Georgia Institute of Technology
NEELY NUCLEAR RESEARCH CENTER
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February 22, 1993

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, 1992 through December 31, 1992 (calendar year 1992). 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 1992: one involving the addition of an alarm which activates when the power level on picoammeter #1 and picoammeter #2 deviates by more than 10% from steady state; another comprised replacing the reactor primary and secondary coolant temperature measuring systems with a new one; and the third, consisted of replacing the temperature probe wells of the primary and secondary coolant. 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 4.0 MW using a 17-element core. An 8-element fuel exchange to enhance self protection was performed. A weld failure of fuel element B015 was discovered and reported to the NRC and the manufacturer, B&W. Otherwise, fuel performance has continued to be satisfactory with no known problems.

JE47

c. Changes in Operating Procedures

The list of new and/or revised procedures which were approved by the Nuclear Safeguard Committee during calendar year 1992 were as follows:

Proc. #	Title
2002	Reactor Operations-Precritical Startup Checklist and Shift Supervisor Approval
2006	Reactor Shutdown Checklist
9150	Operation and Calibration of Area Radiation Monitors
9312	Sealed Sources Leak Test
9041	Storage Pool Water Sampling and Analysis
9304	Routine Facility Radiation Surveys
9018	Charcoal Cartridge Analysis
9400	Environmental Monitoring
7204	Floor Fuel Storage Water
9250	Facilities Contamination Surveys
9300	Respiratory Protection
6080	Accidental Release of High Levels of Gaseous Activity to the Atmosphere
6090	Personnel Monitoring After Building Evacuation Emergency Situations
9250	Facilities Contamination Surveys
9057	Calibration Procedure for Eberline Model E-120 GM Survey Meter
9058	Calibration Procedure for Eberline Model RM-14 Rate Meter

Proc. #	Title
9060	Calibration Procedure for Bicron Model RSO-5 Survey Meter
9063	Calibration Procedure for Ludlum Model 2 GM Survey Meter
9065	Calibration Procedure for Bicron Model RSO-500 Survey Meter
9072	Calibration Procedure for Eberline Model RO-2 Survey Meter
7285	Calibration of Reactor Coolant Temperature Measuring System
2002	Reactor Operations-Precritical Startup Checklist and Shift Supervisor Approval
2006	Reactor Shutdown Checklist
7202	Control Rod Drop-Time
7226	Scram Insertion Delay Time Measurement
9501	Control & Accountability of Radioactive Sources
7241	Reactor Tank Level Transmitter Maintenance and Calibration Check
4950	Tagging Equipment Out of Service

A list of procedures which were deleted by the Nuclear Safeguards Committee were:

Proc. #	Title
3301	Sampling Log-D20
3302	Sampling Log H20
3303	Sampling Log Blanket Gas
4060	Top Reflector Dump Time Measurement

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
Mr. William Downs, Senior Reactor Operator
Mr. Dixon Parker, 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 to the full time staff, the NNRC employed the following graduate students on part time basis:

Mr. John Hawkinson
Ms. Kathleen Klee
Ms. Hannah Mitchell
Mr. Thomas Evans
Mrs. Hong Ning

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, 1992 through December 31, 1992, the total power generation of the GTRR was 196 MW hours. The reactor was operated a total of 158 hours: 8 hours at power levels equal to or less than 100 kW, 119 hours at power levels 100 kW to 1 MW, and 31 hours at power levels above 1 MW.

3. SHUTDOWNS

During this reporting period there were 3 unscheduled shutdowns. Table 1 gives details.

TABLE 1 UNSCHEDULED REACTOR SHUTDOWNS DURING 1992

Report	Date	Trip Initiation	Reason for Trip	Corrective Action
92-01	4/29	Low ion chamber voltage	Flux Amp #2 trip point had drifted.	Adjusted Flux Amp #2 R3 to correct value.
92-02	10/14	Low shield coolant level	low coolant level in tank led to pump cavitation.	Added coolant to TS-1. Also raised startup level requirements from 3 to 6 in.
92-03	12/10	Low H_2O flow	Flow restriction at impact bars in HX-2.	Cleaned HX-2 impact bars.

4. UNSCHEDULED MAINTENANCE ON SAFETY RELATED SYSTEMS AND COMPONENTS

There were approximately eighteen 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 1992, there were 27 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)	% Tech* Specs
1st Qtr	21.810	1.857×10^{-7}	3.300×10^{-5}	475	81.19
2nd Qtr	11.324	9.643×10^{-8}	1.508×10^{-5}	228	38.97
3rd Qtr	2.686	2.287×10^{-8}	1.000×10^{-5}	95.0	16.24
4th Qtr	3.653	3.111×10^{-8}	5.616×10^{-6}	62.7	10.72

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

(2) IODINES RELEASED

None Measurable
Lower Limit of Detection $< 5.76 \times 10^{-13}$ µCi/cc

(3) PARTICULATES

None Measurable
Lower Limit of Detection $< 6.3 \times 10^{-5}$ µCi
gross beta/gamma
Lower Limit of Detection $< 8.7 \times 10^{-6}$ 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	0.000016	8.00×10^{-11}	<1%
2nd QTR	0.000027	1.35×10^{-10}	<1%
3rd QTR	0.000048	2.40×10^{-10}	<1%
4th QTR	0.000017	8.50×10^{-11}	<1%

*Average release rate values are based on a Georgia Tech campus water discharge rate of 2.00×10^{11} ml/quarter.

(2) TOTAL GROSS RADIOACTIVITY (β/gamma)

	Total Release Ci	Avg. Release* Rate (μCi/cc)	% Tech Specs
1st QTR	2.40×10^{-5}	1.20×10^{-10}	<2%
2nd QTR	3.50×10^{-5}	1.75×10^{-10}	<2%
3rd QTR	5.70×10^{-5}	2.85×10^{-10}	<2%
4th QTR	2.00×10^{-5}	1.00×10^{-10}	<2%

*Average release rate values are based on a Georgia Tech campus water discharge rate of 2.00×10^{11} ml/quarter.

(3) TRITIUM

	Total Release Ci	Avg. Release* Rate (μCi/cc)	% Tech Specs
1st QTR	0.00123	6.15×10^{-9}	<1%
2nd QTR	0.00485	2.43×10^{-8}	<1%
3rd QTR	0.01841	9.21×10^{-8}	<1%
4th QTR	0.00494	2.47×10^{-8}	<1%

*Average release rate values are based on a Georgia Tech campus water discharge rate of 2.00×10^{11} ml/quarter.

