEDULED MAINTENANCE ON SAFETY RELATED SYSTEMS AND COMPONENTS

There were approximately thirteen (13) minor repairs performed on safety-related systems and components. Records of maintenance performed on components are available at NNRC offices for inspection.

5. CHANGES, TESTS AND EXPERIMENTS

During 1995, there were 36 approved experiments which used the GTRR. The experiments were evaluated prior to their approval with regard to section 3.4 of the Technical Specifications. There were no new experiments which required approval from the Nuclear Safeguards Committee.

6. RADIOACTIVE EFFLUENT RELEASES

a. Technical Specification 6.7.(6)(a) - Gaseous Effluents - Summation of All Releases Via Stack, i.e., ground level release.

(1) FISSION AND ACTIVATION GASES

Tritium Released (gaseous)
Non Measurable

Argon-41 Released

	Total Release (Ci)	Total Avg. Release (μCi/cc)**	Avg. Released over period of reactor operation (μCi/cc)	Max. Inst. Release (μCi/sec)*	% Tech Specs
1st Qtr	7.744	6.15 E-08	1.51 E-05	304	51.97
2nd Qtr	1.922	1.53 E-08	2.68 E-06	114	19.49
3rd Qtr	5.802	4.60 E-08	1.26 E-05	285	48.72
4th Qtr	27.008	2.14 E-07	1.61 E-05	266	45.47
Annual	42.476	8.43 E-08	1.29 E-05	304	51.97

*Technical Specifications release limit is 585 μCi/sec.

**Basis = Stack effluent at 34,000 cfm (1.26 E+14 cc/QTR)

(2) IODINES RELEASED

None Measurable
Lower Limit of Detection <1.15 E-14 μCi/cc

(3) PARTICULATES

None Measurable (LB-5100)
Lower Limit of Detection
gross beta/gamma = <5.32 E-06 μCi
Lower Limit of Detection
gross alpha = <3.45 E-06 μCi

b. Liquid Effluents

(1) TOTAL GROSS RADIOACTIVITY (β/gamma)

	Total Release Ci	Average Release Rate* (µCi/cc)	Maximum Conc. Released (µCi/cc)	% Tech Specs
1st QTR	4.65 E-07	1.70 E-12	2.00 E-08	< 1%
2nd QTR	1.41 E-06	5.14 E-12	7.39 E-08	2.5%
3rd QTR	7.90 E-07	2.88 E-12	2.31 E-08	< 1%
4th QTR	1.01 E-06	3.70 E-12	5.18 E-08	1.7%
Annual	3.68 E-06	3.35 E-12	7.39 E-08	2.5%

* Average release rate values are based on a Georgia Tech campus water discharge rate of 2.743×10^{11} ml/quarter.

(2) TOTAL GROSS RADIOACTIVITY (Alpha)

	Total Release Ci	Average Release Rate[b] (µCi/cc)	Maximum Conc. Released (µCi/cc)	% Tech Specs
1st QTR	<MDA[a]	<MDA[a]	<MDA[a]	< 1%
2nd QTR	6.54 E-08	2.38 E-13	4.67 E-09	< 1%
3rd QTR	7.09 E-08	2.58 E-13	5.46 E-09	< 1%
4th QTR	8.70 E-08	3.17 E-13	7.96 E-09	< 1%
Annual	2.23 E-07	2.04 E-13	7.96 E-09	< 1%

a. Lower than minimum detectable activity
b. Average release rate values are based on a Georgia Tech campus water discharge rate of 2.743×10^{11} ml/quarter.

(3) FISSION AND ACTIVATION PRODUCTS

Cobalt-60 is the only activation product released via the liquid pathway from the reactor facility. The Co-60 does not result from reactor operations, but is attributable to material stored in storage pool that is part of the State of Georgia Radioactive Materials License No. 147-1-SNM. No fission products are released via the liquid effluent pathway.

(i) CO^{60} RELEASE

	Total Release Ci	Average Release Rate[b] (μCi/cc)	Maximum Conc. Released (μCi/cc)	% Tech Specs
1st QTR	2.16 E-05	7.88 E-11	5.57 E-07	1.9%
2nd QTR	<MDA[a]	<MDA[a]	<MDA[a]	< 1%
3rd QTR	<MDA[a]	<MDA[a]	<MDA[a]	< 1%
4th QTR	<MDA[a]	<MDA[a]	<MDA[a]	< 1%
Annual	2.16 E-05	1.99 E-11	5.57 E-07	1.9%

a. Lower than miminum detectable activity
b. Average release rate values are based on a Georgia Tech campus water discharge rate of $2.743*10^{11}$ ml/quarter. Co^{60} Lower Limit of Detection = < 1.44 E-7 uCi/cc.

(ii) TRITIUM

	Total Release Ci	Average Release Rate* (μCi/cc)	Maximum Conc. Released (μCi/cc)	% Tech Specs
1st QTR	1.06 E-02	3.85 E-08	2.90 E-04	2.9%
2nd QTR	6.91 E-03	2.52 E-08	2.49 E-04	2.5%
3rd QTR	1.30 E-02	4.74 E-08	1.95 E-04	2.0%
4th QTR	2.05 E-03	7.47 E-09	8.17 E-05	< 1%
Annual	3.25 E-02	2.97 E-08	2.90 E-04	2.9%

* Average release rate values are based on a Georgia Tech campus water discharge rate of $2.743*10^{11}$ ml/quarter.

