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

February 25, 1994

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, 1993 through December 31, 1993 (calendar year 1993). 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 seven facility design changes during calendar year 1993. All 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 3.0 MW using a 17-element core. An 8-element fuel exchange to enhance self protection was performed. Fuel performance 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 1993 were as follows:

Proc. #	Title
0005	Criticality Alarm Testing
2002	Reactor Operations-Precritical Startup Checklist and Shift Supervisor Approval
2300	Bismuth Coolant System-Operation
3105	Bio Medical Facility Operation
4200	Changes in Facility Design
4400	D2 Analysis in Reactor Cover Gas
6010	General Rules and Guides for Handling Emergencies
6020	Response to Heavy Water Leakage in Containment Building
6040	Response to Fire at NNRC
6100	Emergency Notification
7200	Primary Coolant Sampling for Radionuclide Analysis
7220	Containment Building Isolation Test
7260	Automatic Fire Alarm Testing
7500	Bismuth System Operation Check
7910	Calibration Test for Keithley Model 485 Picoammeter
9015	Cooling Water Gamma Monitor
9016	Calibration and Testing of Filter Bank Monitor
9053	Basic Portable Neutron Meter Calibration

Proc. #	Title
9250	Facilities Contamination Surveys
9304	Routine Facility Radiation Surveys
9501	Control & Accountability of Radioactive Sources
9502	Control and Accountability of Radiation Generating Devices
9510	Radioactive Material Shipment

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
Dr. Rodney Ice, Manager of the Office of Radiation Safety
Mr. B. D. Statham, Reactor Supervisor and Electronic Engineer (approximately half time)
Mr. William Downs, Senior Reactor Operator
Mr. Dixon Parker, Senior Reactor Operator
Mr. Jerry Taylor, Senior Safety Engineering Assistant
Mr. Edgar Jawdeh, Health Physics
Mrs. Clara Galleshaw
Mrs. Arlene Robinson Smith
Mr. Nazee Chebeir, Health Physics

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

Ms. Kathleen Klee
Ms. Hannah Mitchell
Mr. Thomas Evans
Mr. Joseph Martin

Mr. Nick Jenkins (Trainee for Reactor Operator)
Mr. Neil Copeland (Trainee for Reactor Operator)
Mr. Jeremy Sweezy (Trainee for Reactor Operator)

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

(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

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

2. POWER GENERATION

For the period January 1, 1993 through December 31, 1993, the total power generation of the GTRR was 75.82 MW hours. The reactor was operated a total of 132 hours: 34.3 hours at power levels equal to or less than 100 kW, 90.2 hours at power levels 100 kW to 1 MW, and 7.5 hours at power levels above 1 MW.

3. SHUTDOWNS

During this reporting period there were 5 unscheduled shutdowns. Table 1 gives details.

TABLE 1 UNSCHEDULED REACTOR SHUTDOWNS DURING 1993

Report	Date	Trip Initiation	Reason for Trip	Corrective Action
93-01	3/8	High Bismuth Coolant Temp	TR-2 Initiated trip although the temperature was normal.	Loose terminal on the TR-2; trip circuit was tightened.
93-02	8/11	Low H20 Flow	H2O flow measuring instrument malfunctioned due to moisture in instrument air system	Drained moisture from instrument air system and repaired air dryer system.
93-03	9/28	Period trip	Operator observing a nonfunctioning Period recorder allowed period to exceed trip limit.	Repaired Period Recorder; modified procedure requiring operators to observe Period meters in addition to the Period recorder.
93-04	10/11	Low Bismuth Coolant Flow	Bismuth Pump switch was accidentally bumped by a student.	Informed students to be aware of how they move about control room.
93-05	10/11	Low Bismuth Coolant Flow	Air in lines caused temporary drop in flow	Allow 3 minutes after starting Bismuth pump to get air out of lines before resetting trip circuits.

4. UNSCHEDULED MAINTENANCE ON SAFETY RELATED SYSTEMS AND COMPONENTS

There were approximately twenty-one (21) 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 1993, there were 35 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)
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)*
1st Qtr	3.752	$2.98*10^{-8}$	$9.83*10^{-6}$	76
2nd Qtr	1.792	$1.42*10^{-8}$	$6.98*10^{-6}$	43.7
3rd Qtr	7.567	$6.01*10^{-8}$	$7.03*10^{-6}$	180
4th Qtr	6.383	$5.07*10^{-8}$	$5.86*10^{-6}$	275.5

*Technical Specifications release limit is 585 μCi/sec.

**Basis = Stack effluent at 34,000 cfm

(2) IODINES RELEASED

None Measurable
Lower Limit of Detection $<1.14*10^{-14}$ μCi/cc

(3) PARTICULATES

None Measurable
Lower Limit of Detection $<6.49*10^{-5}$ μCi
gross beta/gamma
Lower Limit of Detection $<7.07*10^{-6}$ μCi

b. Liquid Effluents

(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-SNM. No fission products are released via the liquid effluent pathway.

(1) CO^{60} RELEASE

	Total Release Ci	Avg. Release* Rate (μCi/cc)	% Tech Specs
1st QTR	$2.45*10^{-5}$	$1.24*10^{-10}$	< 1%
2nd QTR	$2.26*10^{-5}$	$1.30*10^{-10}$	< 1%
3rd QTR	$2.11*10^{-5}$	$1.05*10^{-10}$	< 1%
4th QTR	$1.60*10^{-5}$	$7.99*10^{-11}$	< 1%

*Average release rate values are based on a Georgia Tech campus water discharge rate of $2*10^{11}$ ml/quarter.

