Georgia Institute of Technology

NEELY NUCLEAR RESEARCH CENTER
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February 24, 1992

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, 1991 through December 31, 1991 (calendar year 1991). 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 1991: one involving the annunciator panel flasher, another involving the power level measuring channels, picoamreter #1 and picoammeter #2, and the third pertains to the upgrade of the cooling tower. All three design changes are described in Appendix A.

b. Performance Characteristics

During the reporting period, the reactor was operated at power levels up to 4.5 MW using a 17-element core. An 8-element fuel exchange to enhance self protection was performed during the reporting period. Fuel performance has continued to be satisfactory with no known problems.

c. Changes in Operating Procedures

The list of new and/or revised procedures which were approved by the Nuclear Safeguard Committee during calendar year 1991 were as follows:

Proc. #	Title
0004	Safeguards Events Log Entries
4501	Fuel Element Self Protection Measurement
H-300	Training Requirements for Hot-Cell Operators
3109	Instructions for Experiment Approvals
6090	Personnel Monitoring After Building Evacuation in Emergency Situations
6100	Emergency Notifications
7245	Reactor Shutdown Margin Determination
9037	Tritium Determination in Urine
9501	Control and Accountability of Radio-active Sources
2603	Response to Loss of Electric Power
2604	Response to Inoperable Control Element
2605	Response to Leaks in Heat Exchanger
2015	Reactor Power Calibration
7245	Reactor Shutdown Margin Determination
7250	Complete List of Set Points for Modes 1 & 2
7220	Containment Building Isolation Test
9017	Stack Grab Samples
2601	Response to Reactor Scram Initiated by a Safety System
2602	Response to an Alarm Annunciator

Proc. #	Title
3600	Special Nuclear Material Inventory
6100	Emergency Modification
9510	Radioactive Material Shipment
2002	Precritical Check List and Shift Supervisor Approval
	Radiation Safety Manual

The list of old procedures which were rescinded and relegated to Systems Manual by the Nuclear Safeguards Committee in 1991 were:

Proc. #	Title
5000	Objectives and Code for Emergency Procedures
5001	Power Trip - Scram
5002	Power Trip No. 2 - Scram
5003	Period Trip - Scram
5004	Period Trip No. 2 - Scram
5005	Magnet Actuator Amplifier - Scram
5006	Low Ion Chamber Voltage - Scram
5007	Calibrate S itches - Scram
5008	Low D_2O Flow - Scram
5009	High D_2 OVER Temperature - Scram
5010	Reactor Tank Low Level - Scram
5012	Drain Valves Open - Scram
5013	No D_2O Overflow - Scram

Proc. #	Title
5014	Doors Open - Scram
5015	Reactor Isolation Valves Not Open - Scram
5022	High H_2O Temperature-Delay Scram
5023	Low H_2O Flow-Delay Scram
5024	Control Air Low Pressure - Delay Scram
5030	Low Shield Coolant Flow - Delay Scram
5031	High Shield Coolant Temp - Delay Scram
5032	Low Bismuth Coolant Flow - Delay Scram
5033	High Bismuth Coolant Temp.-Delay Scram
5052	Building Radiation High-Alarm
5053	Stack Exhaust High Activity - Alarm
5054	Radiation High Vent Duct - Alarm
5055	H_2O High Radiation - Alarm
5056	Low Neutron Count Rate - Annunciator
5057	D_2O Leak - Alarm
5058	Process Room Doors Open - Alarm
5059	ECCS - Alarm
5060	Outside Servo Range - Alarm
5061	Regulating Rod Low Limit - Alarm
5062	Regulating Rod High Limit - Alarm
5063	CW Basin Low Level Alarm
5064	Vent System Low Flow - Alarm

Proc. #	Title
5065	Low D_2O Temperature Alarm
5066	High D_2O Conductivity - Alarm
5067	High D_2O Conductivity No. 2 - Annunciator
5066	Low H_2O Temperature Alarm
5010(5069)	Reactor Tank Low Level - Annunciator
5070	Low Helium Flow - Alarm
5071	Low Nitrogen Level - Alarm
5072	High Nitrogen Level - Alarm
5073	Low Recombiner Temperature - Alarm
5105	Electric Power Failures
5109	Inoperable Control Element of Position Indicator
5120	Moderator Leak in Heat Exchanger

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. Changes in Plant Staff and Committee Membership

Dr. R.A. Karam, Director, Nuclear Research Center
Dr. Betty Revsin, Associate Director and Acting Manager of the Office of Radiation Safety
Mr. B. D. Statham, Reactor Supervisor and Electronic Engineer
Mr. William Downs, Senior Reactor Operator
Mr. David Cox, Reactor Operator (terminated 10/28/9.)
Mr. Dixon Parker, Reactor Operator
Mr. Jerry Taylor, Senior Safety Engineering Assistant
Mr. Edgar Jawdeh, Health Physics
Mrs. Clara Galleshaw
Mrs. Arlene Robinson Smith

In addition to the full time staff, the NNRC employs the following graduate students on part time basis:

Mr. John Hawkinson
Ms. Kathleen Klee
Mr. David Hustead
Mr. Nazih Chbeir

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. James Mahaffey, Vice Chairman
Discipline: Instrumentation and Control, Nuclear Engineering, Reactor Operations

(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. Kent Barefield, Member
Discipline: Organometallic Chemistry

(8) Dr. James Gordon, Member
Discipline: Medicine

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

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

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

(12) Mr. James Ohara, Member
Discipline: Health Physics

2. POWER GENERATION

For the period January 1, 1991 through December 31, 1991, the total power generation of the GTRR was 276 MW hours. The reactor was operated a total of 292 hours: 24 hours at power levels equal to or less than 100 kW, 243 hours at power level 100 kW to 1 MW, and 25 hours at power levels above 1 MW.