(4) GROSS ALPHA RADIOACTIVITY RELEASED

None Measurable

Lower Limit of Detection -
$<7.07 \times 10^{-6}$ on 11/18/92

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

From Reactor Building

1st QTR . . . 3.67×10^{7} ml
2nd QTR . . . 5.37×10^{7} ml
3rd QTR . . . 9.61×10^{7} ml
4th QTR . . . 3.71×10^{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. None the less, 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 badge showed radiation exposure above background levels, film badge # 16 being the highest value, followed by badges # 9 & 11. Badge # 16 exposure is an anomaly of unknown origin possibly attributable to environmental conditions,e.g. rain & excessive heat. Badges #9 & 11 are located around the barn area (radioactive waste storage area). The exposure readings are probably due to the presence of 100 mCi of Radium-226 in the aforementioned area. During the months of August through October, various radium sources were collected from campus users, consolidated for shipment and shielding, and disposed of as radioactive waste.

(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.

All of the film badge locations were similar

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

(e) 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.

NEELY NUCLEAR RESEARCH CENTER
ENVIRONMENTAL RADIATION SURVEILLANCE*
1992

	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	M	M	M	M
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	M	M	M	M
09815	M	M	M	M	M	M	M	M	M	M	M	M
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	M	M	M	M	M	M	M	M	M	M
09820	M	M	M	M	M	M	M	M	M	M	M	M
09821	M	M	M	M	M	M	M	M	M	M	M	M
09822	M	M	M	M	M	M	M	M	M	M	M	M
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	M	M	M	M	M	M	M	M	M	M
09827	M	M	M	M	**		M	M	M	M	M	M
09828	M	M	M	M	M	M	M	M	M	M	M	M
09829	**		**		M	M	M	M	M	M	M	M
09830	M	M	M	M	M	M	M	M	M	M	M	M

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

** Damaged film badge.

NEELY NUCLEAR RESEARCH CENTER
ENVIRONMENTAL RADIATION SURVEILLANCE*
1992

	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	M	M	M	M	M	M	M	M	M	M	M	M
09802	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09803	M	M	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	M	M
09805	M	M	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	M	M
09807	M	M	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	M	M
09809	M	M	10	10	20	20	M	M	M	M	M	M	30	30
09810	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09811	M	M	M	M	20	20	10	10	M	M	M	M	30	30
09812	M	M	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	M	M
09814	M	M	M	M	M	M	M	M	M	M	M	M	10	10
09815	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09816	M	M	M	M	40	40	M	M	M	M	M	M	40	40
09817	M	M	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	M	M
09819	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09820	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09821	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09822	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09823	M	M	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	M	M
09825	M	M	M	M	M	M	M	M	M	M	M	M	M	M
09826	M	M	M	M	M	M	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	M	M	M	M	M	M	M	M
09829	**		**		M	M	M	M	M	M	M	M	M	M
09830	M	M	M	M	M	M	M	M	M	M	M	M	M	M

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

** Damaged film badge.

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 then 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 = 0.54 rem

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

Average annual level -	49	mrem
Highest annual level -	150	mrem
Lowest annual level -	< 10	mrem.

d. Person-Rem for Ga Tech campus users.

Person-Rem = 2.04 rem

The highest, lowest and average levels of personnel exposure due to experimental use of radionuclide primarily P-32, I-131, and S-35 and/or the use of x-ray machines.

Average annual level -	68	mrem
Highest annual level -	250	mrem
Lowest annual level -	< 10	mrem.

e. Category of exposure

NNRC Radiation Workers

Annual exposure	# Radiation workers
< 10 mrem	7
10 mrem - 49 mrem	7
50 mrem - 99 mrem	2
100 mrem - 149 mrem	1
150 mrem - 199 mrem	1
≥ 200 mrem	0

Ga Tech On-campus users

Annual exposure	# Radiation workers
< 10 mrem	32
10 mrem - 49 mrem	17
50 mrem - 99 mrem	5
100 mrem - 149 mrem	0
150 mrem - 199 mrem	7
≥ 200 mrem	1

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.

APPEND.X A

Facility Modifications

FACILITY MODIFICATION REQUEST

92-001

PICOAMMETER MONITORS FOR
PICOAMMETER #1 AND #2

The Nuclear Safeguards Committee approved a change in instruments for power level measuring channels #1 and #2 from the old GE vacuum tube picoammeter to Keithleys' Model #485/4853.

It is desirable to have an alarm added to the Keithleys' such that once the power level deviates by more than 10% from steady state operation, the alarm activates with sonalert. The device is intended as an additional aid to the operator to maintain "Cognizance" of reactor power level.

The attached document gives more details.

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 92-001

TITLE: PICO AMMETER MONITOR

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	1-24-92
APPROVALS:		
Director NNRC:	R. A. Karam	1/27/92
Nuclear Safeguards Committee:		

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX

FACILITY MODIFICATION NO: 92-001

TITLE: PICOAMMETER MONITOR

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
	NONE		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
	NONE		

Reviewed By: ____________________ Date: __________

Picoammeter Monitor

1.0 PURPOSE

The purpose of this facility modification is to add a device that will give an audible alarm when the one of the Reactor Picoammeters deviates from a preset point.

2.0 SCOPE

The proposal is to add the Picoammeter Monitor that will be connected to two Keithley model 485 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 Schematic for the Picoammeter Monitor