(2) TOTAL GROSS RADIOACTIVITY (β/gamma)

	Total Release Ci	Avg. Release* Rate (μCi/cc)	% Tech Specs
1st QTR	$1.27*10^{-6}$	$6.34*10^{-12}$	< 1%
2nd QTR	$4.75*10^{-7}$	$2.38*10^{-11}$	< 1%
3rd QTR	$1.71*10^{-6}$	$8.57*10^{-12}$	< 1%
4th QTR	$1.90*10^{-7}$	$9.50*10^{-12}$	< 1%

*Average release rate values are based on a Georgia Tech campus water discharge rate of $2*10^{11}$ ml/quarter.

(3) TRITIUM

	Total Release Ci	Avg. Release* Rate (μCi/cc)	% Tech Specs
1st QTR	$2.54*10^{-3}$	$1.27*10^{-8}$	< 1%
2nd QTR	$1.76*10^{-3}$	$8.78*10^{-9}$	< 1%
3rd QTR	$9.47*10^{-3}$	$4.74*10^{-8}$	< 1%
4th QTR	$1.64*10^{-3}$	$8.22*10^{-9}$	< 1%

*Average release rate values are based on a Georgia Tech campus water discharge rate of $2*10^{11}$ ml/quarter.

(4) GROSS ALPHA RADIOACTIVITY RELEASED

None Measurable

Lower Limit of Detection -
$<7.07*10^{-6}$ μCi

(5) VOLUME OF WATER RELEASED (ml/Quarter)

From Reactor Building

1st QTR . . . $5.45 * 10^{7}$ ml
2nd QTR . . . $5.68 * 10^{7}$ ml
3rd QTR . . . $8.02 * 10^{7}$ ml
4th QTR . . . $3.97 * 10^{7}$ ml

(6) VOLUME OF DILUTION WATER USED DURING EACH QUARTER

From Georgia Tech Campus

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

7. **ENVIRONMENTAL MONITORING:**(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 down-wind from the reactor.

b. Total assays = 30 sites X 12 months X 2 assays/site = 720 assays. These data 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. 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 the 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 a worst case exposure.

Several badges indicated radiation exposure above background levels. Badges No. 2, 14 and 15 all gave higher readings in May, June and August. These badges are located on the roof of the Neely Nuclear research Center (NNRC) facility. See the attached figure. The badges were attached to metallic surfaces with duct tape. Generalized increased badge readings were also observed during the months of July and August, two unusually warm months of the year for Atlanta in 1993.

An analysis of the indicated exposures indicates:

(1) The badge (No.1) which would most probably indicate significant gaseous/particulate radioactive release showed only "M" minimal exposure all year.

(2) The generalized indicated exposures do not correlate with nuclear operations. See workload at bottom of Table.

(3) Communication with Dr. W. G. Vernetson (TRTR Mtg., Oct. 1993) at the University of Florida nuclear reactor indicate that they also were experiencing anomalous environmental exposures.

Based upon our analysis, we conclude that the exposures were not correlated with reactor operations and were due to either reflective heat effects or systematic error associated with the badges or badge processing. To prevent reoccurence we 1) installed new dosimetry holders that would allow air circulation around the dosimeters, i.e. prevent reflective heating from attched surfaces, and 2) switched from film badge dosimeters to environmentally specific thermoluminescent dosimeters (TLD's). The switch to TLD's will also provide a more sensitive (unit millirems versus tens of millirems) assessment plus minimize any "heating effect" that may have caused our previous anomolous readings.

d. The highest, lowest and 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.

Average annual level - 27* mrem
Highest annual level - 130* mrem
Lowest annual level - < 10 mrem.

* There are reasons to question the validity of these numbers (see 7c above).

e. Based upon the EPA Comply Code, the maximum cumulative radiation dose above natural background radiation which could be received by an individual continuously present in an unrestricted area during reactor's operation would be less than the lower limits of detection (LLD), i.e. < 10 mrem.

8. <u>Occupational Personnel Radiation Exposure:</u>

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 $\leq$ 10 mrem. A monthly radiation dosimetry report is issued for the personnel of the Neely Nuclear Research Reactor. All personnel dosimetry data is kept at NNRC. Summary of personnel dosimetry follows.

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 = 0.16 rem

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

   Average annual level - 8.4 mrem
   Highest annual level - 40 mrem
   Lowest annual level - < 10 mrem.

d. Category of exposure

NNRC Radiation Workers

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

NEELY NUCLEAR RESEARCH CENTER
ENVIRONMENTAL RADIATION SURVEILLANCE*
1993

	JAN		FEB		MAR		APR		MAY		JUN	
BADGE #	D	S	D	S	D	S	D	S	D	S	D	S
09801	M	M	M	M	M	M	M	M	M	M	M	M
09802	M	M	M	M	M	M	M	M	20	20	40	40
09803	M	M	M	M	M	M	M	M	M	M	M	M
09804	M	M	M	M	M	M	M	M	M	M	M	M
09805	M	M	M	M	M	M	M	M	M	M	M	M
09806	M	M	M	M	M	M	M	M	M	M	M	M
09807	M	M	M	M	M	M	M	M	M	M	M	M
09808	M	M	M	M	M	M	M	M	M	M	M	M
09809	M	M	M	M	M	M	M	M	M	M	M	M
09810	M	M	M	M	M	M	M	M	M	M	M	M
09811	M	M	M	M	M	M	M	M	M	M	M	M
09812	M	M	M	M	M	M	M	M	M	M	M	M
09813	M	M	M	M	M	M	M	M	M	M	M	M
09814	M	M	M	M	M	M	M	M	10	10	50	50
09815	M	M	M	M	M	M	M	M	10	10	50	50
09816	M	M	M	M	M	M	M	M	M	M	M	M
09817	M	M	M	M	M	M	M	M	M	M	M	M
09818	M	M	M	M	M	M	M	M	M	M	M	M
09819	M	M	MISSING***		M	M	M	M	M	M	10	10
09820	M	M	MISSING***		M	M	M	M	M	M	M	M
09821	M	M	M	M	M	M	M	M	M	M	10	10
09822	M	M	M	M	M	M	M	M	M	M	10	10
09823	M	M	M	M	M	M	M	M	M	M	M	M
09824	M	M	M	M	M	M	M	M	M	M	M	M
09825	M	M	M	M	M	M	M	M	M	M	M	M
09826	M	M	MISSING***		M	M	M	M	M	M	M	M
09827	M	M	M	M	M	M	M	M	M	M	M	M
09828	M	M	M	M	M	M	MISSING***		M	M	M	M
09829	M	M	M	M	M	M	M	M	M	M	M	M
09830	M	M	M	M	M	M	M	M	M	M	M	M
Workload MW Hrs	10.90		7.44		2.06		.014		7.42		5.21	

* Sum of natural radiation, direct radiation from facility, and gaseous radioactive effluents. units in millirems (mr). No background or control substraction has been considered. Detection by film badge dosimeters, and processed by Landauer. Lower limit of detection is 10mR.