(4) TOTAL VOLUME OF LIQUID WASTE RELEASED

1st QTR . . . 5.53 E+07 ml
2nd QTR . . . 3.90 E+07 ml
3rd QTR . . . 1.19 E+08 ml
4th QTR . . . 4.88 E+07 ml
ANNUAL . . . 2.62 E+08 ml

(5) GEORGIA TECH VOLUME OF DILUTION WATER USED

1st QTR . . . 2.743 E+11 ml
2nd QTR . . . 2.743 E+11 ml
3rd QTR . . . 2.743 E+11 ml
4th QTR . . . 2.743 E+11 ml
ANNUAL . . . 1.097 E+12 ml

7. <u>ENVIRONMENTAL MONITORING:</u> (Tech. Spec. 6.7.a(7))

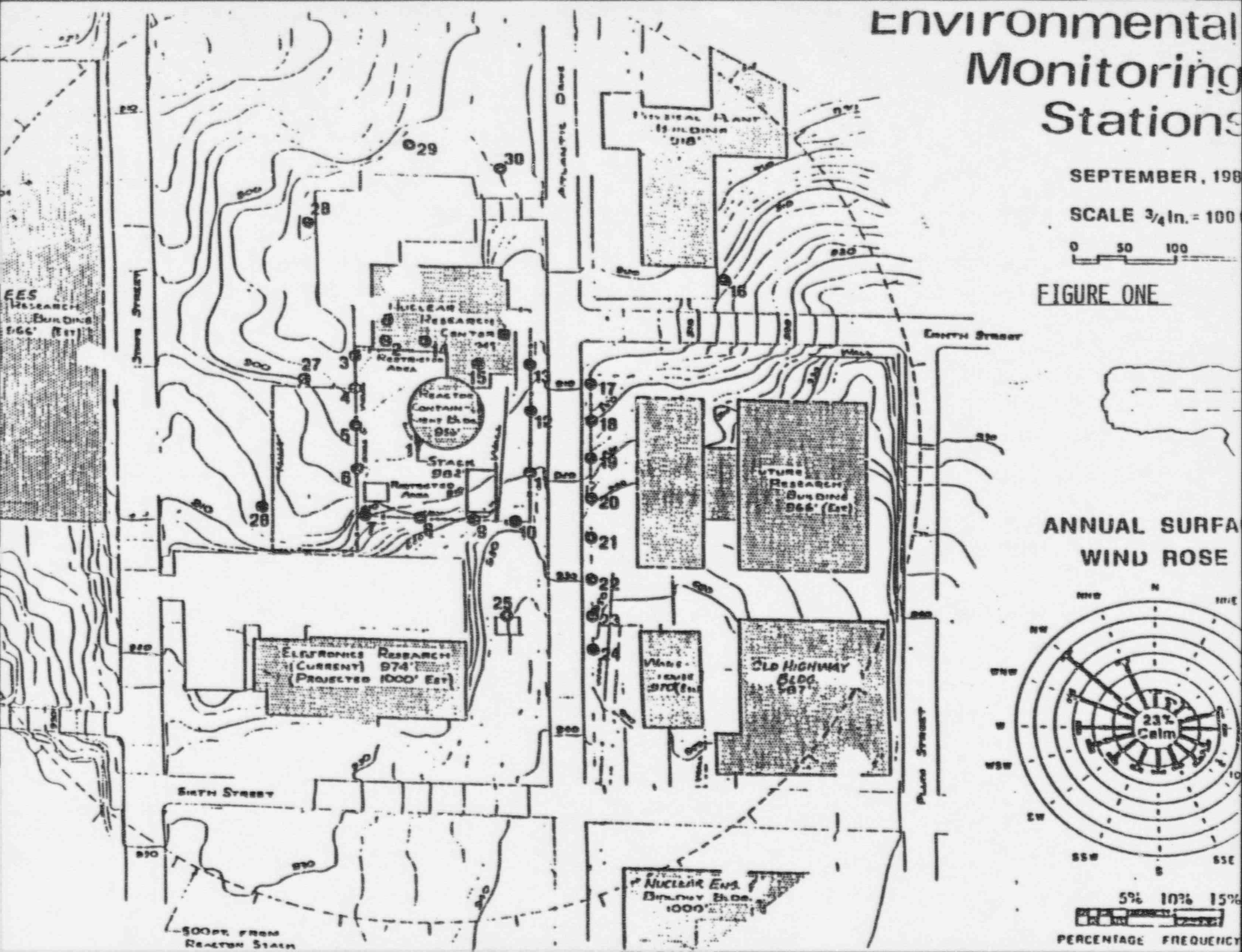
   a. Thirty sites are monitored for environmental radiation. The parameter monitored for Georgia Tech Research Reactor (GTRR) operations is that of direct radiation from the facility and from emitted gaseous effluents (predominantly Ar-41). The location of the sites relative to the reactor are shown in Figure 1, "Environmental Monitoring Stations". The sites are predominantly around the reactor perimeter fence or downwind from the reactor.

   b. Total assays = 30 sites X 4 quarters = 120 assays. The results are reported in the Environmental Radiation Surveillance table (attached). The letter M was used to designate any reading which was less than the minimum detectable limit.

   c. Monitors are Landauer "X9" aluminum oxide thermoluminescent dosimeters (TLD). The dosimeters meet ANSI standards.

      The dosimeters positioned around the facility showed only very low radiation exposure due to the reactor operations. Radiation exposure due to reactor operations is best estimated from TLD #1 positioned inside the reactor building stack. Exposure recorded by this film badge is directly attributable to reactor operations. Because of its location, i.e. inside the reactor building stack, it is not representative of environmental exposures, but rather, represents a worst case exposure.

      Thermoluminescent dosimeter (TLD #9) is located on the perimeter fence near the Georgia Tech Short-Term Radioactive Waste storage and preparation facility licensed by the State of Georgia.