3. SHUTDOWNS

During this reporting period there were 4 unscheduled shutdowns. Table 1 gives details.

TABLE 1 UNSCHEDULED SHUTDOWNS DURING 1991

Report	Date	Trip Initiation	Reason for Trip	Corrective Action
91-0	2/15	Operator	Criticality Alarm	None: Accidental trip, pool level setting was being tested when actuation took place.
91-1	4/29	Air pressure low	During the process of unisolating containment building, pressure drops momentarily. This happened when air pressure was near low limit causing pressure to go below set point.	None needed. This is characteristic of the system design.
91-2	7/28	General Power Failure	Georgia Power	None

91-3	10/11	Reactor Tank D_2O level low	LA-D2 pressure switch measures incremental differential pressure in a range of 12 in. H_2O. The switch dead-band is six inches. Consequently it is not sensitive. Density changes in D_2O due to temperature rise often actuates set point.	Evaluating replacement with more sensitive switch.

4. UNSCHEDULED MAINTENANCE ON SAFETY RELATED SYSTEMS AND COMPONENTS

There were approximately thirty 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 1991, there were 39 approved experiments which used the GTRR. The experiments were evaluated prior to their approval with regard to section 3.4 of the Technical Specifications.

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)
None Measurable

Argon-41 Released

	Total Release (Ci)	Avg Release Rate (uCi/cc)	Max. Instantaneous Release (uCi/sec)	% Tech* Specs (uCi/sec)
1st QTR 32%	5.369	3.84×10^{-8}	190	
2nd QTR 18%	11.043	7.89×10^{-8}	106	
3rd QTR 29%	17.497	1.25×10^{-7}	171	
4th QTR 49%	25.304	1.81×10^{-7}	285	

*Computation based on the Maximum Instantaneous Release Rate as evaluated against a TS release limit of 585 uCi/sec.

(2) IODINES RELEASED

None Measurable
Lower Limit of Detection $<2.5 \times 10^{-14}$

(3) PARTICULATES

None Measurable
Lower Limit of Detection gross beta/gamma 6.46×10^{-5}uCi

Lower Limit of Detection gross alpha 6.85×10^{-6}uCi

b. Technical Specification 6.7(6)(b) - Liquid Effluent - Summation of all Reactors (R-97)

1. 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 the spent fuel storage pool that is part of the State of Georgia Radioactive Materials License No. 147-1. No fission products are released via the liquid effluent pathway.

	Total Release	Avg Release* Rate (uCi/cc)	% Tech Specs	Est.Total Error (%)
1st QTR	0.000015	7.50×10^{-11}	< 1%	7.07%
2nd QTR	0.000066	3.37×10^{-11}	< 1%	10.25%
3rd QTR	0.000100	5.00×10^{-11}	< 1%	12.52%
4th QTR	0.000014	7.00×10^{-11}	< 1%	21.01%

*Average release rate values are based on a Georgia Tech campus water discharge rate of 2.09×10^{11} ml/quarter

2. TOTAL GROSS RADIOACTIVITY (β/gamma)

	Total Release (Ci)	Avg Release* Rate (u/Ci/cc)	% Tech Spec
1st QTR	7.30×10^{-07}	3.65×10^{-12}	< 2%
2nd QTR	1.04×10^{-05}	5.20×10^{-11}	< 2%
3rd QTR	3.01×10^{-05}	1.51×10^{-10}	< 2%
4th QTR	7.55×10^{-06}	3.78×10^{-11}	< 2%

*Average release rate values are based on a Georgia Tech campus water discharge rate of 2.0×10^{11} ml/quarter.

3. TRITIUM

	Total Release (Ci)	Avg Release* Rate (u/Ci/cc)	% Tech Spec
1st QTR	0.00831	4.16×10^{-8}	< 1%
2nd QTR	0.00515	2.58×10^{-8}	< 1%
3rd QTR	0.00804	4.02×10^{-8}	< 1%
4th QTR	0.01617	8.09×10^{-8}	< 1%

*Average release rate values are based on a Georgia Tech campus water discharge rate of 2.0×10^{11} ml/quarter.

4. GROSS ALPHA RADIOACTIVITY RELEASED

None Measurable

Lower Limit of Detection - $< 8.7 \times 10^{-6}$ uCi/ml

5. VOLUME OF WATER RELEASED (ml/Quarter)

From Reactor Building

1st QTR . . . 1.55×10^7
2nd QTR . . . 9.27×10^7
3rd QTR . . . 1.38×10^8
4th QTR . . . 3.69×10^7

6. VOLUME OF DILUTION WATER USED DURING EACH QUARTER

From Georgia Tech Campus

1st QTR . . . 2.0×10^{11}
2nd QTR . . . 2.0×10^{11}
3rd QTR . . . 2.0×10^{11}
4th QTR . . . 2.0×10^{11}

ENVIRONMENTAL MONITORING

TECHNICAL SPECIFICATION 5.7(7)

(a) and (b) - The environmental parameter monitored for GTRR operations is that of direct radiation from the facility and from gaseous effluents via a system of 30 film badges positions around the perimeter fence and other similar locations (see Figure 1, Environmental Monitoring Stations).

(c) - The film badge used for environmental monitoring, which is provided by a NVLAP certified vendor, has a lower limit of detection of < 10 mrem.