4.3 Related Procedures

4.3.1 None

5.0 SYSTEM DESCRIPTION

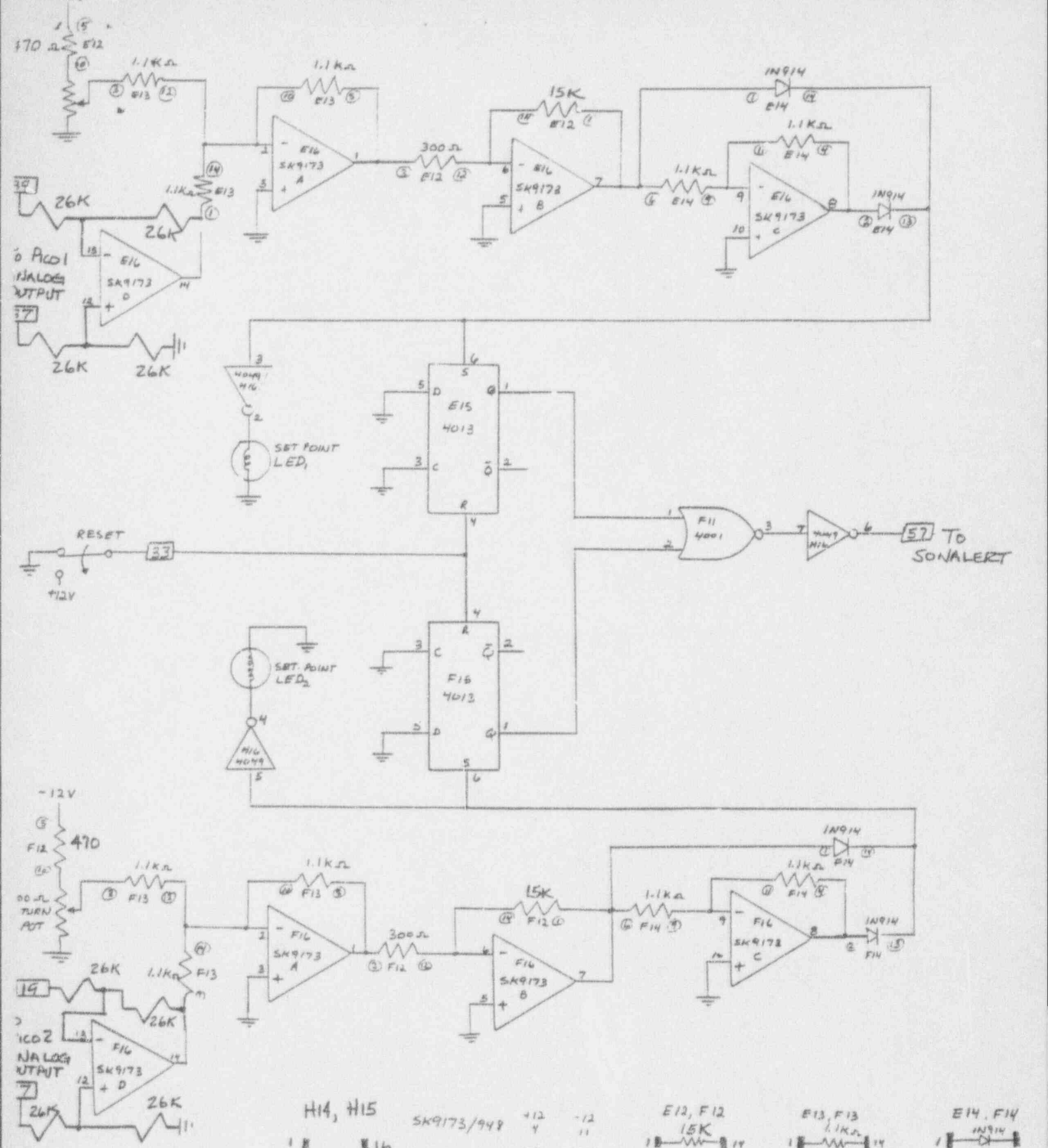
5.1 Picoammeter Monitor

The Picoammeter Monitor (PM) will be connected to the Picoammeters analog output. The PM will be energized only when Reactor power is stablized. When energized, the PM comes on in the alarm condition. The operator will adjust the #1 set point potentiometer until the #1 set point LED is a its brightest level; this action will be repeated for #2 set point potentiometer. The operator will then press the reset switch to turn the alarm condition off and silence the Sonalert. Light Emit Dio

Should either Picoammeter indication increase or decrease by approximately 10% of full scale value (for what ever scale it may be on) the PM will go to alarm mode. This condition requires operator action to reset the PM. A PM alarm could be caused by a change in Reactor power level (autocontroller problem) or malfunction in the one of the Picoammeter measuring circuits.

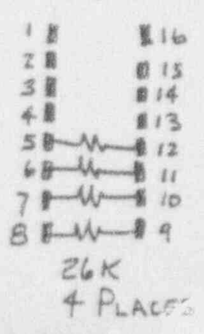
During power level changes the operator would turn the PM off until a new level is reached. At the new level the set points would be adjusted.

PICOAMMETER MONITOR

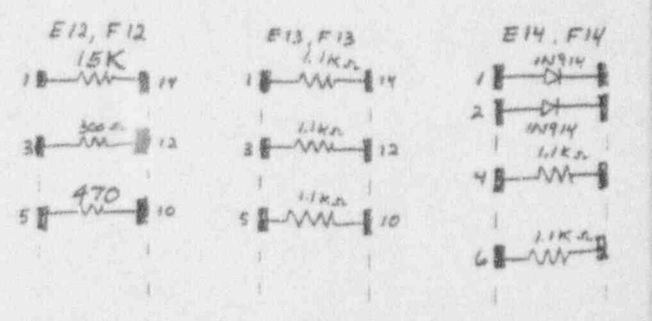


SIGNAL	BUS PIN#
+12V	21, 73
-12V	25, 77
	29, 69
	33
V.	37
/IN 2	17
LARM	57
.ED 1	61
.ED 2	65

Ω POTS



SK9173/948	+12 4	-12 11
	+12	GND
MC14001	14	7
MC14013	14	7
MC14049	1	8



APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 92-002

TITLE: REPLACEMENT OF REACTOR PRIMARY AND SECONDARY COOLANT TEMPERATURE MEASURING SYSTEMS

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	3-10-92
APPROVALS:		
Director NNRC:	R. A. Karam	3/10/92
Nuclear Safeguards Committee:		

Approved 3/12/92

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 92-002

TITLE: REPLACEMENT OF REACTOR PRIMARY AND SECONDARY COOLANT TEMPERATURE MEASURING SYSTEMS

DRAWINGS:

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

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE
2002	REACTOR OPERATIONS- PRECRITICAL		
	STARTUP CHECKLIST AND SHIFT		
	SUPERVISOR APPROVAL		
7276	TEMPERATURE ~~CALIBRATED~~ RECORDER		
	CALIBRATION RTD INPUT		
7283	RTD CALIBRATION		

Reviewed By: ______________________ Date: __________

REPLACEMENT OF REACTOR PRIMARY AND SECONDARY COOLANT TEMPERATURE MEASURING SYSTEMS

FACILITY MODIFICATION 92-002

1.0 PURPOSE

The purpose of this facility modification is to replace the existing primary and secondary coolant temperature measurement, alarm and scram systems with modern equipment.

2.0 SCOPE

The proposal is to replace the resistance temperature detectors (RTD) and temperature recorders with new RTDs, digital panel meters (DPM) and new recorders.