** Damaged film badge

*** Represent lost film badges for unknown reasons.

NEELY NUCLEAR RESEARCH CENTER
ENVIRONMENTAL RADIATION SURVEILLANCE*
1993

	JUL		AUG		SEP		OCT		NOV		DEC		YEAR	
BADGE#	D	S	D	S	D	S	D	S	D	S	D	S	D	S
09801	M	M	MISSING***		M	M	M	M	M	M	M	M	M	M
09802	M	M	30	30	20	20	M	M	M	M	M	M	110	110
09803	M	M	10	10	M	M	M	M	M	M	M	M	10	10
09804	10	10	10	10	M	M	M	M	M	M	M	M	20	20
09805	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09806	10	10	10	10	M	M	M	M	M	M	M	M	20	20
09807	M	M	10	10	20	20	M	M	M	M	M	M	30	30
09808	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09809	10	10	20	20	10	10	M	M	M	M	**		40	40
09810	M	M	M	M	M	M	M	M	10	10	M	M	10	10
09811	10	10	10	10	M	M	M	M	M	M	**		20	20
09812	M	M	10	10	M	M	M	M	M	M	M	M	10	10
09813	10	10	20	20	M	M	M	M	M	M	M	M	30	30
09814	**		40	40	20	20	M	M	M	M	M	M	120	120
09815	M	M	40	40	30	30	M	M	M	M	M	M	130	130
09816	10	10	10	10	M	M	M	M	M	M	M	M	20	20
09817	20	20	10	10	M	M	M	M	M	M	M	M	30	30
09818	M	M	10	10	M	M	M	M	M	M	M	M	10	10
09819	M	M	MISSING***		M	M	M	M	M	M	**		10	10
09820	10	10	20	20	M	M	M	M	M	M	M	M	30	30
09821	10	10	MISSING***		M	M	M	M	M	M	M	M	20	20
09822	10	10	10	10	M	M	M	M	M	M	M	M	30	30
09823	10	10	10	10	M	M	M	M	M	M	M	M	20	20
09824	10	10	10	10	M	M	M	M	M	M	M	M	20	20
09825	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09826	M	M	10	10	M	M	M	M	M	M	M	M	10	10
09827	20	20	M	M	M	M	M	M	M	M	M	M	20	20
09828	M	M	10	10	M	M	M	M	M	M	M	M	10	10
09829	M	M	M	M	M	M	M	M	MISSING***		M	M	M	M
09830	20	20	10	10	M	M	M	M	M	M	M	M	30	30
Wrkload MW Hrs	.02		9.91		1.82		22.65		5.10		3.26			

* Sum of natural radiation, direct radiation from facility, and gaseous radioactive effluents. units in millirems (mr). No background or control substraction has been considered. Detection by film badge dosimeters, and processed by Landauer. Lower limits of detection is 10mR.

** Damaged film badge

*** Represent lost film badges for unknown causes.

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/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.

APPENDIX A

Facility Modifications

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 93-001

TITLE: CWGM Flow Indicator

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:	Dixon F. Parker	2-17-93
APPROVALS:		
Director NNRC:	R. A. Karam	2/22/93
Nuclear Safeguards Committee:	NSC Approved	2/25/93

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 93-001

TITLE: CWGM Flow Indicator

DRAWINGS: No changes in drawings needed

NUMBER	TITLE	REVISED BY	DATE

PROCEDURES: No changes to procedures needed

NUMBER	TITLE	REVISED BY	DATE

Reviewed By: [signature] Date: FEB 25 1994

System Installed + Functioning properly

Facility Modification 93-001
CWGM Flow Indicator

1.0 PURPOSE

The purpose of this facility modification is to replace the existing non-operational flow indicator in the secondary coolant system line to the Cooling Water Gamma Monitor (CWGM) with a new one.

2.0 SCOPE

This modification applies only to the one time replacement of the flow indicator in the CWGM water line.

3.0 RESPONSIBILITY

The responsibility for the approval of this facility modification lies with the NNRC director with the concurrence of the Nuclear Safeguards Committee. The installation of the flow indicator will be by the NNRC Staff.