Environmental
Monitoring
Stations
SEPTEMBER, 198
SCALE 3/4 In. = 100
0 50 100
FIGURE ONE
ANNUAL SURFA
WIND ROSE
23% Calm
5% 10% 15%
PERCENTAGE FREQUENCY
EES Research Building 566' (Est)
Nuclear Research Center 941'
Restricted Area
Reactor Containment Bldg. 950'
Stack 982'
Electronics Research (Current) 974' (Projected 1000' Est)
Future Research Building 966' (Est)
Old Highway Bldg. 987'
Nuclear Eng. Biology Bldg. 1000'
Physical Plant Building 918'
Atlantic Drive
Eighth Street
Sixth Street
Plum Street
500 ft. from Reactor Stack

Thermoluminescent dosimeters (TLD's 17 through 24) are closely position to a granite wall. We attribute the majority of exposure to these dosimeters to natural radioactivity in the granite.

Landaurer reports that 8 dosimeters out of 30, averaged over the year, have radiation levels greater than local background. **Note:** The exposures on the Table assume a person at that site for 24 hours per day, 365 days per year.

d. The highest, lowest and the annual average levels of radiation for the sampling point (TLD #9) with the highest average radiation exposure and location of that point with respect to the site.

   Average Annual Level - 15.3 mrem/yr
   Highest Level - 9.7 mrem/qtr
   Lowest Level 4.6 mrem/qtr

e. The gross dose rate readings for all TLDs from all stations varied between 30 and 60 mrem per quarter. The control TLD station varied between 34 and 60 mrem per quarter. This range of variation produced some net dose rate readings (gross reading minus control or background reading) that are negative. The negative readings are replaced by the letter M in the Table. Statistically no conclusions can be made about the environmental dose attributed to the GTRR operation. It is very small.

NEELY NUCLEAR RESEARCH CENTER
ENVIRONMENTAL RADIATION SURVEILLANCE*
1995

	A Jan 1 - Mar 31	B Apr 1 - June 30	C July 1 - Sept 30	D Oct 1 - Dec 30	E 1995 Total Laudauer	F Comments
1	5.7	M	6.5	8.8	5.0	In stack
2	M	M	M	M	M	
3	4.5	M	M	4.3	M	
4	5.6	M	M	4.5	M	
5	3.3	M	0.2	1.7	M	
6	8.1	M	M	2.5	M	
7	4.0	M	1.2	2.7	M	
8	9.0	M	3.2	5.4	11.3	
9	9.7	M	4.6	8.3	15.3	Rad Waste Barn
10	8.4	M	1.6	0.8	1.5	
11	5.2	M	0.1	3.8	0.7	
12	6.6	M	M	5.6	M	
13	9.2	M	6.7	3.4	11.2	
14	0.3	M	M	M	M	
15	M	M	M	M	M	
16	M	M	2.5	4.8	M	
17	10.8	0.4	4.4	4.9	20.4	
18	5.0	M	M	2.6	M	
19	5.2	M	2.0	4.5	M	
20	6.0	M	M	1.3	M	
21	6.6	M	1.7	2.6	M	
22	6.6	M	0.7	Ab	M	
23	6.0	M	M	2.7	M	
24	8.4	M	1.9	6.8	9.9	Granite Wall
25	2.7	M	M	M	M	
26	1.2	M	M	M	M	
27	M	M	M	1.4	M	
28	6.0	M	M	M	M	
29	M	M	M	M	M	
30	1.6	M	Ab	M	M	
Workload MW-HRS	38.84	12.08	34.56	159.50	244.98	

*Sum of natural radiation, direct radiation from facility and gaseous radioactive effluents monitored with Al_2O_3 TLD's less control badge kept at GT Police Dept. Badges processed by Landauer. The lower limit of detection is 0.1 mrem. All negative readings are indicated by M. Absent = Ab.

8. OCCUPATIONAL PERSONNEL RADIATION EXPOSURE 1995:

Radiation workers of Georgia Institute of Technology are monitored through the use of film badges which are provided by a NVLAP certified vendor and have a lower limit of detection of ≤ 10 mrem. A monthly radiation dosimetry report is issued for the personnel of the Neely Nuclear Research Reactor, a summary shown in Table 1.

a. **Summary of exposure for persons under 18 years of age greater than mrem -**

None

b. **Summary of occupational exposures greater than 500 mrem -**

None

c. Person-Rem for the Neely Nuclear Research Center - R-97.

Person-Rem = Sum of occupational workers = 0.490 rem

The highest, lowest and average levels of personnel exposure due to reactor and hot cell operations:

Average annual level - 20 mrem
Highest annual level - 100 mrem
Lowest annual level - < 10 mrem.

d. Category of exposure

NNRC Radiation Workers

Annual Deep Dose	# Radiation workers
< 10 mrem	11
10 mrem - 49 mrem	8
50 mrem - 99 mrem	5
100 mrem - 149 mrem	0
150 mrem - 199 mrem	0
≥ 200 mrem	0

Total Workers 24/490 mrem total

Should there be any questions concerning this report, please let us know.

Sincerely,

R. A. Karam

R. A. Karam, Ph.D., Director
Neely Nuclear Research Center

RAK/dmcg

cc: 1. Dr. Jean-Lou Chameau
2. Dr. John White
3. Members Nuclear Safeguards Committee
4. Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555
5. Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

APPENDIX A

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 95-001

TITLE: Hot Cell Window Level Alarm

1. Will the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report be increased? [yes/no] No

2. Will the possibility for an accident or malfunction of a different type than evaluated previously in the safety analysis report be created? [yes/no] No

3. Will the margin of safety as defined in the basis for any technical specification be reduced? [yes/no] No

4. Is the proposed change an unreviewed safety question? [yes/no] No

NOTE: If additional space is needed to justify conclusion(s) please attach extra sheet(s).

