State University of New York at Buffalo

NUCLEAR SCIENCE AND TECHNOLOGY FACILITY

March 10, 1983

FILE RE:

Docket 50-57

License R-77

Cecil O. Thomas, Chief
Standardization and Special Projects Branch
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Thomas:

In support of our application for renewal of our operating license R-77, I have enclosed a partial response to your questions dated January 21, 1983. Questions 5 and 10 require additional effort, and I anticipate submittal of answers to these two questions within two weeks.

Sincerely,

Louis G. Henry

Louis G. Henry
Acting Director

amf
enc.
cc: Charles C. Thomas
Los Alamos National Laboratory

A020

1.a Section V-D of the Hazards Summary Report (HSR), Revision 2, September 23, 1983 presents an excess reactivity requirement of 2.37% delta K/K to compensate for fuel burnup and fission product accumulation. This number was calculated based on an assumed continuous 1,000 day operation at 2Mw. Core loadings at NSTF are typically operated for six months to a year, and thus there is no need to include nearly three years of operation when calculating excess reactivity requirements. The figure of .45% delta K/K, presented in the SAR update, is therefore more appropriate, and should be used in lieu of the original figure.

1.b The period of reactor operation necessary to equilibriate samarium reactivity in the NSTF reactor is exceedingly long. It may therefore be considered a constant when calculating or measuring reactivities for the purpose of assuring compliance with reactivity limits and, therefore, for establishing these limits. The "nominal period of operation" implied by section V-F of the HSR is, therefore, not realistic, and said reference should be ignored.

1.c It is requested that:

(1) Specification 3.1.a be changed to read "the shutdown margin relative to the cold xenon-free critical condition shall be greater than or equal to the worth of the most reactive safety control blade plus 0.5% delta K/K".

(2) Specification 3.1.c be changed to read "the worth of individual experiments shall be limited as follows:

Experiment	Maximum Worth
movable	± 0.3% delta K/K
unsecured	± 0.6% delta K/K"

(3) The last sentence on page 8 be deleted in its entirety.

The reactivity limits of individual experiments fall within the window of safety established by the NSTF pulse test program, which was previously authorized by the Commission.

1.d The submittal dated 5/8/74 included an analysis of the impact of extending allowable fuel burnup on the postulated results of the design basis accident. Specifically, re-calculation of the fission product inventory was necessary. The figures, as presented in the May 8th, 1974 submittal are correct, and supersede the referenced parts of Section VII-c.

2. It is requested that Section 3.3.2.c (not 3.3.2.b) be replaced with the following: "The outputs of the building air, stack air, and stack particulate monitors shall be recorded on a strip chart."

3.a A modified Table 3.1 is provided.

3.b & c Already deleted as per response to question 1-c, points 1 and 2.

3.d A modified page 5 to the SAR update is provided.

4. Diagrams and curves are provided as requested.

6. It is our intention to install an electro-mechanical flow velocity sensor in the stack exhaust line. The electronic signal will be transmitted to an audible and visual alarm in the reactor control room. We anticipate installation within sixty days.

7. Fq (not Fg) = 3.5. The values used to calculate a f_0 in the original HSR were used with the following modifications:
(1) the scram setting factor (1.05) was deleted, as it is not appropriate for calculating a safety limit.
(2) the film correlation factor (1.2) was also removed. This phenomenon is incorporated in the value of "0.7" in the correlation provided in Figure 6.4 of the Appendix D, Analysis to Support Safety Limits and Safety Limit Settings.

(3) the flow distribution factor of 1.06 was increased to 1.18 (a more conservative value). Fq is therefore calculated as follows:

$$Fq = (1.02 \times 1.01 \times 1.18 \times 1.07 \times 1.08) \times 2.5$$

$$Fq = 3.5$$

8. The primary water level in the reactor tank is recorded each morning of operation. This reading is logged. A graph of cumulative water addition is also kept in the primary water log. There has been no evidence of any further leakage. The abandoned primary outlet pipe can function as a "tell-tale" drain, should the welded closure on this penetration begin to leak. The other abandoned pipes serve no useful purpose.