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 Operator's Manual, Omegarometer series # 10294ML-99 (manual for the DPM)

4.2 Operator's Manual, Omegarometer series # 10262ML-99 (manual for DPM dual setpoint control)

4.3 Operator's Manual, Omegaline Recorder model 620-2V

5.0 SYSTEM DESCRIPTION

5.1 Existing system description

Equipment:
a. 2 each RTDs for primary coolant
b. 2 each RTDs for secondary coolant
c. 1 each temperature recorder for primary coolant
d. 1 each temperature recorder for secondary coolant

There is an RTD in the primary coolant heat exchanger inlet and outlet; also an RTD in the secondary coolant heat exchanger inlet and outlet. One temperature recorder will accommodate 2 RTDs. There are relays and mechanically actuated switches in each recorder that make up the scram, high temperature annunciator and low temperature annunciator contacts.

5.2 Primary Problem With Existing System

The temperature recorder is a single input type recorder with the two RTD inputs sequentially switched with a mechanically driven stepper switch. This arrangement is satisfactory for an input device that has high resistance but the resistance of the RTD is 100 ohms near the mid range of the measured temperature. A resistance change of 0.2 ohm is approximately equal to 1°F. The stepper switches require frequent cleaning and still their performance is poor.

5.3 Other Problem With Existing System

The recorder scale is 11 inches wide for a span of 100°F; this makes resolution of 0.1°F very difficult.

Replacement parts are difficult to impossible to locate.

5.4 Proposed system description

Equipment:
4 each Omega model PR-11-2-100-1/4-6-A platinum RTDs
4 each Omegarometer model DP2101R2 Digital Panel Meters
4 each Omegarometer dual setpoint controls (in DPM)
2 each Omegaline model 620-2V recorders

D_2O measurement, alarm and scram circuit is shown in figure 1. The low temperature annunciator requires a closed contact across TC10 and TC11 to enable reset. Both DPM low set points will be at 53°F, should either DPM indication be less than this point the Low D_2O Temp annunciator will alarm. This low temperature alarm is only for the reactor operator information.

The High D_2O Temp scram requires a closed contact across TC6 and TC7 to enable reset. The high set point for TRA-D1-1 DPM is 137°F and the high set point for TRA-D1-2 DPM is 123°F. For a closed contact to be present at TC6 and TC7 there must be power to TRA-D1-1 DPM and the indicated temperatures less than the high set points.

The High D_2O Temp annunciator requires a closed contact across TC8 and TC9 to enable reset. This is furnished by contacts of K1.

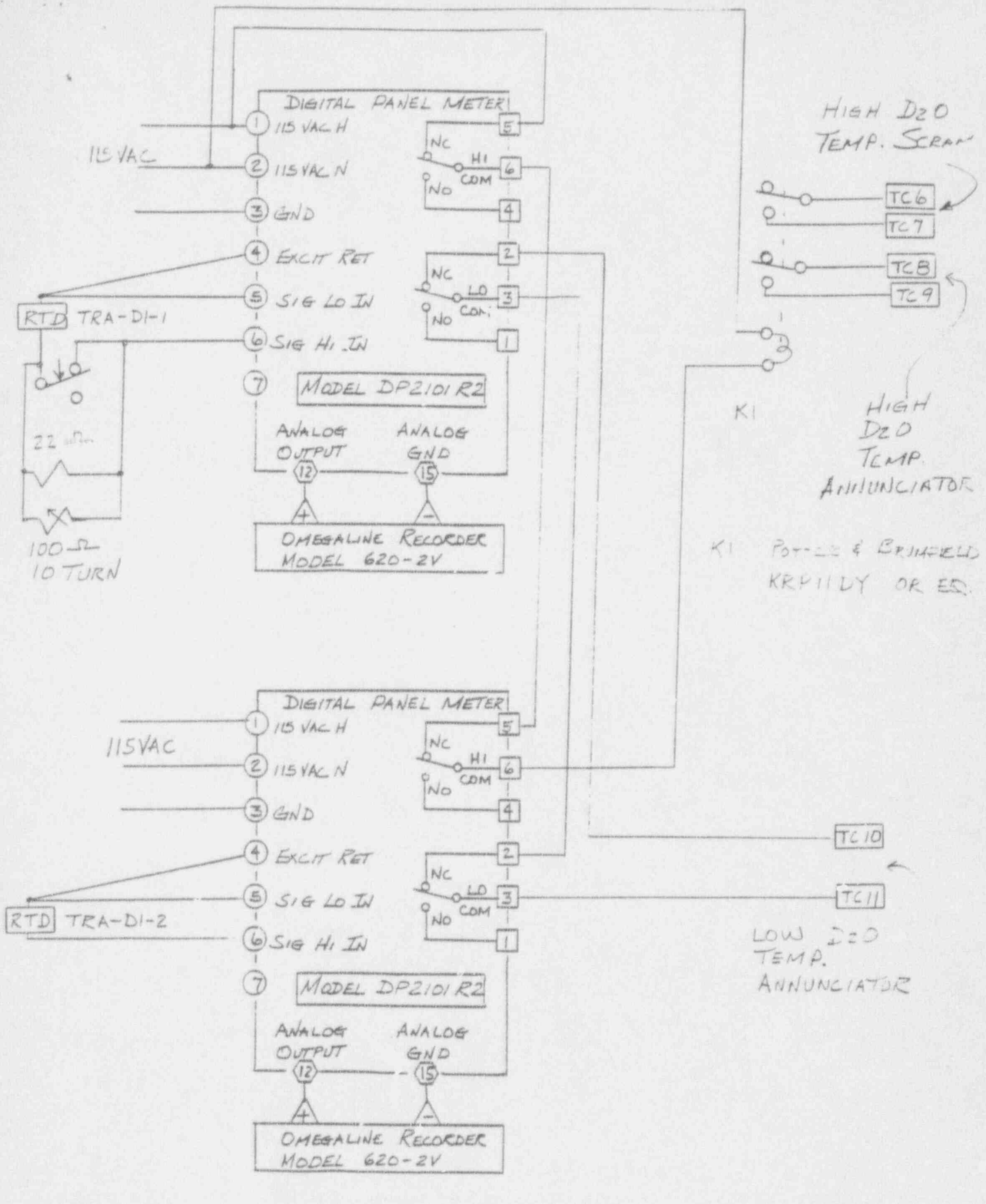
TRA-D1-1 RTD is connected through normally closed contacts of a spring loaded switch. Weekly pre-critical startup checklist requires the operator to test the High D_2O Temp scram point. The operator will press and hold the spring loaded switch while adjusting the 100 ohm 10 turn pot to test the scram set point. Once the switch is released the pot and resistor are shorted across.