None of the film badges positioned around the facility showed radiation exposure due to reactor operations. If radiation exposure due to reactor operations were expected to occur, it would most likely be seen in film badge #1 which is positioned inside of the reactor building stack. Therefore, exposure recorded by this film badge would be directly attributable to reactor operations. Nonetheless, because of its location inside the reactor building stack, it would not be representative of environmental exposures, but rather would represent worst case exposure.

Several badges showed radiation exposure above background levels, film badges #14 and 15 being the highest values. Badge #14 is located on the roof of the laboratory building while badge #15 is located on the roof of the hot cell. Exposures registered by these badges as well as badges #2, #9 and #12 are attributable to environmental damage, e.g., rain and excessive heat.

(d) - Highest, lowest and annual average levels of radiation for the sampling point with the highest average radiation exposure due to reactor operations and location of that point with respect to the site -

All of the film badge locations were similar

Average annual level - < 10 mrem
Highest annual level - < 10 mrem
Lowest annual level - < 10 mrem

OCCUPATIONAL PERSONNEL RADIATION EXPOSURE

a. Summary of exposure for persons under 18 years of age greater than 50 mrem -

None

b. Summary of occupational exposures greater than 500 mrem -

None

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

Sincerely yours,

R.A. Karam, Ph.D.,
Director

Neely Nuclear Research
Center

RAK/ccg

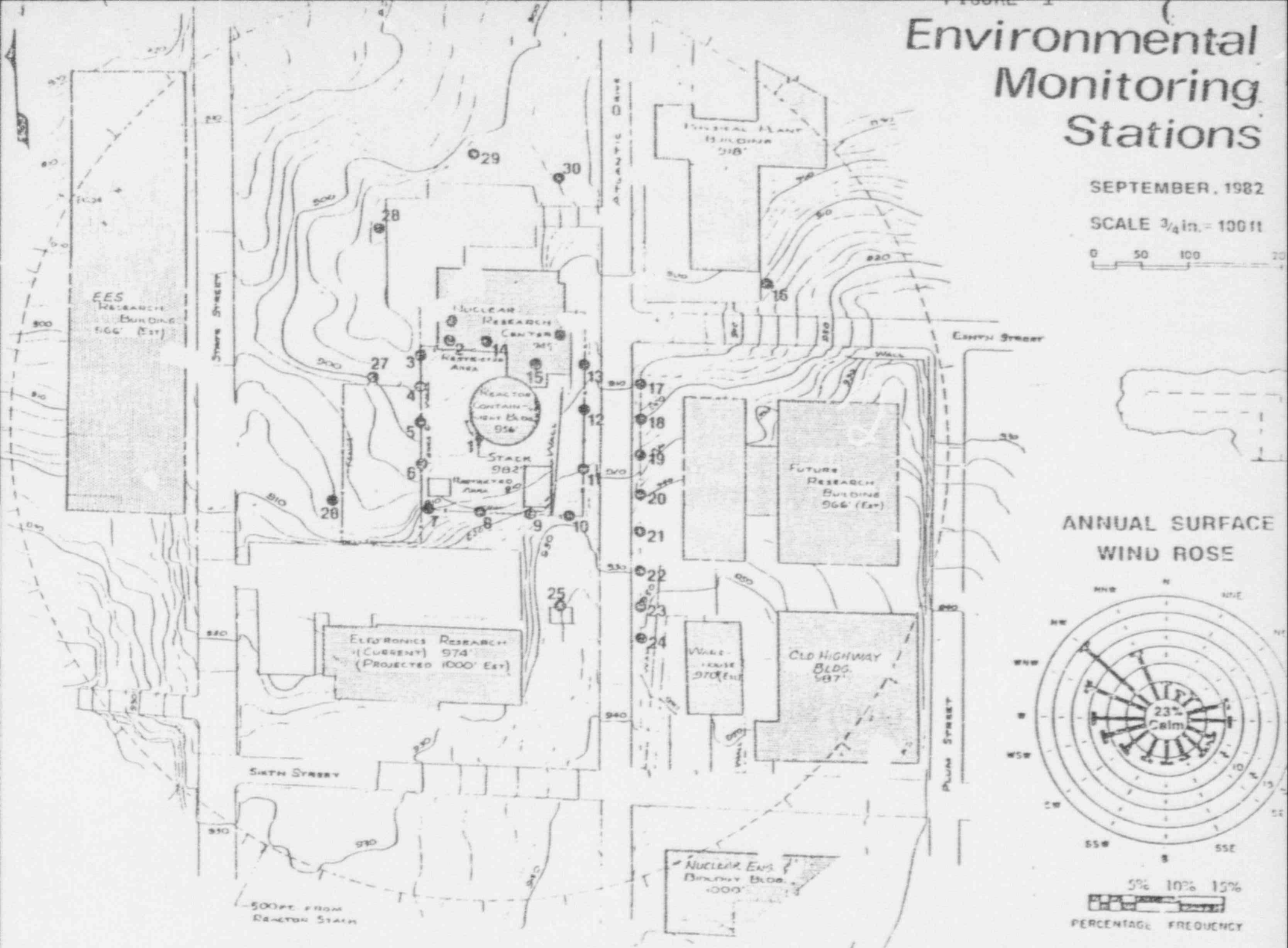
cc: 1. Dr. Gary W. Poehlein
2. Members Nuclear Safeguards Committee
3. Director, Office of Nuclear reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C.
4. Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D. C.