			DATE:
PREPARED BY:	Parker/Newby	[signature]	2/9/95
APPROVALS:			
Director NNRC:		R.A. Karam	2/9/95
Nuclear Safeguards Committee:		[signature]	2/9/95

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: ____________

TITLE: Hot Cell Window Level Alarm

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
	N/A		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
	N/A		

Reviewed By: ______________________ Date: __________

Facility Modification 95-001

Hot Cell Window Level Alarm

Description

The hot cell windows are two zinc bromide filled viewing windows necessary for the operation of the hot cell. The zinc bromide acts as a radiation shield during hot cell experiments. If the level of either window were to drop below the upper steel shielding of the window assembly while sources were present in the hot cell, a beam of radiation would escape through the window possibly endangering the operators or the public. The current window level alarm is connected to the criticality alarm system. The new level alarm will be connected to the criticality alarm system in the same manner. This modification improves the window level detection system and does not change the intended function of the system.

Current Design

The current hot cell window level alarms consist of a float switch assembly in each window. An acrylic float connected to a rod and plate activates a microswitch if the zinc bromide level in either window drops (Diagram 1). The trip level is adjusted by moving the plate up or down the metal rod. The current system has several problems. First, the sensitivity of the current system is not sufficient to ensure that the alarm is activated if only a small drop in window level occurs. Second, the acrylic floats are seriously degraded by the window solution and need frequent replacement. Third, the microswitch is deteriorating due to corrosion caused by the zinc bromide solution. Finally, the float rod guide occasionally sticks in the window plug and could permit a leak of several inches before breaking free and activating the alarm.

New Design

The new system senses the window level by monitoring the conductivity between two copper probes dipping down into the zinc bromide (Diagram 2). Since the walls of the windows are lined in copper, deterioration of the probes is not anticipated. The sensing circuit (see Circuit Diagram) supplies around 25 millivolts and one microamp. This current is not sufficient to deteriorate the zinc bromide solution. The height of the two probes is easily adjusted and the alarm is activated as soon as the solution breaks contact with either probe. Each window will have its own set of probes but uses a common circuit.

The probes will be threaded so that the trip level can be adjusted by tightening or loosening a nut on the probe and observing the probe. The system will be tested by slowly extracting one of the probes from the liquid until an alarm sounds. Operators can check to see if the system is operating by observing LED's on the circuit board. This system will allow for more reliable operation of the window level alarm. Also, this design allows for more accurate control over the trip point.

Failure Modes

The window level alarm is designed to be fail safe. The relay that activates the alarm is normally opened and must be energized to deactivate the alarm. If the power to the circuit or the continuity of any wire is lost the alarm will activate. Also, the probes have been designed so that neither touches the same surface except for the zinc bromide. This will prevent any spilled solution from keeping the circuit closed if the window level should drop. The design also prevents the probes from contacting each other and defeating the alarm.