H_2O measurement, alarm and scram circuits are shown in figure 2. TRA-H1-2 DPM high set point is 134°F. Circuit description is similar to the D_2O primary system.

Figure 3 shows figure 1 and figure 2 connection into the existing wiring.

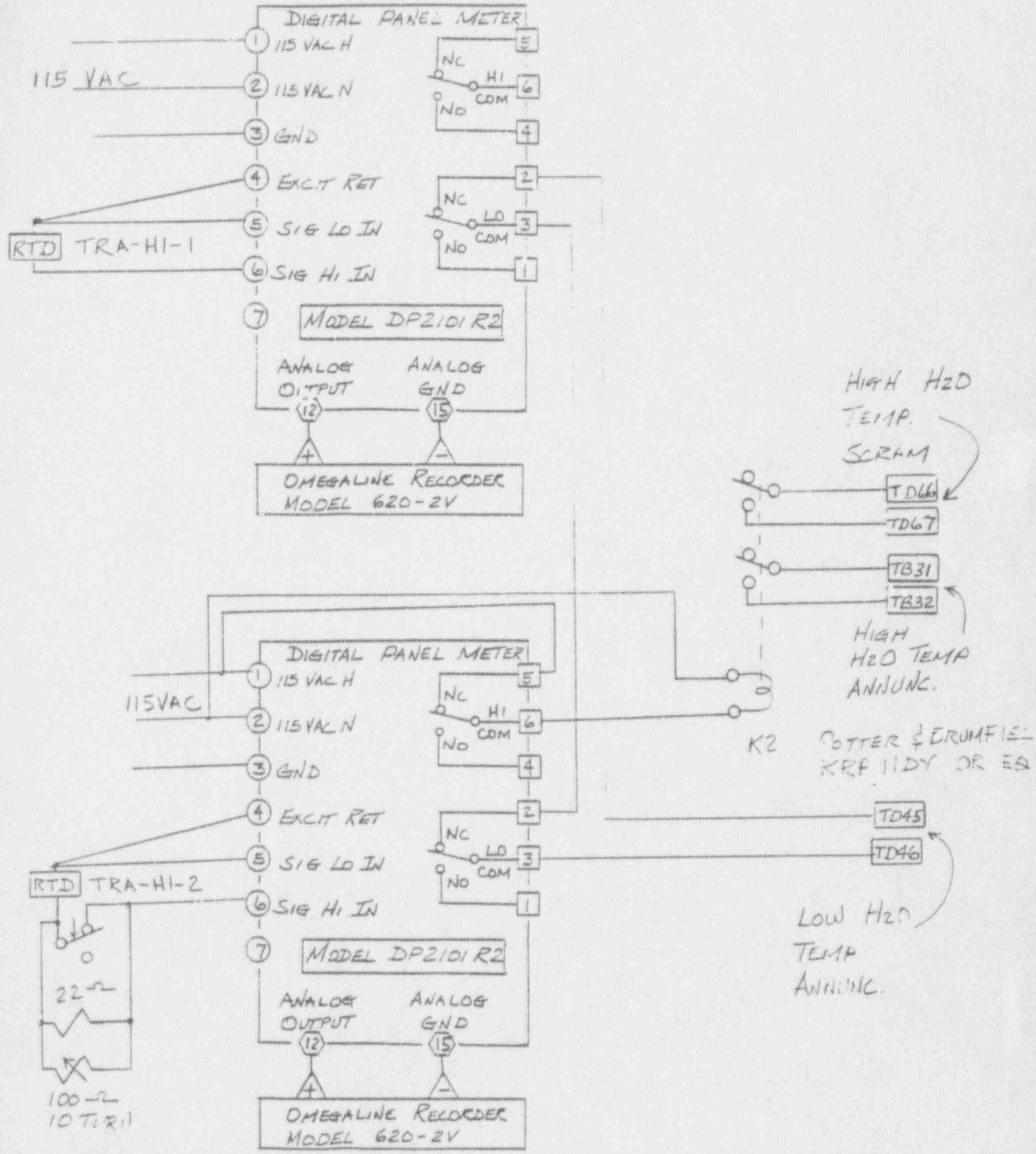
5.5 Technical Specifications and Procedure Requirements

Appendix A contains reprints of pages 7, 8 and 11 of the Technical Specifications and page 6 of procedure 7250. These are requirements that are applicable to this facility modification.