4.0 REFERENCES

Procedure 2002 - Precritical Startup Checklist and Shift Supervisor Approval

5.0 SYSTEM DESCRIPTION

5.1 Existing Flow Indicator

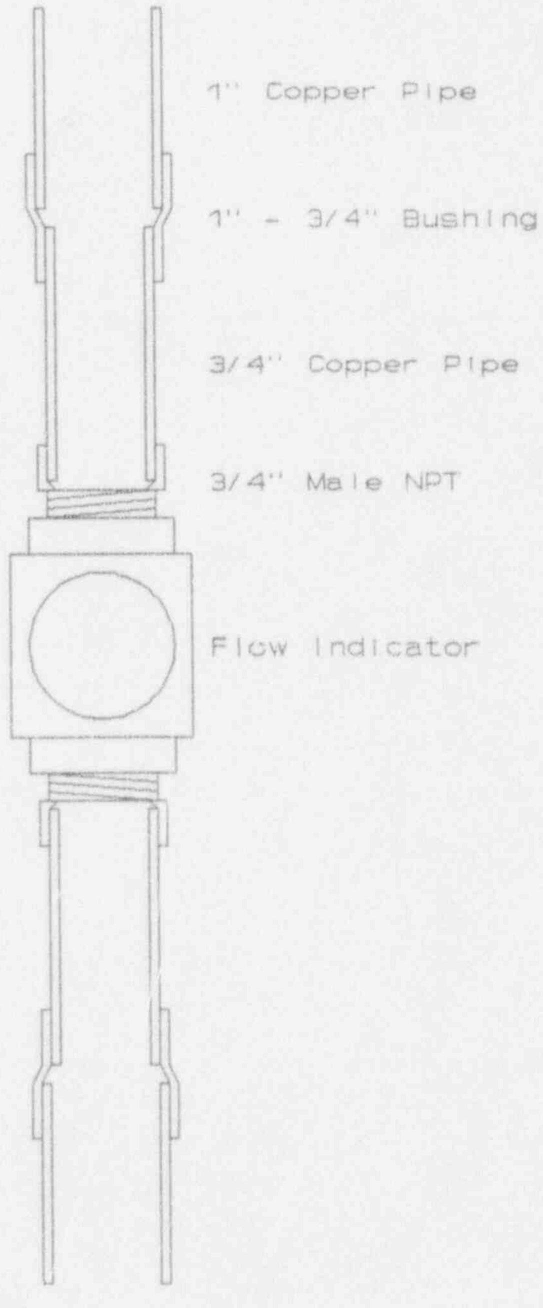
A brass rotary sight flow indicator with 3/4" female NPT fittings manufactured by Ernst W. C. & G. Company is currently installed in the 1" copper line. The copper line is reduced to 3/4" for the flow indicator (see diagram). The flow indicator functions by giving visual indication of flow by verifying that the vanes inside the indicator are rotating. This indication is sufficient, but the existing indicator no longer functions.

5.2 Proposed Flow Indicator

The proposed flow indicator is very similar to the existing one. A brass rotary sight flow indicator with 3/4" female NPT fittings has been selected from McMaster Carr Company. The indicator has a maximum operating temperature of 200°F and maximum operating pressure of 125 psig. The operating temperature and pressure are well below these values.

6.0 IMPLEMENTATION

1. Drain water lines around the CWGM.
2. Cut the 3/4" pipe on one side of the flow indicator.
3. Remove the current flow indicator.
4. Unsolder the cut 3/4" pipe from the 1"-3/4" bushing.
5. Make a new 3/4" pipe section with 3/4" male NPT fitting.
6. Screw the 3/4" pipe section onto the new flow indicator.
7. Screw the new flow indicator onto the uncut 3/4" pipe.
8. Solder the 3/4" pipe into the existing 1"-3/4" bushing.



APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 93-002

TITLE: Replacement of LA-D2 Pressure Switch

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:	D.F. Parker	2-18-93
APPROVALS:		
Director NNRC:	R.A. Karam	2/22/93
Nuclear Safeguards Committee:	NSC Approved	2/25/93

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 93-002

TITLE: Replacement of LA-D2 Pressure Switch

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
045-50-008	Primary Coolant System	J. Burleigh	8/19/93

PROCEDURES: None needed

NUMBER	TITLE	REVISED BY	DATE

Reviewed By: [signature] Date: FEB 26 1994

Facility Modification 93-002
Replacement of LA-D2 Pressure Switch

1.0 PURPOSE

The purpose of this facility modification is to replace the old pressure switch that provides the input for the scram from the reactor tank level (LA-D2) with a new one.

2.0 SCOPE

This modification applies only to the one time replacement of the LA-D2 pressure switch.

3.0 RESPONSIBILITY

The responsibility for the approval of this facility modification lies with the NNRC director with the concurrence of the Nuclear Safeguards Committee. The installation of the pressure switch will be by the NNRC Staff.

4.0 REFERENCES

Procedure 2006 - Reactor Shutdown Checklist

Drawing 045-50-008 - Primary Cooling Water System (D_2O)

5.0 SYSTEM DESCRIPTION

5.1 Existing Pressure Switch

The existing pressure switch was manufactured by Meletron Corp. The switch has a range of 0.75-20 psig and a proof pressure of 65 psig. The switch contacts are rated for 10 Amps at 125 Volts. The switch is located next to the pipe chase and approximately 13.5 feet below the dump level of the reactor tank. The pressure switch has been a continual source of problems and causes spurious reactor scrams for two reasons. The first is that the smallest division for the setpoint adjustment is one psig, not sensitive enough. The second is that the switch is not a differential pressure switch; it is affected by the pressure of the cover gas system.

5.2 Proposed Pressure Switch

The proposed pressure switch is manufactured by Omega Engineering. The switch is model PSW-364. It has a range of 23" to 150" of water and a proof pressure of 21.6 psig. The electrical contacts are rated for 10 amps at 125 volts. The setpoint repeatability is 1% of full scale or 1.5" of water, and will provide adequate sensitivity. The switch is a differential pressure switch, and will be connected to the cover gas system to negate the pressure supplied by the cover gas. The proof pressure is still well above the maximum pressure the switch will be subjected to. The new switch will be located in approximately the same place, but will be a maximum of 12 feet below the dump level of the reactor.

6.0 IMPLEMENTATION

1. Drain primary water lines around the pressure switch.
2. Mount the new switch on the wall.
3. Remove the D_2O line from the existing switch, and connect it to the high pressure side of the new switch.
4. Install a line from the cover gas to the low pressure side of the new switch.
5. Remove the alarm contacts from the old switch and connect them to the new switch.
6. Check the new switch for proper function.
7. Remove the old switch from the wall.