Environmental Monitoring Stations

SEPTEMBER, 1982

SCALE 3/4 in. = 100 ft

0 50 100



ANNUAL SURFACE WIND ROSE

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 91-001

TITLE: FLASHER FOR ANNUNCIATOR PANEL

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	7-31-91
APPROVALS:		
Director NNRC:	R. P. Kaiser	7/31/91
Nuclear Safeguards Committee:	APPROVED 8/1/91	

SUPPORT DOCUMENTATION FOR
FACILITY MODIFICATION NO. 91-001
FLASHER FOR ANNUNCIATOR PANEL

Flasher description:
The annunciator panels contain one flasher. The function of the flasher is to blink the lights of an annunciator in the alarm condition until acknowledged by an operator. In addition an audible alarm sounds until acknowledged; the audible alarm is independent of the flasher.

Condition:
The Scam model ACSF-1 flasher has failed and the lights of the annunciator in the alarm condition are on, instead of flashing.

Direct Replacement flasher:
The Scam Instrument Corporation apparently does not exist in the Chicago area, as evidenced by the failure to get a number from the phone company. A search of Nuclear Equipment Buyers Guide showed no listing for Scam Instrument Corporation.

ACSF-1 flasher operation:
The ACSF-1 includes a motor and a cam operated switch, the switch has both normally open and normally closed contacts. 120 VAC H is connected to one side of the motor (connector pin 7) at all times. When 120 VAC N is connected to the other side of the motor (connector pin 10), the motor will run. The lights of the annunciator in the alarm condition will be on until the normally closed cam switch goes open, the lights then go off and remain off until the switch closes. When the acknowledge button is pressed the motor will stop immediately, if the normally open cam switch is open, otherwise the motor will continue to run until the open position is reached.

Electronic flasher operation:
The electronic flasher contains a CMOS 12 bit binary counter (CD4040), a W232D-3-12 solid state relay (SSR) and a 12 VDC power supply power. In the reset condition terminal S will have approximately 120 VAC H supplied through the audible alarm, this 120 VAC H is converted to + 12 VDC and used to keep the CD4040 in the reset condition. The SSR is turned on while CD4040 is reset. An annunciator alarm condition will cause terminal S to go to 120 VAC N, this allows the CD4040 to count at a 60 Hertz rate. After 64 counts, CD4040 pin 2 goes high and the SSR is turned off. Another 64 counts, CD4040 pin 2 goes low and the SSR is turned on. When the acknowledge button is pressed, CD4040 is reset.

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 91-001

TITLE: FLASHER FOR ANNUNCIATOR PANEL

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
NONE	FLASHER FOR ANNUNCIATOR PANEL	[signature]	7/30/91