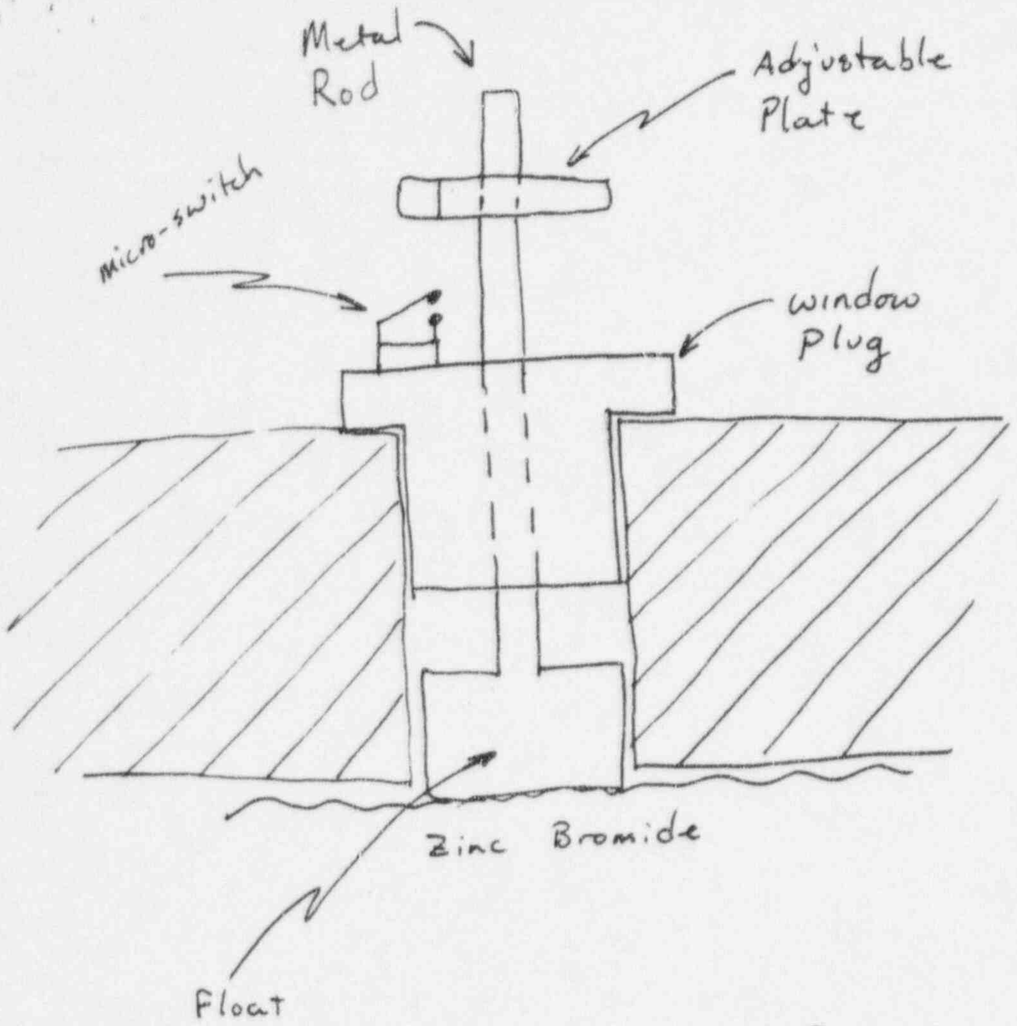


Diagram 1, Current Design

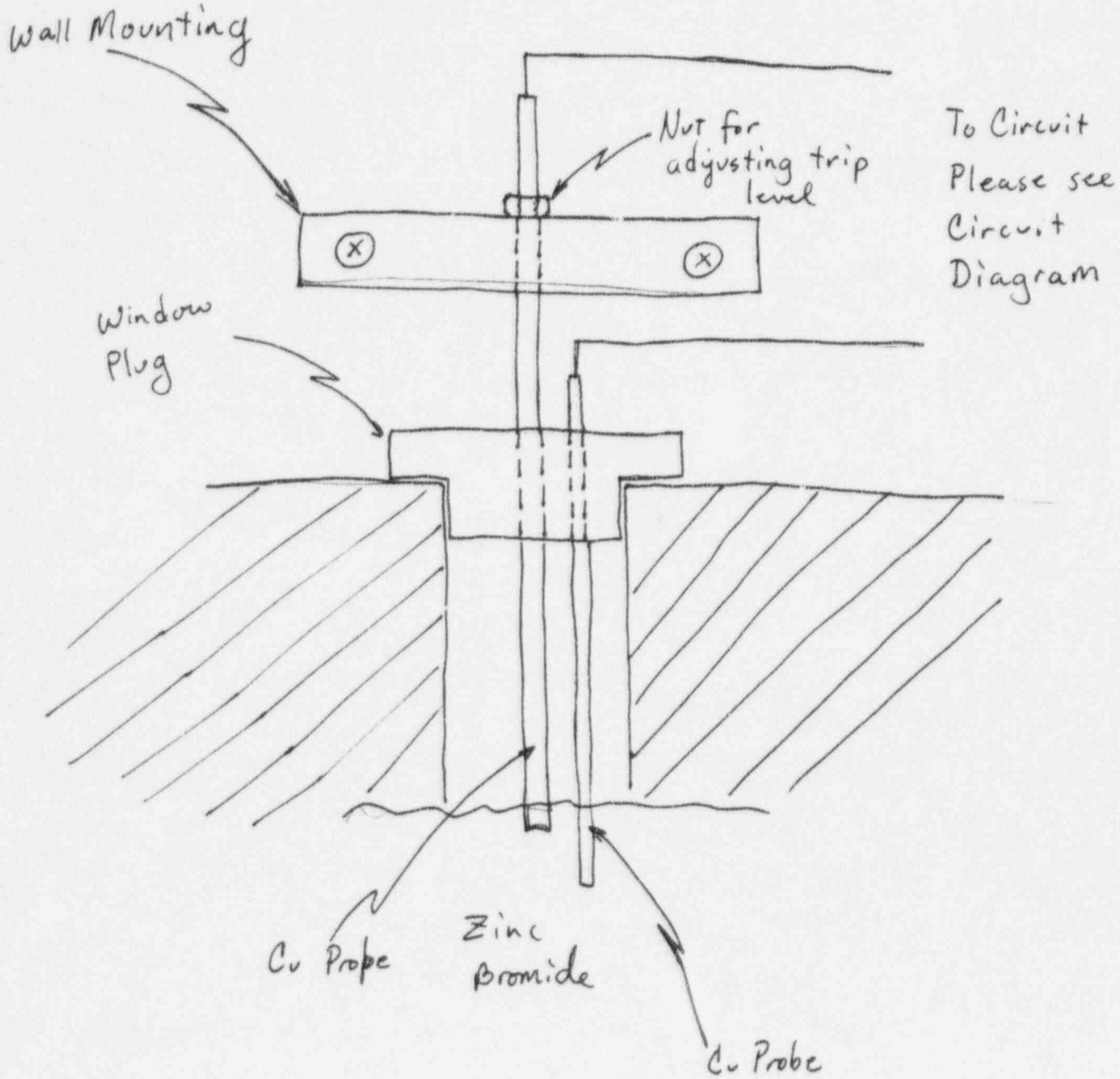
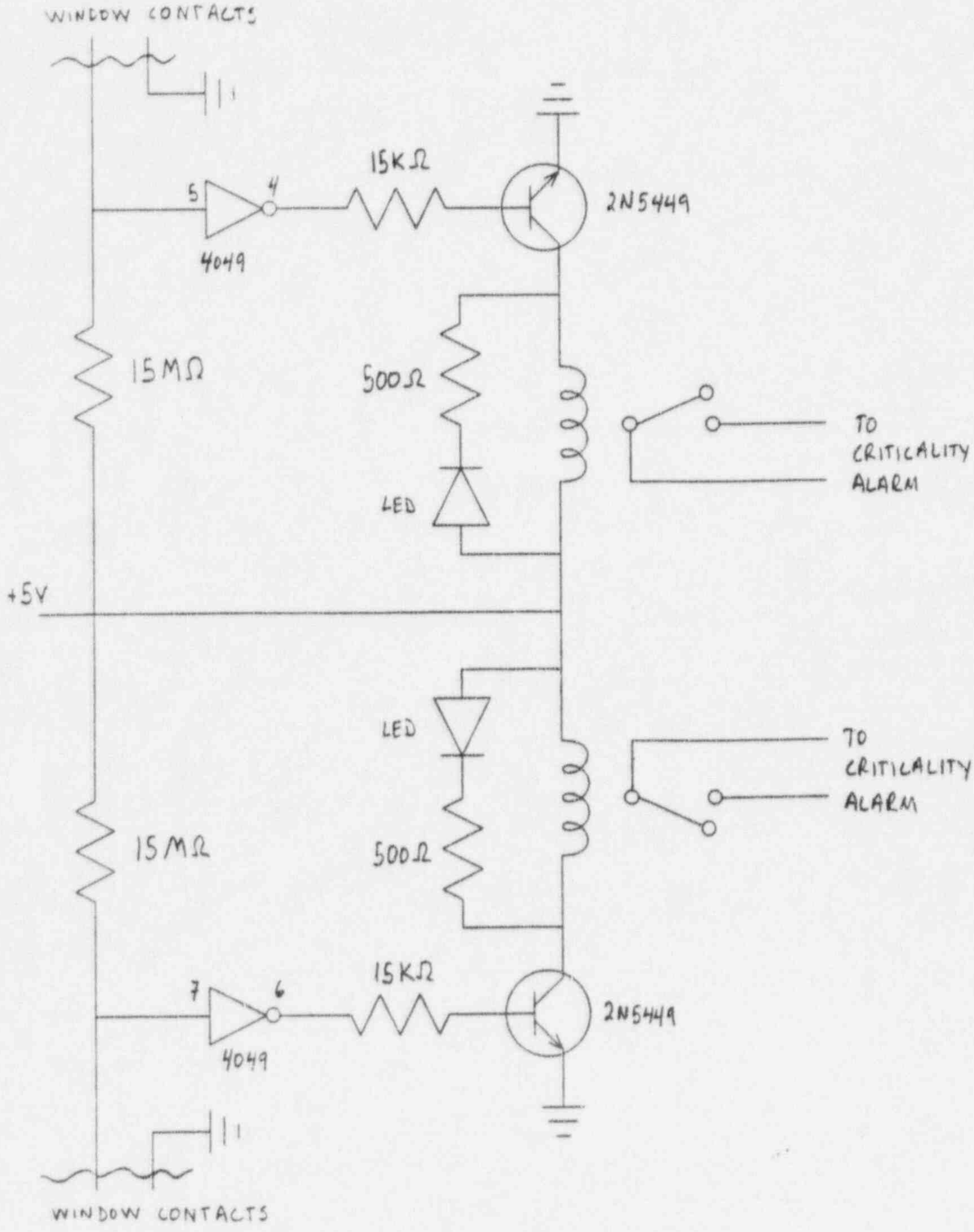