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 93-003

TITLE: Replacement of LIA-D1 Remote Level Indication System

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:	D.F. Parker	2-23-93
APPROVALS:		
Director NNRC:	R.A. Karam	2/25/93
Nuclear Safeguards Committee:	NSC Approved	2/25/93

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 93-003

TITLE: Replacement of LIA-D1 Remote Level Indication System

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
0045-62-001	Control Room	Not needed	FEB 25 '94

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
7241	Reactor Tank level Transmitter Maint	D.F. Parker	1/13/94

Reviewed By: [signature] Date: FEB 25 '94

Facility Modification 93-003
Replacement of LIA-D1 Remote Level Indication System

1.0 PURPOSE

The purpose of this facility modification is to replace the old remote level indication system (LIA-D1) with a new one that provides readout in the control room of the reactor tank level and input for a reactor tank level scram.

2.0 SCOPE

This modification applies only to the one time replacement of the LIA-D1 remote level indication system.

3.0 RESPONSIBILITY

The responsibility for the approval of this facility modification lies with the NNRC director with the concurrence of the Nuclear Safeguards Committee. The installation of the pressure switch will be by the NNRC Staff.

4.0 REFERENCES

Procedure 2002 - Precritical Startup Checklist and Shift Supervisor Approval

Procedure 7241 - Reactor Tank Level Transmitter Maintenance and Calibration Check.

Drawing 045-50-008 - Primary Cooling Water System (D_2O)

Drawing 045-62-001 - Control Room

5.0 SYSTEM DESCRIPTION

5.1 Existing Remote Level Indication System

The existing remote level indication system consists of three instruments and associated piping and wiring. The three instruments are:

1. A Mercury Differential Pressure Transmitter (LI-D1T) manufactured by Honeywell,

2. a pressure gauge denoted as a Pneumatic Indicating Receiver (LI-D1I) manufactured by J. P. Marsh, and

3. a Pressure Switch manufactured by Meletron Corp (LA-D1).

The specifications for all three instruments are attached.

The pressure transmitter (LI-D1T) is connected to the Reactor Tank Level Column (LC-D1) and senses the pressure differential of the reactor tank water level and the cover gas pressure. This differential pressure is converted to an air signal and is transmitted to the control room via air lines. In the control room this air signal is connected to the pneumatic indicator (LI-D1I) to provide a readout and to the pressure switch (LIA-D1) to provide a scram.

The current system is no longer able to maintain an acceptable accuracy, and requires excessive maintenance.

5.2 Proposed Remote Level Indicating System

The proposed level indicating system will consist of two instruments, both manufactured by Omega Engineering, and associated piping and wiring. The instruments are:

1. A Heavy Duty Industrial Differential Pressure Transmitter,
2. a Digital Panel Indicator with Excitation and Dual Alarm Relays.

The specifications for these instruments are attached.

The new pressure transmitter will be connected to the Reactor Tank Level Column in the same manner as the existing transmitter. The differential pressure is converted into an electrical signal that is transmitted to the control room. In the control room this electrical signal will be connected to the Digital Panel Meter to provide readout and a scram signal.

The proposed system has much better accuracy than the original and will be more reliable and provide digital indication of the reactor tank level in the control room. The new system is also compatible with a proposed change to allow the reactor tank level to be viewed in the Emergency Command Center during emergencies.

6.0 IMPLEMENTATION

1. Drain primary water lines around the old pressure transmitter.
2. Remove the existing transmitter.
3. Mount the new transmitter.
4. Install the water and gas lines from Reactor Tank Level Column to the new transmitter.
5. Remove the pressure gauge and pressure switch from the control room.
6. Install the panel meter in the control room.
7. Run wires from the new pressure transmitter to the panel meter.
8. Connect the panel meter to the scram input.
9. Calibrate the system per the manufacturer's instructions.
10. Ensure the system is functioning properly.
11. Remove air lines.

Mercuryless Pressure Transmitter

Brown Instrument Division
Minneapolis-Honey well Regulator Company
Wayne & Windrim Avenues
Philadelphia, Pennsylvania

Model #228N4C - As defined by the following specifications

Conditions for Operation:

Fluid:	Heavy Water
Temperature:	100°F
Pressure:	6 psig
Range:	0 to 100 inches of D_2O differential.

Construction:

Case:	Black baked enamel, vapor proof, with yoke for 2 inch pipe or mounting bracket.
Manometer:	Bellows type with pulsation damper.
Materials:	Body - stainless steel Bellows - stainless steel Bolting - alloy steel
Connections:	Air - 1/4" NPT Manometer - 1/2" NPT flanged process connection to connect to three valve manifold.
Accessories:	Transmitter shall be complete with integrally mounted combination air filter & reducing valve.

Pneumatic Indicating Receiver

J. P. Marsh Company
Skokie, Illinois

Model -Master Movement Dial Indicating Refinery Pressure (12") as defined by the following specification.

Construction:

Case:	Black baked enamel, smooth finish, flush mount.
Dial:	12"
Scale:	0-100 uniform scale.
Input Air:	3-15 psi from transmitter.
Accuracy:	1/2%

Other:

Remarks:	Standard accessories & legend plate required.
Use:	Panel mounted receiver for Brown Instrument Model 228N4C non-indication bellows type differential, transmitter with pulsation damper.

Pressure Switch

Meletron Corp.

Model:	420E
Catalogue #:	420E 6SS10A
Range:	0.5-10.0 psig
Proof Pressure:	25 psig

Heavy Duty Industrial Differential Pressure Transmitter

Omega Engineering Inc.
Stamford, CT

Model PX761-150WDI-B3-FL-FL as defined by the following specifications.

Conditions for Operation:

Fluid:	Water
Temperature:	-22 to 212°F
Proof Pressure:	2000 psig
Range:	Adjustable 0-25 to 0-150 inches of H_2O differential.