PROCEDURES: UPON APPROVAL by NSC Replace Flasher by Electronic Flasher

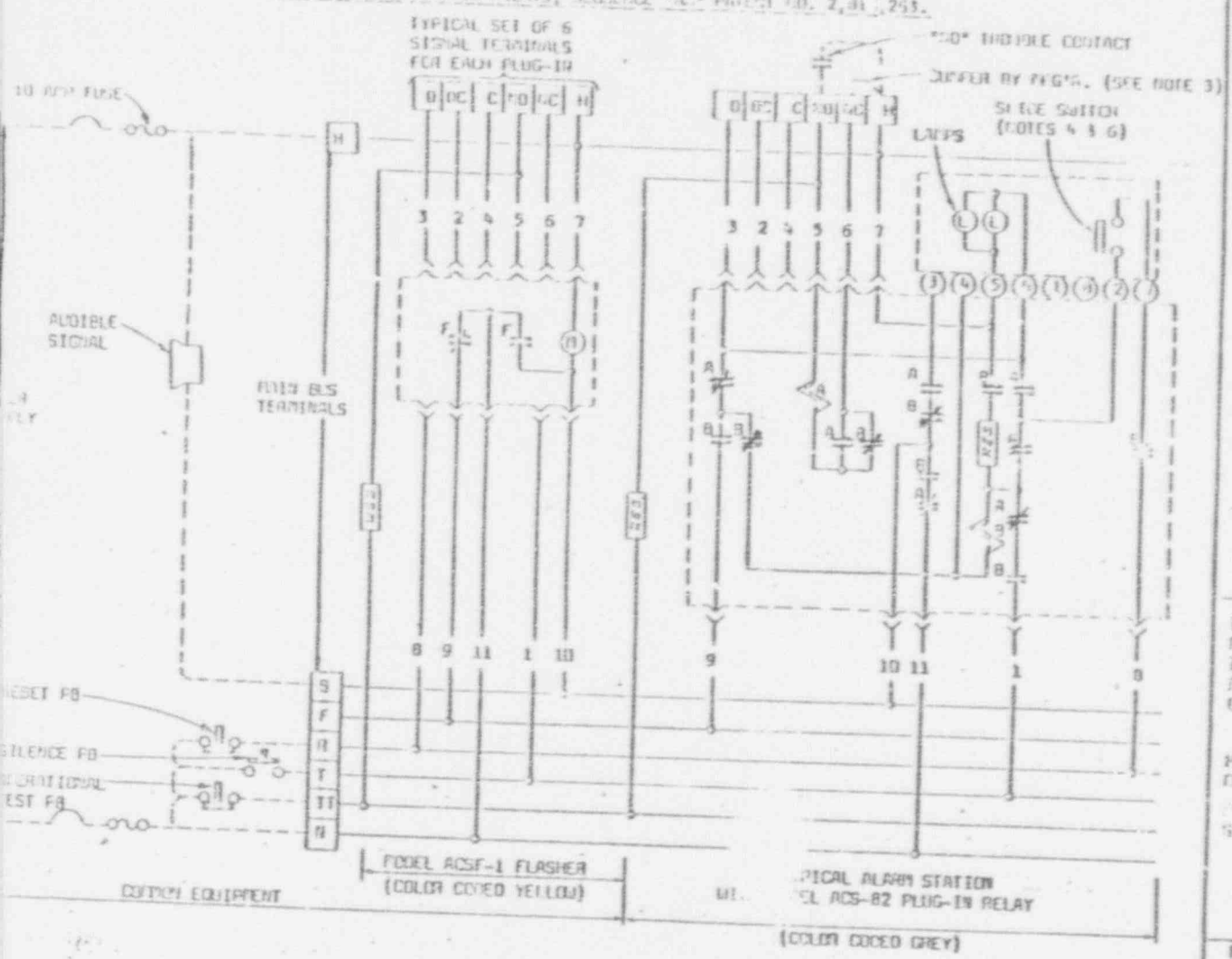
NUMBER	TITLE	REVISED BY	DATE

Reviewed By: [signature] Date: 7/30/91

Drawing AN-1 and AN-2

CAM Instrument Corporation By Thomas Mitsos 1/29/62

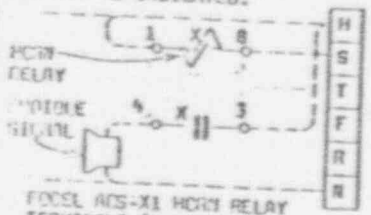
TYPICAL ELECTRICAL DIAGRAM FOR "DC" LINE ARRANGEMENTS
MODEL ACS-82 PLUG-IN RELAYS; SEQUENCE "A" PARTS NO. 2,31 ,253.



NOTES:

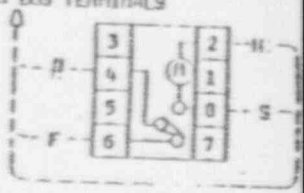
1. ALL RELAYS ARE SHOWN DE-ENERGIZED.
2. MAKE OR BREAK CONTACT RATING FOR NON-INDUCTIVE LOAD IS:
   a) 2 AMPS AT 120/60 VAC.
   b) 65 WATTS AT 125 VDC.
3. TROUBLE CONTACTS: JUMPERS ARE INSTALLED AT EVERY ALARM POSITION BETWEEN TERMINALS (H) AND (NC).
   a) WHEN USING "ND" TROUBLE CONTACTS LEAVE JUMPING IN PLACE AND CONNECT "NO" TROUBLE CONTACTS BETWEEN TERMINALS (H) AND (NO).
   b) WHEN USING "NC" TROUBLE CONTACTS REMOVE JUMPERS AND CONNECT "NC" TROUBLE CONTACTS BETWEEN TERMINALS (H) AND (NC).
4. TEST PROCEDURE:
   a) HOLD OPERATIONAL TEST PB DEPRESSED; ALL STATIONS WILL ALARM.
   b) OPERATE SILENCE PB; ALL STATIONS WILL SILENCE.
   c) RELEASE OPERATIONAL TEST PB:
      1) SEQUENCE AF STATIONS WILL BE NORMAL AGAIN.
      2) SEQUENCE AM STATIONS WILL BE IN READY TO NORMAL.
   d) OPERATE RESET PB; SEQUENCE AM STATIONS WILL RETURN TO NORMAL.
5. LOCK-IN; MOMENTARY ALARMS WILL AUTOMATICALLY BE LOCKED IN.
6. SEQUENCE SELECTION:
   a) FOR SEQUENCE AF; PLACE SLIDE SWITCH IN THE OFF (OPEN) POSITION.
   b) FOR SEQUENCE AM; PLACE SLIDE SWITCH IN ANY CLOSED POSITION.
7. WHEN POWER IS FIRST APPLIED TO THE ANNUNCIATOR, ALL STATIONS WITH SLIDE SWITCH IN THE "CLOSED" POSITION WILL BE IN THE "RETURN TO NORMAL" CONDITION. OPERATE RESET PB TO PUT THESE STATIONS IN THE NORMAL CONDITION.

IF CONTACTS OF ALARM STATION RELAYS ARE OF INSUFFICIENT RATING TO BREAK AUDIBLE SIGNAL CIRCUIT A HORN RELAY MUST BE USED. CONNECT AS INDICATED.

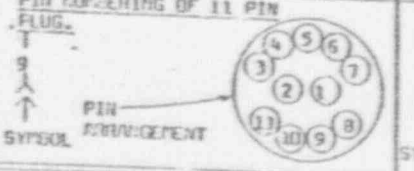


REMOTE FLASHER MODEL ACSF-2

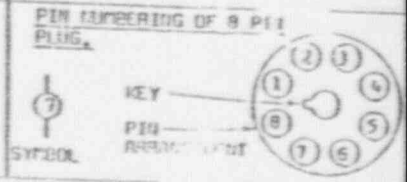
IF THE REMOTE MOUNTED MODEL ACSF-2 FLASHER IS USED INSTEAD OF THE INTEGRAL MODEL ACSF-1, CONNECT AS SHOWN.



PIN NUMBERING OF 11 PIN PLUG.



PIN NUMBERING OF 8 PIN PLUG.