Diagram 2, New Design

Hot Cell Window Alarm System



4049 - V_{CC} = Pin 1
- V_{SS} = Pin 8

Inputs 4 and 6 used
Inputs 3, 9, 11, and 14 to ground

Transistor 2N5449

Relay Magnecraft W171DIP-7

95-001

MEMORANDUM

DATE: 3/6/95

TO: Jerry Taylor, Manager of Hot Cell Operations

FROM: Dixon F. Parker, Reactor Supervisor DFP

SUBJECT: Hot Cell Window Level Alarm

As part of the recent modification of the hot cell window level alarm system the Nuclear Safeguards Committee stipulated that formal testing of the new sensor be performed. Also, you must verify the operability and you familiarization with the system in writing prior to commencing any operation in the hot cell. The system is described in the facility modification package.

Several points to keep in mind while testing the system are:

1. Do not touch the metal part of the middle probe with bare skin as this will ground out the system and prevent the test from working properly.

2. The center probe must not extend below the steel plates on the upper portion of the window. If a window leak occurred this would cause a slit beam to appear prior to the alarm being activated.

3. The side probe will not necessarily give an alarm if it becomes uncovered.

4. The probe level can be easily verified by visually inspecting liquid level in the 1.5 inch hole where liquid is added to the windows. The probes can be seen projecting below the surface of the liquid.

I suggest that you test the system several times to gain familiarity with the sensitivity and adjustment capability of the new probes. Dr. Karam has requested that he be present when you do so. After testing send a memo to file describing what testing actions you have taken, and confirming your acceptance of the system. Also I have attached a training sheet for you to sign. I have given verbal instruction on the system operation to you, Dr. Ice, Peter Newby, and Billy Statham. Please ensure that all of them sign the sheet. Also, I will train any additional personnel that you feel need to be familiar with the system.

pc: Karam, Ice

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 95-002

TITLE: REPLACEMENT OF THE FIRE ALARM TRANSMISSION UNIT

1. Will the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report be increased? [yes/no] NO

2. Will the possibility for an accident or malfunction of a different type than evaluated previously in the safety analysis report be created? [yes/no] NO

3. Will the margin of safety as defined in the basis for any technical specification be reduced? [yes/no] NO

4. Is the proposed change an unreviewed safety question? [yes/no] NO

NOTE: If additional space is needed to justify conclusion(s) please attach extra sheet(s).