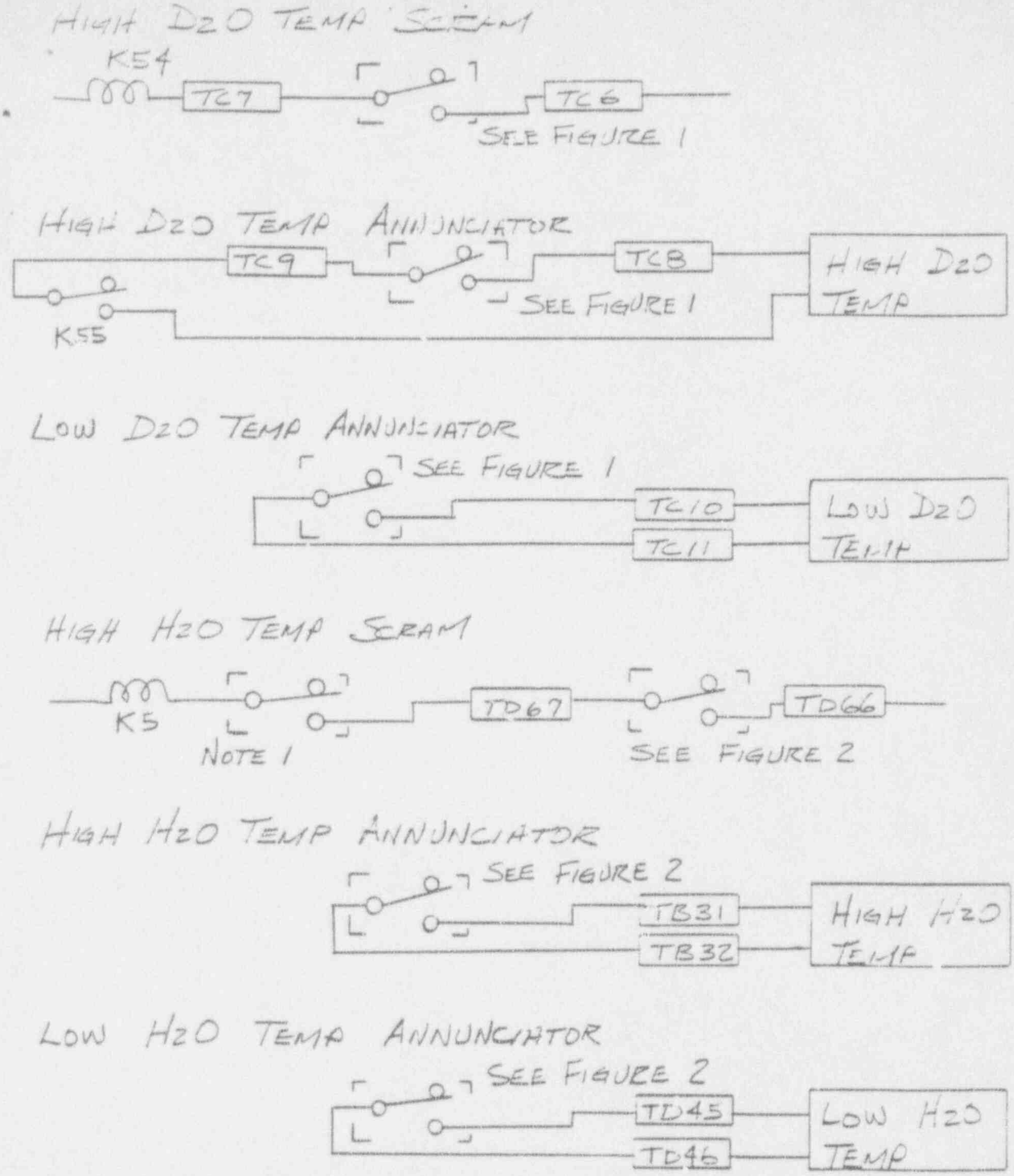
D2O TEMPERATURE MEASUREMENT, ALARM & SCRAM CIRCUIT

FIGURE 1



H_2O TEMPERATURE MEASUREMENT, ALARM & SCRAM CIRCUIT

FIGURE 2



NOTE 1:

K5 IS DELYED SCRAM RELAY.
IN ADDITION TO HIGH H_2O TEMP, THERE SIX(6) OTHER CONTACTS THAT MUST BE CLOSED BEFORE THE DELAYED SCRAM RELAY CAN BE ENERGIZED.

SCHEMATIC - D_2O & H_2O TEMPERATURE CIRCUITS CONNECTION TO EXISTING WIRING

FIGURE 3

APPENDIX A

2.1.2 SAFETY LIMITS IN THE NATURAL CONVECTION MODE

APPLICABILITY

This specification applies to the interrelated variables associated with the core thermal and hydraulic performance in the natural convection mode of operation.

SPECIFICATION

The reactor thermal power shall not exceed two (2) kW.

BASIS

Experience with the GTRR has shown that no damage to the core and no boiling occurs without forced convection coolant flow at power levels up to two kW.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 LIMITING SAFETY SYSTEM SETTINGS IN THE FORCED CONVECTION MODE

APPLICABILITY

Applies to the settings of those instruments monitoring the safety limits.

OBJECTIVE

To assure automatic protective action is initiated before a safety limit is exceeded.

SPECIFICATION

The safety system trip settings shall be as follows:

Thermal Power	5.5MW
Reactor Coolant Flow	1625 GPM
Reactor Outlet Temperature	139 °F

BASIS

The trip settings are chosen so that the reactor is operated with no incipient boiling. An analysis was made showing that at 1800 gallons per minute total coolant flow, five MW thermal power an inlet reactor coolant temperature of 114°F and the application of all the engineering uncertainty factors, a maximum fuel surface temperature 8°F less than the local D_2O saturation temperature might occur.(1)

REFERENCE

(1) Letter, R. S. Kirkland to USAEC, October 22, 1971, Response No. 10.

2.2.2 LIMITING SAFETY SYSTEM SETTINGS IN NATURAL CONVECTION MODE

APPLICABILITY

Applies to the values of safety system settings when operating in the natural convection mode.

OBJECTIVE

To assure the reactor is not operated at a power level sufficient to cause fuel damage.

SPECIFICATION

The reactor thermal power safety system setting shall not exceed 1.1 kW when operating in the natural convection mode.

BASIS

In the natural convection mode of reactor operation the main coolant pumps are not operating. The reactor isolation valves may be closed so that only internal, natural convection is available for cooling. Experience with the GTRR has shown that the reactor can be operated at one kW indefinitely without exceeding a bulk reactor temperature of 123°F.

APPENDIX A

TRIP (SCRAM) CONDITIONS AND SetpointS

TYPE OF TRIP	MODE 1	MODE 2
Power Trip	1.25 MW	5.5 MW
Period Trip	15 second -10 second	same
Reactor Tank Low Level	> 66"	same
Low D_2O Flow	1000 gpm	1625 gpm
High D_2O Temp	125° F	139° F
Low Ion Chamber Voltage	650 VDC	same
Calibrate Switches	Not in Operate Position	same
Magnetic Actuator Amp	Not Energized	same
Reactor Isolation Valves Not Open	Valves Leave Open Position	same
Drain Valves Open	Valves Open	same
No D_2O Overflow	No Reactor Vessel overflow	same
Doors Open	20 psig on Truck, Personnel, Emergency door(s)	same
High H_2O Temp	134° F	same
Low H_2O Flow	900 gpm	900 gpm
Control Air Pressure	60 psig	same
Low Shield Coolant Flow	30 gpm	same
High Shield Coolant Temp	120° F	same
Low Bismuth Coolant Flow	0.75 gpm	same
High Bismuth Coolant Temp	120° F	same

TABLE 3.1

REQUIRED SAFETY CHANNELS

Channel	Setpoint	Minimum No. Required	Function
Start up	2 cps	1 [a]	Minimum countrate permissive rod withdrawal interlock
Period trip	$\leq$10 sec (pos or neg)	2 [c]	Scram
Power trip	5.5 MW	2 [c]	Scram
Low D_2O flow	1625 gpm	2 [b] [c]	Scram
High D_2O Temperature	139°F	2 [c]	Scram
Low D_2O Level	$\leq$12" below over flow	2 [c]	Isolate reactor vessel Scram Initiate ECCS
No D_2O Overflow	-	1	Scram
Manual scram	-	1	Scram
Reflector drain	-	1	Backup scram
Containment doors open	-	1 per airlock	Scram
Reactor isolation valves closed	-	2 [c] per valve	Scram

(a) Required during startup and for operation with less than 1 decade overlap between the startup channel and the pico-ammeter channel.