ALARM SEQUENCE CHART

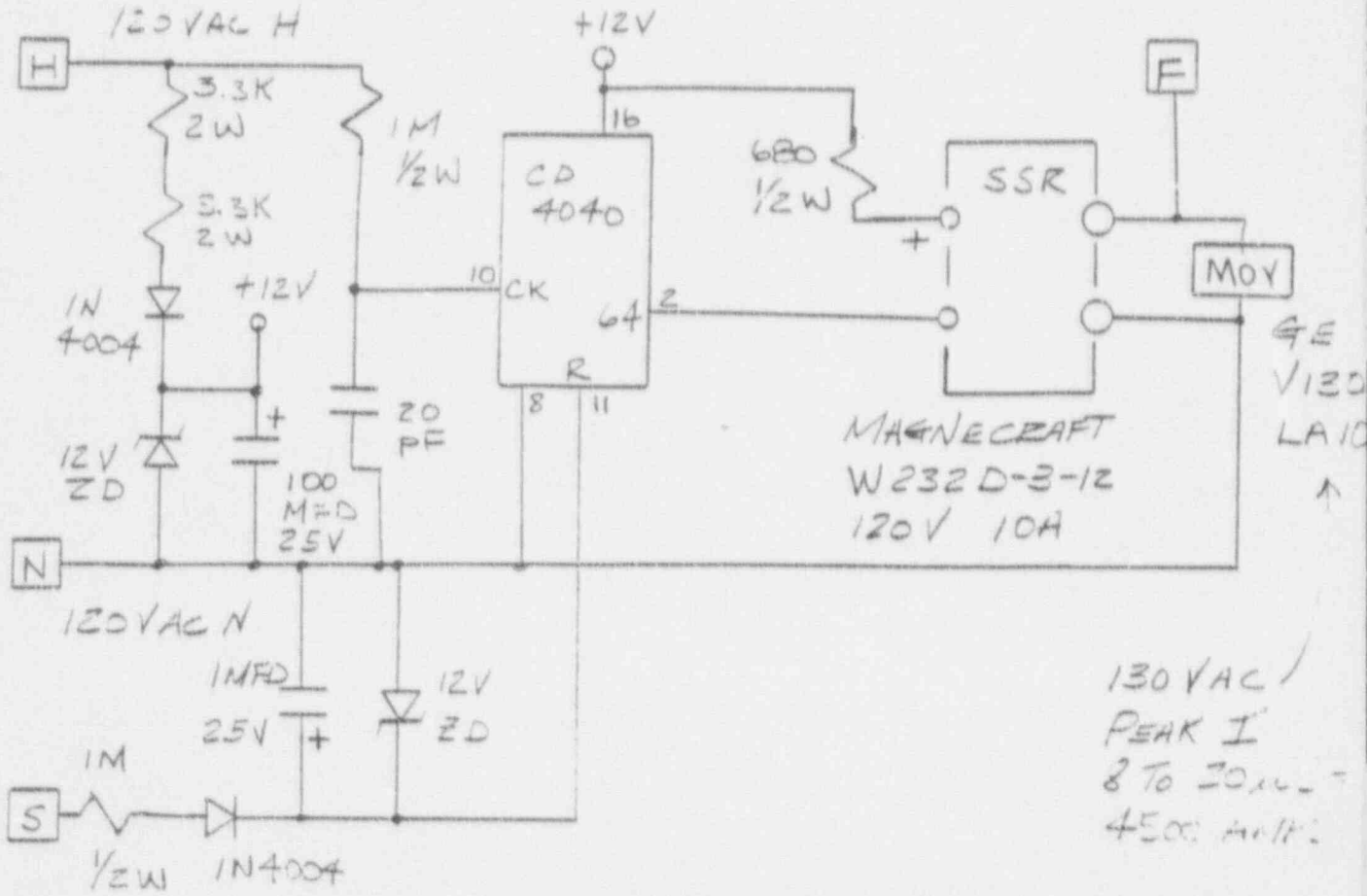
ICE	NORMAL	ALARM	SILENCE	RET. TO NORMAL (NOTE 7) SWITCH OF SEQ. AM	SWITCH OFF SEQ. AF	NORMAL AGAIN
S BLE SIGNAL CE CONTACT "A" "B"	OFF OFF NORMAL ENERGIZED ENERGIZED	FLASHING ON ABNORMAL DE-ENER. ENERGIZED	BRIGHT OFF ABNORMAL DE-ENER. DE-ENER.	BRIGHT OFF NORMAL ENERGIZED DE-ENER.	OFF OFF NORMAL ENERGIZED ENERGIZED	OFF OFF NORMAL ENERGIZED ENERGIZED

THE INSTRUMENT CORPORATION
CHICAGO, ILLINOIS

DATE	4-29-58
DRAWN	DF
APP'D.	TM
REV.	1-23-60

DRG. NO. ACS-82AM-F | SHT. 5

ELECTRONIC FLASHER FOR ANNUNCIATOR PANEL
REPLACEMENT FOR SCAM MODEL ACSF-1



CD 4040 - 12 BIT BINARY COUNTER

Georgia Institute of Technology

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Facility Modification Request 91-002

Power Level Measuring Channels
Picoammeter #1 and #2

The GTRR Power Level Measuring Channels (Picoammeter #1 and #2) are old GE vacuum tube instruments which are not manufactured any more. These instruments are 25 years old and frequently break down. Spare parts are nearly impossible to obtain. Consequently, it is desired to replace the instruments with Keithly picoammeters model #485/4853.

Recently, we received funding from the Department of Energy to purchase and install the two picoammeters. The attached document gives more details.

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 91-002

TITLE: PICOAMMETER REPLACEMENT

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
045-62-002 SHT 2	INSTRUMENTATION & CONTROL SCHEMATICS		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
2002	REACTOR OPERATIONS - PRECRITICAL STARTUP CHECKLIST AND SHIFT SUPERVISOR APPROVAL.		
72-74	PICOAMMETER CALIBRATION.		