		DATE:
PREPARED BY:	Billy Statham	3-21-95
APPROVALS:		
Director NNRC:	R.A. Karam	3/23/95
Nuclear Safeguards Committee:	[illegible]	3/23/95

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 95-002

TITLE: REPLACEMENT OF THE FIRE ALARM TRANSMISSION UNIT

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
	NONE		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
7260	AUTOMATIC FIRE ALARM TESTING NO PROCEDURAL CHANGE NEEDED. THE FUNCTION OF THE REPLACEMENT FIRE ALARM TRANSMISSION UNIT IS SAME AS OLD UNIT.	N/A	

Reviewed By: ______________________ Date: __________

FACILITY MODIFICATION 95-002
REPLACEMENT OF THE FIRE ALARM TRANSMISSION UNIT

1.0 PURPOSE

The purpose of this facility modification is to replace the fire alarm transmission unit.

2.0 SCOPE

The proposal is to replace the fire alarm transmission unit.

3.0 RESPONSIBILITY

The approval for this modification lies with the NNRC director with the concurrence of the Nuclear Safeguards Committee.

4.0 REFERENCES

4.1 Procedure 7260, Automatic Fire Alarm Testing

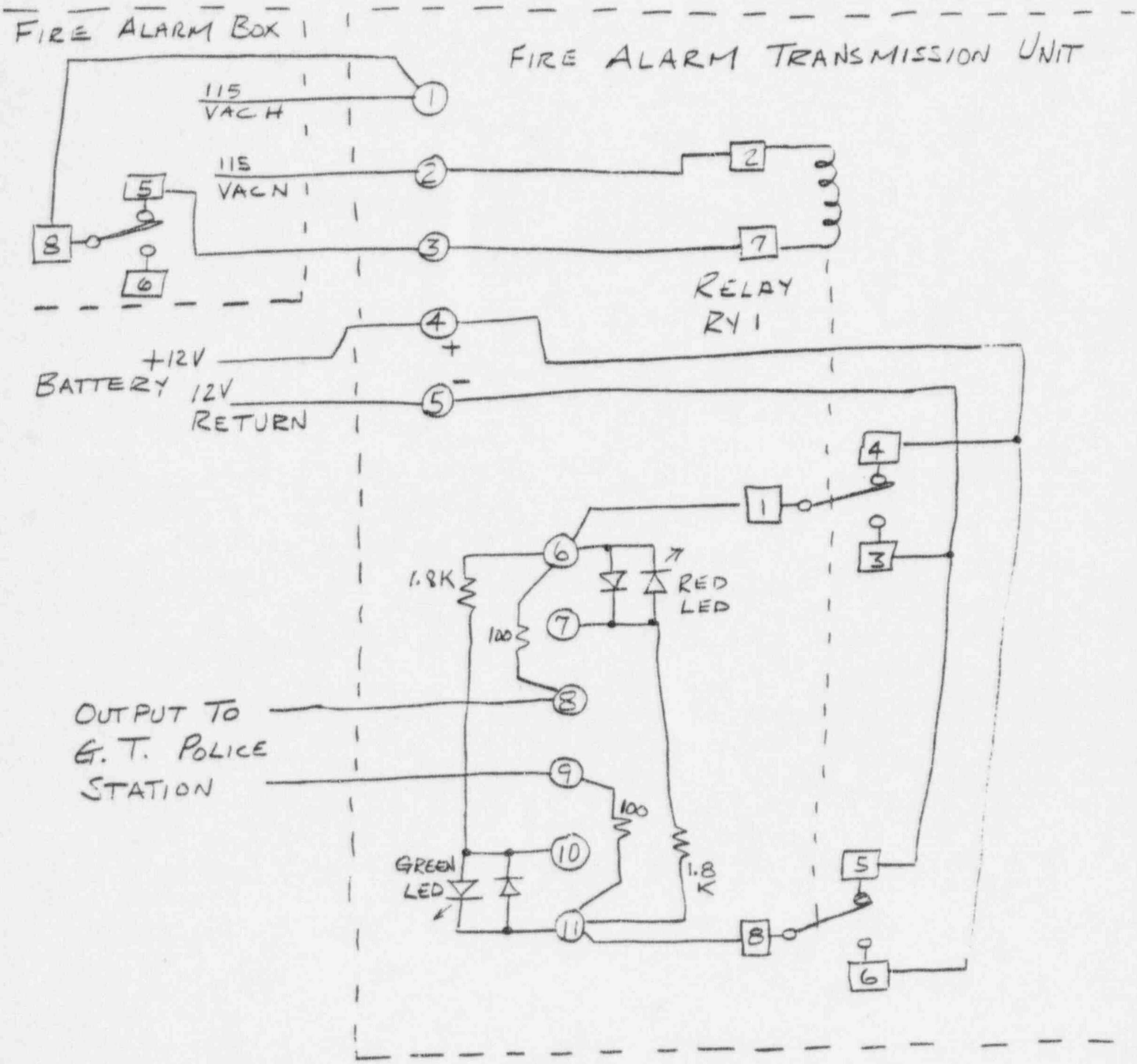
5.0 SYSTEM DESCRIPTION

NOTE: The fire alarm transmission unit sends a signal to the Georgia Tech Police Station (GTPS) indicating the condition of the fire alarm at the Neely Nuclear Research Center (NNRC).

5.1 The old fire alarm transmission unit developed a problem and upon careful inspection of the unit, it was determined that it was not practical to repair this unit.