Construction:

Electrical Housing:	Epoxy painted low copper aluminum.
Gauge Type:	Capacitance sensor, using 316SS isolation diaphragm with silicone oil fill.
Materials:	Body - stainless steel Diaphragm - stainless steel Bolting - alloy steel
Connections:	Electrical - 1/2" NPT Gauge - 1/4" NPTF
Accuracy:	0.25%
Output:	4-20mA two wire
Excitation:	12-45Vdc

Digital Panel Meter

Omega Engineering, Inc.
Stamford, CT

Model DP205 panel meter as defined by the following specification.

Construction:

Case:	Aluminum case.
Display:	4 digit, red LED, 0.56" high
Range:	-1999 to 9999 counts.
Input:	4-20mA, 0-100mV, 0-10V
Accuracy:	0.02%

Other:

Operating Temp:	32 to 122°F.
Analog Output:	Scalable 0-10V or 4-20mA.
Excitation:	12V @ 100mA or 24V @ 50mA
Relays:	Dual 250 Vac, 6 amp, SPDT
Power:	115 Vac, 6 watts maximum

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 93-004

TITLE: EMERGENCY COMMAND CENTER MONITORING

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	2-18-9
APPROVALS:		
Director NNRC:	R.A. Abram	2/25/93
Nuclear Safeguards Committee:	NSC Approved	2/25/93

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 93-004

TITLE: EMERGENCY COMMAND CENTER MONITORING

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
045-62-002	INSTRUMENTATION & CONTROL	D.F. Parker/s/......	FEB 25 ...
SHEET 2	SCHEMATICS		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
2002	REACTOR OPERATIONS -	Parker	[illegible]
	PRECRITICAL STARTUP		
	CHECKLIST AND SHIFT		
	SUPERVISOR APPROVAL.		

Reviewed By: [signature] Date: FEB 25 [illegible]

FACILITY MODIFICATION 93-004

EMERGENCY COMMAND CENTER MONITORING

1.0 PURPOSE

The purpose of this facility modification is to increase the GTRR parameters that can be monitored in the ECC (Emergency Command Center).

2.0 SCOPE

At present, the reactor tank level can be monitored in the ECC via Closed Circuit TeleVision (CCTV). Both the camera and monitor require utility power to function. This proposal will add three (3) GTRR fuel element temperatures to the parameters that can be monitored in the ECC. In addition, battery backup system will be provided.

3.0 RESPONSIBILITY

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

4.0 REFERENCES

4.1 Omega DPS3200-TC Programmable Process Scanner Manual

4.2 Related Procedures

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

5.0 SYSTEM DESCRIPTION

5.1 Existing System

The existing system is comprised of a CCTV camera in the GTRR control room, a monitor in the ECC and a connecting coaxial cable. The camera and monitor are powered by utility power. The camera is energized during reactor operation at power levels greater than 1 MW (per procedure 2002). The camera views the pneumatic instrument that indicates the D_2O level in the GTRR vessel.

5.2 Proposed New System for the Control Room

5.2.1 Add an Omega DPS3200-TC Programmable Process Scanner to the system. The Scanner is capable of scanning up to seven (7) inputs with mix of signals from four (4) different type thermocouples, 4-20 mA and 0-10 V. The scanner has an alpha numeric LED (Light Emitting Diode) display.

5.2.2 Replace the camera with a low light level type camera. This camera will have the capability of functioning with only the light from the scanner display.

5.2.3 Add a battery backup system that is automatically energized if utility power is lost during reactor operation at a power level greater than 1 MW.

5.2.4 Connect three (3) of the fuel element thermocouples to the Scanner.

5.2.5 The instrumentation for the measuring the D_2O level in the GTRR vessel is being changed; reference facility modification 93-003. Connect this instrumentation into one (1) of the Scanner's 4-20 mA inputs.

5.3 Proposed New System for the ECC

5.3.1 Add a battery backup system for the existing monitor that can be manually activated.

6.0 OBJECTIVE

To enable the Emergency Director to make a more informed add water to the ECCS (Emergency Core Coolant System) decision.

Programmable Process Scanners
7-Channel Input

OMEGA Sensors Shown	
Code	Description
A	LCL-113G Load Cell, **$69**
B	PX602-060GV Pressure Transducer, **$175**
C	LD500 LVDT Sensor, **$150**
D	KMTSS-125G-6 Thermocouple, **$24**
E	PX181-100G5V Pressure Transducer, **$169**
F	LCC-50 Load Cell, **$350**
G	PX410-500SI Pressure Transducer, **$295**

DPS3000 Series From **$595**

- Independent Scaling
- Thermocouple, Voltage, Current Input
- Rate of Change and Min or Max Display
- 4 Digit Display Plus 3 Alphanumeric

The DPS3000 series scanners accept seven individual inputs. The DPS3200 can accept different input types on each channel, while the DPS3100 is limited to one input type on all seven channels.

Both the DPS3100 and DPS3200 meters can be setup to display in any of six operating modes. The unit can scan each channel; display elapsed time; display the channel with highest or lowest reading; scan each channels' deviation from a predetermined setpoint; scan the difference between a 'master' channel and the other inputs.

Specifications

Accuracy: 0.5% Rdg (J, K, T, E: 1°C; S 2°C; R,B 3°C)
CJC Error: 1°C (10°C to 40°C)
Resolution: 1°C/1°F (0.1°C/F for thermistor input); .025% FS
Power: 120 Vac, 60 Hz; optional 240 Vac, 50 Hz; optional 7 to 12 Vdc, 900 mA
Scan Rate: fixed, 2 channels/s
Channel Display Time: 1 to 999 s per channel
Alarms (optional): mechanical relay: 1 A @ 28 Vdc or .5 A @ 120 Vac, SPDT; dc driver: 5 Vdc @ 50 mA. One alarm per channel, (channel 7 is monitoring only)
Deadband: 2 to full scale, adjustable
Cutout: 45 x 92 mm (1.772" x 3.662"); 1/8 DIN; 9.0" depth

To Order (Specify Model Number)