Reviewed By: R. A. Kerr Date: 12/19/91

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 91-002

TITLE: PICOAMMETER REPLACEMENT

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:	B. STATHAM	12-17-91
APPROVALS:		
Director NNRC:	R. A. Karam	12/19/91
Nuclear Safeguards Committee:	[signature]	12/19/91

Picoammeter Replacement

1.0 PURPOSE

The purpose of this facility modification is to replace the Reactor Instrumentation Picoammeters with new modern technology picoammeters.

2.0 SCOPE

The proposal is to replace the existing General Electric Micromicroammeters with Keithley model 485 Autoranging Picoammeters.

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 Keithley Model 485 Autoranging Picoammeter Instruction Manual

4.2 General Electric Stable Micromicroammeter Catalog # 534E745G3,G4

4.3 Related Procedures

4.3.1 Procedure 7274 Picoammeter Calibration

4.3.2 Procedure 2002 Reactor Operations - Precritical Startup Checklist and Shift Supervisor Approval

5.0 SYSTEM DESCRIPTION

5.1 Existing Picoammeters

The existing picoammeters are General Electric Micromicroammeters equipped with a manually operated range switch. Each range decade has a x1 and a x2.5 position. There is a contact that is closed when the range switch is on the 2.5E-10 amp position. The 2.5E-10 amp position is used for reactor startup and the closed contact is a startup permissive. The Micromicroammeter has a 0 to 10 millivolt output used to drive the Power Level Recorder. There are no reactor scram signals from the Micromicroammeter.

5.2 Proposed New Picoammeters

The proposed new picoammeters are Keithley model 485 autoranging picoammeters. A 485 range change is 1 decade; in the auto mode the most sensitive on scale range is automatically selected.

The 485 autoranging feature eliminates the need for the startup permissive and this permissive will be deleted.

The 485 analog output is 1 millivolt per count; therefore a Power Level Recorder resistive divider network is necessary. The schematic of this purposed network is included. The network contains an x1, x3 switch; this feature allows the reactor operator to increase the Power Level Recorder reading by a factor of 3. The x3 position could be used when the Power Level Recorder is < 20% full scale on the x1 position and autocontroller operation is to be used.

Check Calibration of both Picoammeters

APPENDIX A

Dikata

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: ~~90-005~~ 90-002 RAK See memo 3/13/91

TITLE: REPAIR and UPGRADE COOLING TOWER

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 ✓

   The new materials used are the same or better capability than the present materials.

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

   Change does not alter the margin of safety defined in the technical specification.

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).

   This does not present an unreviewed safety question.

DATE:

PREPARED BY: William H. Downs September 10, 1990

APPROVALS:

for Director NNRC: B K Peerson 9/26/90

Nuclear Safeguards Committee: _____ _____

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 90-005
TITLE: REPAIR and UPGRADE COOLING TOWER

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
None			

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
None			

Reviewed By: BK Revsin Date: 9/26/90

Facility Modification
90-005

Cooling Tower
Repair and Upgrade

Approvals: BK Rivera for Director NNRC — 9/26/90 Date

Nuclear Safeguards Committee — Date

Cooling Tower
Repair and Upgrade

1.0 PURPOSE

The purpose of this facility change is to rejuvinate the cooling tower and bring it up to original operating specifications.

2.0 SCOPE

This proposal is to replace the existing wooden lath fill, plywood top deck and cold water basins with newer 'state of the art' components.

3.0 RESPONSIBILITY

The approval for this modification lies with the Neely Nuclear Research Center management with the concurrence of the Nuclear Safeguards Committee.

4.0 REFERENCES

4.1 GTRR 5 Mw Safety Analysis Report section 4.4.8.2, page 78

4.2 GTRR Technical Specifications Table 4.2
section 4.4.d

4.3 Related Procedures

4.3.1 Procedure 5063 CW Basin Low Level - Alarm

4.3.2 Procedure 9019 Cooling Tower Tritium Analysis

5.0 SYSTEM DESCRIPTION

5.1 EXISTING COOLING TOWER

The existing cooling tower has a wooden lath fill. It has a plywood top deck and cold water basins The louvres are asbestos.

5.2 PROPOSED UPGRADE

9/26/90 to be BKR

The proposed upgrade will install a new 3/4" top deck and cold water basins. The cold water basins will have removable to eliminate algae growth. New target nozzles will be installed also.

The wooden lath fill and drift eliminators will be removed and replaced with Munters PVC. The support lumber will be redwood.

Remove & replace the asbestos louvres with 12 oz. fiberglass.

Plywood will be Douglas Fir. Support lumber will be redwood. All lumber will be pressure treated. Hardware will be stainless steel.

5.3 POSSIBLE FAILURE MODES

None.

6.0 RECORDS

The changes in the system will be incorporated in the facility drawings listed in section 4.0.

TED MARSDEN, INC.

Specializing in Cooling Tower Services & Repairs

956 Strap Hinge Trail
Stone Mountain, GA 30083
(404) 292-8109

July 25, 1988

Georgia Institute Of Technology
Neely Nuclear Research Center
900 Atlantic Drive
Atlanta, Ga. 30332

Attention: Mr. Dean Mc Dowell

Subject: Marley Cooling Tower, Model #68-102

Dear Mr. McDowell:

This week, we conducted an inspection of this tower. Herein are our recommendations for getting the tower back into condition to operate as originally designed (1200 GPM -116-87-79 degrees). The structure is sound except for the top deck area.

We propose:

1. Install new 3/4" plywood top deck and cold water basins; the cold water basins will have removable covers to eliminate algae growth. Install new target nozzles.
2. Remove all wooden lath fill and drift eliminators. Install Munters PVC fill and drift eliminators with redwood supports. Munters fill is 'state of the art' in cellular film cooling. Their drift eliminator has the lowest pressure drop and is the most efficient in today's market.
3. Replace all louvers with 12 oz. fiberglass.