(b) Not required for natural convection operation

(c) One of the twelve required safety channels may be bypassed for a period not to exceed 8 hours for test, repair, or calibration

APPENDIX A

10 CFR 50.59 SAFETY EVALUATION QUESTIONNAIRE

FACILITY MODIFICATION NO: 92-003

TITLE: REPLACEMENT OF REACTOR PRIMARY AND SECONDARY COOLANT TEMPERATURE PROBE WELLS

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	6-10-92
APPROVALS:		
Director NNRC:	R. [illegible]	6/10/92
Nuclear Safeguards Committee:	Approved 6/25/92 RAK	

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

FACILITY MODIFICATION DOCUMENTATION CHECKLIST
APPENDIX B

FACILITY MODIFICATION NO: 92-003

TITLE: REPLACEMENT OF REACTOR PRIMARY AND SECONDARY COOLANT TEMPERATURE PROBE WELLS

DRAWINGS:

NUMBER	TITLE	REVISED BY	DATE
045-51-001	D_2O PIPING DIAGRAM		
SH 2 OF 2	AND LAYOUT		

PROCEDURES:

NUMBER	TITLE	REVISED BY	DATE

Reviewed By: ______________________ Date: __________

REPLACEMENT OF REACTOR PRIMARY AND SECONDARY COOLANT TEMPERATURE PROBE WELLS

FACILITY MODIFICATION 92-003

1.0 PURPOSE

The purpose of this facility modification is to replace the existing primary and secondary coolant temperature probe wells with wells sized for the new resistance temperature detectors (RTDs). Installation of the new RTDs was approved by Facility Modification 92-002.

2.0 SCOPE

The proposal is to replace the temperature probe wells.

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 Temperature Handbook pages B-11 and B-28

5.0 SYSTEM DESCRIPTION

5.1 Existing temperature probe wells

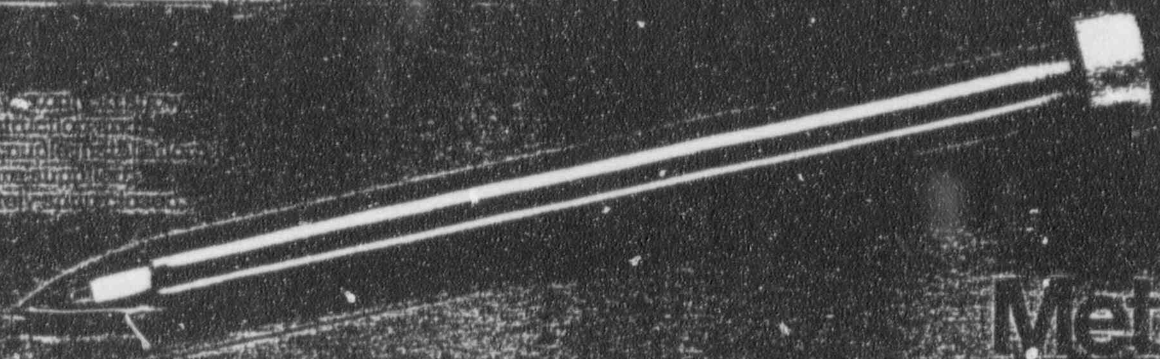
Equipment:
a. 2 each temperature probe wells for primary coolant
b. 2 each temperature probe wells for secondary coolant

5.2 Problem with existing temperature probe wells

The existing temperature probe wells inside diameter is approximately 0.375 inches and the new RTD outside diameter is 0.25 inches. Initially the plan was install a brass sleeve over the new RTDs to fill the space and use the existing wells. More efficient heat transfer will occur by using wells sized for the new RTDs.

5.4 Proposed temperature probe wells

Equipment:
4 each Omega model 3/4-260S-U4 1/2-304SS



Meta
Protection Tubes

Nominal 3/4" Schedule 40 Black Steel Pipe (.824 I.D. x 1.050 O.D.)			
THREAD	CAT. NO.	LENGTH	PRICE
3/4"	3/4"-PTBS-12	12"	$28
3/4"	3/4"-PTBS-18	18"	31
3/4"	3/4"-PTBS-24	24"	33

Nominal 1/2" Schedule 40 Black Steel Pipe (.622 I.D. x .840 O.D.)			
THREAD	CAT. NO.	LENGTH	PRICE
1/2"	1/2"-PTBS-12	12"	$25
1/2"	1/2"-PTBS-18	18"	28
1/2"	1/2"-PTBS-24	24"	30

Nominal 3/4" Schedule 40-304 SS Pipe (.824 I.D. x 1.050 O.D.)			
THREAD	CAT. NO.	LENGTH	PRICE
3/4"	3/4"-PTSS-12	12"	$43
3/4"	3/4"-PTSS-18	18"	52
3/4"	3/4"-PTSS-24	24"	70

Nominal 1/2" Schedule 40-304 SS Pipe (.622 I.D. x .840 O.D.)			
THREAD	CAT. NO.	LENGTH	PRICE
1/2"	1/2"-PTSS-12	12"	$40
1/2"	1/2"-PTSS-18	18"	48
1/2"	1/2"-PTSS-24	24"	67

Technical DATA
Male NPT thread on one end, spun closed on the other end.

Material	Max. Continuous Operating Temp.	Mech. Strength Cold	Mech. Strength Hot	Applications	Remarks
304SS	1650°F	Excellent	Fair	Food processing, Dairy products, Petroleum prod., mild acids, alkalies	Embrittles in 800°F to 1400°F range.
Black Steel	1200°F	Excellent	Good	Molten Babbit, Tin, Lead and Magnesium	Low cost

How to order:
Specify thread—catalog number.
Example: 3/4"-PTSS-12.

Mounting Flanges and Bushings with standard N.P.T. thread available upon request. Consult Sales for Pricing.

Discount Schedule:

1 - 10 units	Net
11 - 24 units	10%
25 - 100 units	20%
101 and up	Consult Sales

Quantity discount applies to like tubes. Other sizes, materials and configurations by quotation.