Model No.	Price	Input Type(s)
DPS3100-(*)	$595	see * below
DPS3200-TC	695	J, K, T, E, 4-20 mA, 0-10 V
DPS3200-R	695	R, 4-20 mA, 0-10 V
DPS3200-S	695	S, 4-20 mA, 0-10 V
DPS3200-B	695	B, 4-20 mA, 0-10 V
DPS32[illegible]	695	0-100 mV, 4-20 mA, 0-10 V
DPS320[illegible]	6[illegible]5	400 series thermistor, −22 to 212°F (−30 to 100°C)

** Specify input type: J, K, T, E, R, S, B*
Ordering Example: *DPS3100-J,* ***$595.*** *DPS3100 7-channel scanner, dedicated to type J thermocouple input.*

Options-DPS3100, DPS3200 and DP3400

Ordering Add'l Suffix	Price	Description
-1	$ 5	240 Vac power
-2	25	7 to 12 Vdc power
-3	110	6 alarms, 4 SPDT, 2 dc drivers
-4	110	6 alarms, all dc drivers
-5	125	Analog output, 1 mV/°C (DP3400 only)

Minor Change Number: By: Date: / /	NEELY NUCLEAR RESEARCH CENTER CHANGES IN GTRR DESIGN	Procedure 4200 Revision 00 Approved 04/28/89 Page 3 of 4

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 93-005

TITLE: REPLACEMENT OF TIMER IN SECURITY SYSTEM

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	7-6-93
APPROVALS:		
Director NNRC:	R. A. Karam	7/6/93
Nuclear Safeguards Committee:	[signature]	7/23/93

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 93-005

TITLE: REPLACEMENT OF TIMER IN SECURITY SYSTEM

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
	N/A		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
	N/A		

Reviewed By: [signature] Date: 7/27/93

FACILITY MODIFICATION 93-005
REPLACEMENT OF TIMER IN SECURITY SYSTEM

1.0 PURPOSE

The purpose of this facility modification is to replace the timer in the security system.

2.0 SCOPE

The proposal is to replace the timer.

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 Included sheet that shows the current model and the proposed replacement model.

5.0 SYSTEM DESCRIPTION

NOTE: The timer in the security system activates the card lock at the entrance to reactor control zone during business hours.

NOTE: The timer must be capable of continued operation during a power outage and contain a minimum of two Single Pole Single Throw (SPST) Normally Open (NO) contacts. The size of the timer must be such that it will fit in the security box.

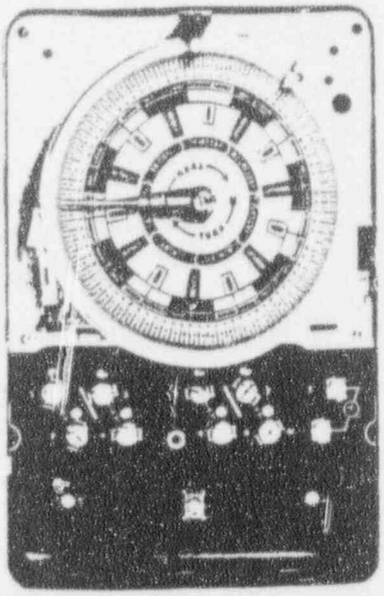
5.1 Existing timer is a Paragon model 7618-56. The unit looses time at the rate of approximately 1 hour per week. It is an obsolete model with no replacement parts available from Paragon. A recent attempt to correct the problem was unsuccessful.

5.2 Proposed replacement timer is an Omron model H5L-A. This unit meets the requirements of the security system having two SPST NO contacts and battery backup to keep the unit operational during a power outage. The model H5L-A timer will fit in security box.

7618-56 CURRENT MODEL

7600 SERIES

Three Hour Minimum ON Or OFF Time Four Pole, Quartz Carry-Over



7-day calendar permits different ON/OFF schedules on different days of the week. Program up to four ON/OFF operations per day, 28 per week. Three hour minimum and 21 hour maximum ON or OFF time per day. Independent four pole design allows for SPST, DPST or SPDT switching. Manual override lever temporarily reverses switch operation without permanently disturbing preset schedule. Switch slider bar assures positive switching. Heavy duty industrial type synchronous motor. Heavy duty terminals accommodate up to AWG #8 wire. Standard NEMA 3 indoor/outdoor metal enclosure side hinged with combination 1/2-3/4 inch knockouts in bottom and side. Hasp for padlock or seal. Quartz carry-over to take over during power outage. Quartz carry-over keeps control on time for a maximum of 7 days.

How To Specify

Installer shall furnish and install Paragon (Model No.) 7-Day General Purpose Time Control with (4PST-4 NO, 2 NO-2 NC) switch. Contacts to be rated at 40 amperes tungsten or 40 amperes noninductive per pole up to 240 volts. Control shall have NEMA 3 indoor/outdoor metal enclosure.

EDP No. 4988	Time Control Model No.	Switch Type	Wiring Diag. See pg. 49	Switch Rating Per Pole				AC Line		Enclosure Type	Shipping Wgt.	
			Figure:	Amp	Amp T	VA	Hp	Volts	Hz		Lb.	Kg.
71670	7617-56	4PST-4 NO	27	40	40	690	-	120	50/60	1	10	4.5
76172	7617-62	4PST-4 NO	27	40	40	690	--	208-240	50/60	1	10	4.5
⇒ 76180	7618-56	2 NO-2 NC	28	40	-	690	1	120	50/60	1	10	4.5
76182	7618-62	2 NO-2 NC	28	40	-	690	1	208-240	50/60	1	10	4.5

For enclosure dimensions, see page 39.
Furnished with " sets of ON OFF trippers for additional sets see page 41

SPDT, DPST or DPDT operation possible by using extra jumpers (furnished).
Hp rating (one pole only)