Contract price - $ 27,800.00 + 4% @ 29,712

Our contract price includes supervision, labor, materials, freight, and disposal of all debris. Plywood will be Douglas Fir; support lumber will be redwood; all lumber will be pressure treated. Hardware will be stainless steel.

The asbestos louver blades will be disposed of at a certified landfill; documentation of this will be forwarded to the customer.

We look forward to working with you on this project.

Sincerely,

Philip D. Marsden, President

TED MARSDEN, INC.

Enclosure

PM/jm

Reactor Cooling Tower Test

Date 2-27-91

Time	11:58	1316	1444	1532	
Reactor Power	994	2002	2963	3853	
Relative Humidity	42 40.1	35 34.6	36 30.9	35 31.2	
Outside Temperature	51 53	54 56.5	55 59	56 58.5	
Primary Flow	1760	1765	1790	1800	
Primary Inlet Temp.	77.5	83.0	95.9	106.5	
Primary Outlet Temp.	81.0	91.0	106.1	126.0	
Secondary Flow	940	930	920	930	
Secondary Hot Temp.	79.0	69.0	100.1	112.4	
Secondary Cold Temp.	69.4	86.0	75.5	80.4	
Shield Flow	34	34	34	34	
Shield Hot Temp.	72	72.8	78.0	83.9	
Shield Cold Temp.	72	71.8	77.0	83.0	
Fan Condition	#1	#1&2	#1&2	#1&2	
Bypass Valve	75%	75%	75%	75%	

Operator D. Cox

Reviewed B Statham

Reactor Cooling Tower Test

Date 12 Dec 90

Time	0949	1109	1223	1346	
Reactor Power	1004	2952	3053	3903	
Relative Humidity	60%	40%	40%	35%	
Outside Temperature	50	59	59	61	
Primary Flow	1770	1780	1795, 105.3	1810	
Primary Inlet Temp.	79.7	85	97.5 BS 97.6	108	
Primary Outlet Temp.	83.52	92.1	105 BS	121.3	
Secondary Flow	970	965	955	970	
Secondary Hot Temp.	80.9	88	102	114	
Secondary Cold Temp.	72 -	71.1	78.6	82.6	
Shield Flow	36	36	36	36	
Shield Hot Temp.	72.1	74.5	81.0	85	
Shield Cold Temp.	72.3	73.5	79.9	84.5	
Fan Condition	1 off 2 on	both on	Both ON	both on	
Bypass Valve	70%	67.5	80%	97%	

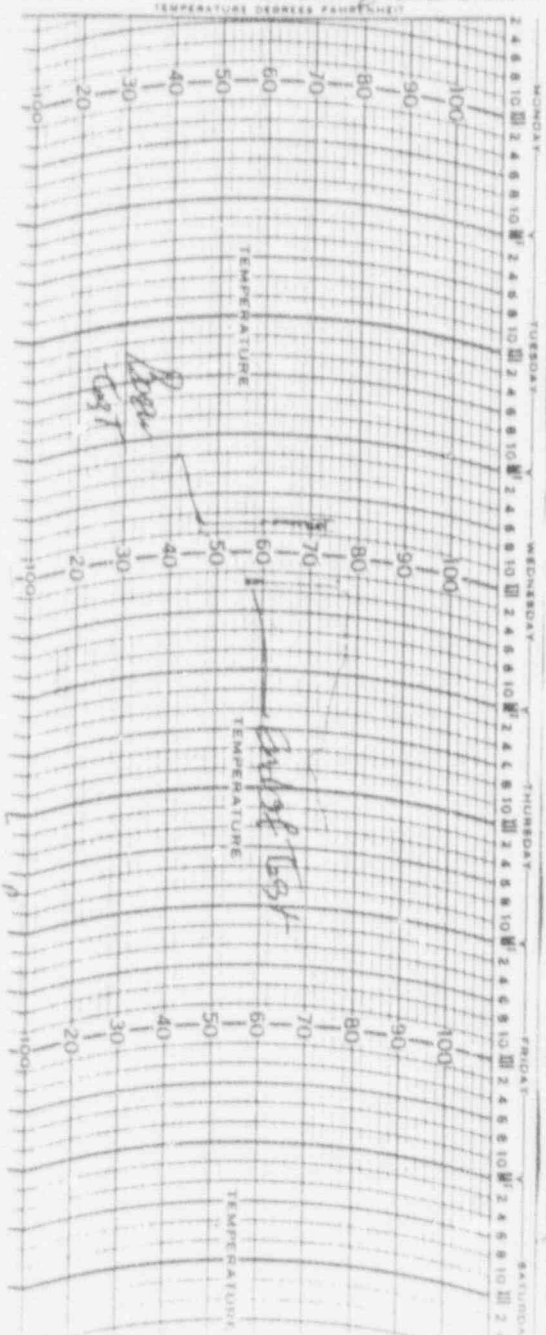
Operator E. Downs

Reviewed D. Statham

HYGRO-THERMOGRAPH
CHART NO. 5-207-W

ELFORT INSTRUMENT COMPANY
BALTIMORE MARYLAND, U.S.A.

DATE 2-27-91 STATION Cooling Tower



HYGRO-THERMOGRAPH
CHART NO. 5-207-W

BELFORT INSTRUMENT COMPANY
BALTIMORE MARYLAND, U.S.A.

INSTRUMENT NO. DATE 12-12-90 STATION Cooling Tower