Series 260S

Standard Threaded Well for 1/4" Diameter Elements

Application:
Standard Length, 1/4" Stem, Bimetal thermometers; #20 gage thermocouple elements; unarmored liquid-in-glass test thermometers. Other temperature sensing elements having .252 in. maximum diameter.

Connection Size:
1/2", 3/4" and 1" NPT are standard. Other thread sizes are available upon request.

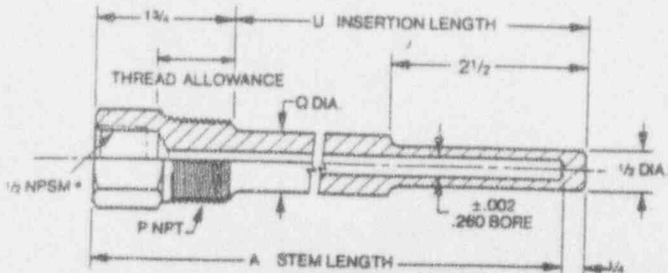
Protective Coatings For Thermowells—
- Resist Corrosion
- For Chemical Baths
- Color-Coded Sensors for Process Control
- Available in PFA Teflon®, Epoxy and other materials

Please consult our Sales Department for complete information.

Materials:
Brass (ASTM B-16); Carbon Steel (C-1018); Stainless Steel A.I.S.I. 304 & A.I.S.I. - 316; Monel. Wells are available also in special materials, prices on request.

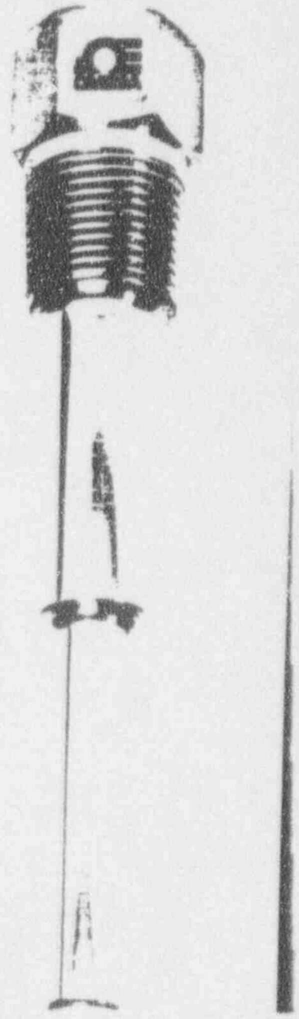
Cap and Chain Options: For Brass cap, add $4 to price and add suffix—CC (Brass) to the end of the catalog number.
For 304SS cap, add $7 to price and add suffix—CC (304SS) to the end of the catalog number.

Series 260S — General Use



All dimensions are in inches.

*NPSM internal pipe thread will accept both NPT and NPS male threads.



When ordering probes with NPT Fittings specify this stem length.

Most Popular Sizes ▶ 1/2" NPT	1/2"-260S-U 2 1/2 □	4	2 1/2	—	$ 22.00	$ 30.00	$16.50	$16.50
	-U 4 1/2 □	6	4 1/2	1/4	26.50	36.00	20.00	20.00
	-U 7 1/2 □	9	7 1/2	"	39.50	53.00	31.50	31.50
	-U10 1/2 □	12	10 1/2	"	48.00	65.00	41.00	41.00
	-U13 1/2 □	15	13 1/2	"	69.00	93.00	53.00	53.00
	-U16 1/2 □	18	16 1/2	"	82.50	111.00	63.50	63.50
	-U22 1/2 □	24	22 1/2	"	106.00	143.00	74.00	74.00
Most Popular Sizes ▶ 3/4" NPT	3/4"-260S-U 2 1/2 □	4	2 1/2	—	22.00	30.00	16.50	16.50
	-U 4 1/2 □	6	4 1/2	3/4	26.50	36.00	20.00	20.00
	-U 7 1/2 □	9	7 1/2	"	39.50	53.00	31.50	31.50
	-U10 1/2 □	12	10 1/2	"	48.00	65.00	41.00	41.00
	-U13 1/2 □	15	13 1/2	"	69.00	93.00	53.00	53.00
	-U16 1/2 □	18	16 1/2	"	82.50	111.00	63.50	63.50
	-U22 1/2 □	24	22 1/2	"	106.00	143.00	74.00	74.00
Most Popular Sizes ▶ 1" NPT	1"-260S-U 2 1/2 □	4	2 1/2	—	29.00	35.50	22.00	22.00
	-U 4 1/2 □	6	4 1/2	7/8	36.50	49.00	26.50	26.50
	-U 7 1/2 □	9	7 1/2	"	48.00	65.00	34.50	34.50
	-U10 1/2 □	12	10 1/2	"	60.00	80.00	46.00	46.00
	-U13 1/2 □	15	13 1/2	"	83.00	111.50	61.00	61.00
	-U16 1/2 □	18	16 1/2	"	97.00	130.50	71.50	71.50
	-U22 1/2 □	24	22 1/2	"	132.50	164.00	83.00	83.00

Material - 304 S.S., Carbon Steel, etc.
Insertion length dimension
Bore size - inch
External thread - NPT

HIGHLIGHTED MODELS STOCKED FOR FAST DELIVERY.

essure - Temperature rating - lbs. per sq. inch

Materials							
Brass	5000	4200	1000	—	—	—	—
Carbon Steel	5200	5000	4800	4600	3500	1500	—
A.I.S.I. - 304	7000	6200	5600	5400	5200	4500	1650
A.I.S.I. - 316	7000	7000	6400	6200	6100	5100	2500
Monel	6500	6000	5400	5300	5200	1500	—

See Page B-27 for Maximum Fluid Velocity.

These wells are compatible with OMEGA® NB1, NB2 (pgs. B-3 & B-4), PR12, PR14 (pg E-5), and NPT style probes (pg. B-4), as well as DialTemp™ Thermometers (Section S).

To order, please specify: 1. Complete Type Number
2. Material
3. Cap & Chain — If Desired

Discount	
1-10	Net
11-24	10%
25-100	20%
101 and up	Consult Sales

Discounts apply to similar thermowell types

Increase Response Rate! Use OT-201 Conductive Silicon Paste (See page J-6)

5.5 The new temperature probe well material is 304SS (stainless steel) which is same as existing wells (ref dwg # 045-51-001 sh 2 of 2).

5.6 Included are copies Omega Temperature Handbook pages B-11 and B-28.