REPLACEMENT MODEL

ELECTRIC CONTROL AND DISTRIBUTION

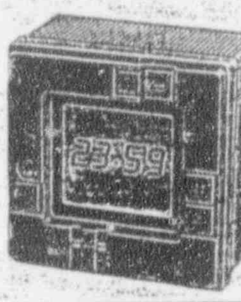
24 HOUR AND 7-DAY PROGRAMMABLE TIMERS

ELECTRONIC, 24 HOUR, PROGRAMMABLE, 2-CHANNEL TIMER

omron

E52800

LR22310



- 2 channels with independant programming for each circuit
- Weekly timer; 24 hours x 7 days using five programming keys
- Manual On-Off switching for each control output without changing program
- 4 adjustable cycle models
- 10-year memory protection by b battery
- Easy-to-read, 0.5" high LCD display
- Surface, track (DIN type mounting see page 367), and flush mountin 1/4 DIN panel cutout
- Dimensions: 3¾H x 3¾W x 2¼"D

ELECTRICAL SPECIFICATIONS						TIMER SPECIFICATIONS			ORDERING DATA			
No. of Poles	Contact Load Rating @ 60 Hz				Timer Input Volts @ 50/60 Hz	Cycle Length	Minimum Cycle Interval	Max. On-Off Cycles	Omron Model	Stock No.	List	Each
	Form	Max. Amps/Pole, Resistive 125VAC	250VAC	30VDC								
2	SPST	15	10	10	100-240VAC	1 Min to 23 Hrs 59 Min	1 Minute between On-Off periods	8	H5L-A	4A342	$146.00	$144.5

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 93-006

TITLE: Replacement of Sump and Pump for the Bismuth Cooling water Collection System

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:	D.F. Parker	9/22/93
APPROVALS:		
Director NNRC:	R.A. Karam	9/23/93
Nuclear Safeguards Committee:	[illegible]	9/23/93

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 93-006

TITLE: Sump/Pump Replacement Bismuth CWCS

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
045-53-004 4/2	Bismuth Cooling Water Collection System	Not needed DPP	FEB 25 1994

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
2300	Bismuth Cooling System Operation	Parker	FEB 25 1994

Reviewed By: [signature] Date: FEB 25 1994

Facility Modification 93-006

Replacement of Sump and Sump Pump for the Bismuth Collection System

1.0 PURPOSE

The purpose of this facility modification is to replace the sump and sump pump for the bismuth cooling water collection system.

2.0 SCOPE

The modification applies only to the one time replacement of the sump and the sump pump for the bismuth collection system.

3.0 RESPONSIBILITY

The responsibility for approval of this facility modification lies with the NNRC director with the concurrence of the Nuclear Safeguards Committee. The implementation will be done by the NNRC staff.

4.0 REFERENCES

Procedure 2300 - Bismuth Cooling System Operation

Drawing 045-53-0004 - sheet 2 of 2 -- Bismuth Cooling Water Collection System

5.0 SYSTEM DESCRIPTION

5.1 Existing Sump and Sump Pump

The current sump and sump pump is an integral unit with a maximum sump capacity of approximately one gallon and an original pumping capacity of one half gallon per minute. The pump has a built in level switch to control pumping. Recently the pump has become clogged several times from silt like material that is getting into the collection system. The pump has also been damaged by this silt like material and will no longer pump a sufficient amount of water.

5.2 Proposed Sump and Sump Pump

A new sump with capacity of 15 gallons will be used. The sump will be a polyethylene drum. A new sump pump with a 1.5 gallon per minute pumping capacity and the ability to pass solid materials is proposed. The pump has a built in level switch to control pumping. The pump is manufactured by Flotec. The current piping for the system can be used.

6.0 IMPLEMENTATION

1. Unplug the current pump.
2. Disconnect the plumbing fittings.
3. Remove old sump and pump.
4. Place new sump and pump.
5. Connect plumbing fittings.
6. Plug in pump.
7. Test for proper operation.

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 93-007

TITLE: Level Alarm on Bismuth Water Collection System

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:	D.F. Parker	9/22/93
APPROVALS:		
Director NNRC:	R.A. Karam	9/23/93
Nuclear Safeguards Committee:	[illegible signature]	9/23/93

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 93-007

TITLE: Level Alarm on the Bismuth Water Collect System

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
045-53-004 2/2	Bismuth Cooling Water Collection System		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
2300	Bismuth Cooling System Operation		

Work on this ~~modf~~ modification continuing as of FEB 25 199?

DFP

Reviewed By: ____________ Date: ____________

Facility Modification 93-007
Installation of a Level Alarm on the Bismuth Water Collection System

1.0 PURPOSE

The purpose of this facility modification is to Install a level alarm with an annunciator in the control room on the bismuth water collection system.

2.0 SCOPE

The modification applies only to the one time installation of the alarm on the bismuth water collection system

3.0 RESPONSIBILITY

The responsibility for approval of this facility modification lies with the NNRC director with the concurrence of the Nuclear Safeguards Committee. The implementation will be done by the NNRC staff.

4.0 REFERENCES

Procedure 2300 -- Bismuth Cooling System Operation

Drawing 045-53-0004 - sheet 2 of 2 -- Bismuth Cooling Water Collection System

5.0 SYSTEM DESCRIPTION

5.1 Existing Alarm

None.

5.2 Proposed Alarm

A float type level switch from McMaster Carr will be installed on the new sump for the bismuth cooling water collection system. The switch will be connected to an audible and visual alarm in an existing NIM bin in the control room. The audible alarm can be silenced, but the visual alarm will remain lit until the level in the sump drops below the alarm level. The alarm level will be approximately half way full on the 15 gallon sump. The level will give approximately 30 minutes of response time before water overflows the sump.

6.0 IMPLEMENTATION

1. Install level switch on sump.
2. Run wire to control room.
3. Connect wire to alarm Nim module.
4. Test system to ensure proper operation.

Environmental Monitoring Stations

SEPTEMBER, 1982

SCALE 3/4 in. = 100 ft

FIGURE ONE