5.2 The proposed replacement unit consists of one (1) relay, two (2) diodes, four (4) resistors and two (2) LEDs making it very straight forward to repair (if necessary). The relay is a time delay relay, set for two (2) second delay on pull in; this reduces the possibility of sending a false fire alarm to the GTPS during a momentary power interruption. The unit contains a green LED to indicate a safe condition and a red LED to indicate an alarm condition. The power for the signal send to the GTPS is taken from an existing battery supply. The battery supply consists of two (2) 12 volt lead acid batteries and a constant trickle charger (supply is also used to power the PA system in case of loss of utility power). Using the battery power for the GTPS signal prevents a false fire alarm from being transmitted during loss of utility power at the NNRC.

FACILITY MODIFICATION 95-002



ⓧ = TERMINAL STRIP

[Y] = RELAY CONNECT POINTS

RY1 IS TIME DELAY RELAY

BATTERY = DUAL 12V LEAD ACID BATTERIES WITH CHARGE

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 95-003

TITLE: REPLACEMENT OF THE REACTOR PRIMARY COOLANT FLOW INDICATING AND RECORDING DEVICE

1. **Will the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report be increased? [yes/no]** No.

The reliability of the Digital Panel Meter should be greater than the 30 year old flow recorder.

2. **Will the possibility for an accident or malfunction of a different type than evaluated previously in the safety analysis report be created? [yes/no]** No

The proposed system will provide the same functions with greater sensitivity.

3. **Will the margin of safety as defined in the basis for any technical specification be reduced? [yes/no]** No

The margin of safety should be increased with the proposed system because of greater reliability and increased accuracy in the ability to set the scram point.

4. **Is the proposed change an unreviewed safety question? [yes/no]** No

Safety questions will be the same for both systems. The proposed system, providing the same functions, with greater reliability and increased sensitivity has no unreviewed safety questions.

		DATE:
PREPARED BY:	Billy Statham	7-18-95
APPROVALS:		
Director NNRC:	R A Karam	7/20/95
Nuclear Safeguards Committee:	[signature]	7-20-95

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 95-003

TITLE: REPLACEMENT OF THE REACTOR PRIMARY COOLANT FLOW INDICATION AND RECORDING DEVICE

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
045-62-001 SHEET 1 OF 4	INSTRUMENTATION AND CONTROL SCHEMATICS		
045-62-001 SHEET 2 OF 4	(SAME)		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
7277	D2O FLOW RECORDER CALIBRATION		
2002	REACTOR OPERATIONS - PRECRITICAL STARTUP CHECKLIST AND SHIFT SUPERVISOR APPROVAL		
2006	REACTOR SHUTDOWN CHECKLIST		

Reviewed By: R.A. Karam Date: 7/20/95

FACILITY MODIFICATION 95-003
REPLACEMENT OF THE REACTOR PRIMARY
COOLANT FLOW INDICATING AND RECORDING DEVICE

1.0 PURPOSE

The purpose of this facility modification is to replace the Reactor primary coolant flow recorder with a Digital Panel Meter and a strip chart recorder.

2.0 SCOPE

The proposal is to replace the Reactor primary coolant flow indicating and recorder system.

3.0 RESPONSIBILITY

The approval for this modification lies with the NNRC director with the concurrence of the Nuclear Safeguards Committee.

4.0 REFERENCES

4.1 Omega DPF700 Operator's Manual

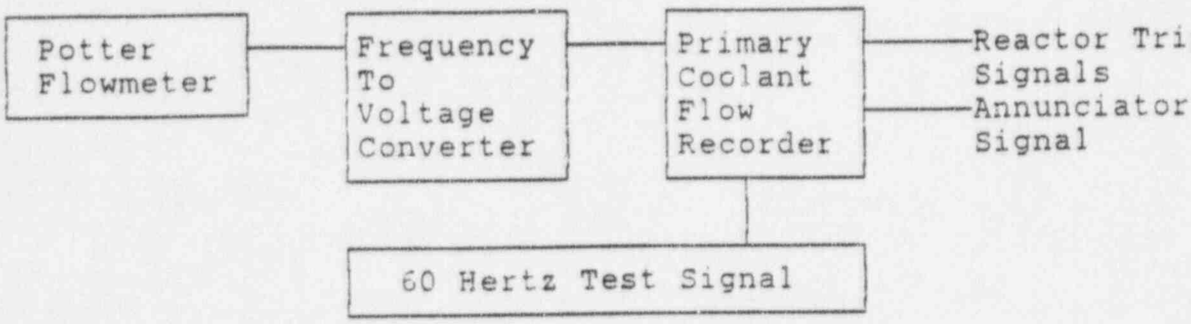
4.2 Omega Operator's Manual for Model 620 Strip Chart Recorder

5.0 SYSTEM DESCRIPTION

5.1 The existing system has a Potter flowmeter which generates a signal whose frequency is proportional to the primary coolant flow rate. An Acromag frequency to voltage converter changes the frequency signal to a DC millivolt signal. A zero (0) to ten (10) millivolt recorder is used to indicate and record this DC millivolt signal. The recorder has a two (2) cam actuated switches, one for generating a reactor trip signal and another for generating the Low D_2O Flow annunciator signal. The recorder has a relay that generates a reactor trip signal should the power to the recorder be turned off. A 60 Hertz line frequency signal can be applied to test the system (providing ~ 440 GPM flow indication).

5.2 The replacement system will utilize the existing Potter flowmeter. An Omega digital panel meter (DPF700) equipped with a dual relay board and analog output board will be used as the indicating device. One relay on the dual relay board will generate both the reactor trip signals; the second relay will generate the annunciator signal. The analog output board will generate a signal for one (1) channel of a dual channel flow recorder (Omega Model 620 strip chart recorder). A test signal that is near the operating range (~1790 GPM flow) will be provided. The second recorder channel is for future secondary coolant flow recording; this will be a separate Facility Modification.

Existing system



Proposed system