9. An interlock has been installed on the primary outlet pool isolation valve which prevents operation of the primary pump if the valve is not full open.

11. & 12. Irradiated fuel storage is addressed by Section 5.7.2 in the new technical specifications. In that section, there are six stipulations which limit the storage of fuel outside of the reactor tank. The maximum number of fuel assemblies to be stored outside of the reactor tank would be limited by these stipulations, and a fixed number may not be calculated at this time. We propose that if greater than 24 fuel assemblies (as already approved by the NRC) are to be stored in the hot cell, criticality and shielding analyses will be performed, and the configuration, analyses, and procedures will be approved by the Nuclear Safety Committee, as required by Stipulation b. to Section 5.7.2. Such analyses would include fuel burnup histories, and take advantage of our experience to date in out-of-tank storage of irradiated fuel.

Currently, facilities for storage of fuel outside of the reactor tank or hot cell do not exist. If at a future date it is necessary to construct such a facility, it would be constructed, approved, and operated as stipulated in Section 5.7.2. Lacking a specific design, personnel exposures cannot be accurately predicted; however, stipulation d., which requires that shielding be adequate to reduce radiation levels at the boundary of a storage facility to less than 100 mr per hour, implies that the fuel storage facility would be well within the level of common experience at NSTF. Furthermore, stipulation a., which requires K_{eff} be less than or equal to .85, coupled with stipulation b., requiring approval by the Nuclear Safety Committee, would seem to assure that an inadvertent criticality would be avoided. With regard to potential safeguards considerations, any fuel storage facility would be located within the containment; i.e., within the Controlled Access Area. The diversion of fuel assemblies from such facilities would likely be more difficult than diverting the material from the reactor tank itself. I foresee, therefore, no negative impact on our ability to prevent or detect the diversion of Special Nuclear Material.

13. In the event of a power failure, the triggering solenoids on the Pratt dampers will open under spring load, and the dampers will close and seal. The solenoids are powered by the emergency generator, and therefore within approximately ten seconds we would have the capability of reopening the dampers. It is unlikely, however, that we would reopen the dampers, because the residual hydraulic pressure would be expended, and it is unlikely that we would be able to reclose the dampers, should it be necessary. If, however, the dampers close and are not reopened, electrical power is not required to maintain the dampers in the closed position. One would therefore expect a slow bleed-off of containment air through the filtered emergency exhaust vent. If the situation existed where the dampers were open, electrical power had been interrupted, and airborne contamination existed, it would be possible to disable the powerhouse fan and there would therefore be no fans drawing air into or out of the containment. A slow bleeding outward of contaminated air would therefore be possible. This, however, would not be any more significant than the bleed past the truck door and airlock gaskets, which would also leak in the absence of electrical power.

14. Primary coolant flow rate may be controlled or changed in two different manners. Currently we control flow by opening or partially closing the primary coolant isolation valve situated near the discharge of the primary pump. It is possible, and we propose to, control the flow rate in an alternative manner should our new technical specifications be approved. A four-inch bypass line which bypasses flow around the primary pump can also be used to control flow rate. We currently maintain this bypass line in a locked shut configuration.

15. The maximum power level in the fission plate when it is in place in the thermal column is less than 1 watt. No provisions for dissipating this power are required other than the ambient air flow and conduction into the thermal column graphite.

16. The NSTF liquid radwaste handling system includes five tanks. There are two 250-gallon tanks, two 600-gallon tanks, and one 10,000-gallon tank. They are buried in an open vault between the reactor containment and Acheson Hall. Waste from various drains in the containment and pump room travel into either the 250-gallon or the 600-gallon tanks (see diagram). When the smaller tanks fill, they are pumped over into the 10,000-gallon tank. When the 10,000-gallon tank is nearly filled, it is valved off, recirculated, and a representative sample is drawn for analysis. This sample is analyzed using gamma spectroscopy against a N.B.S. traceable Marinelli standard which was purchased from New England Nuclear. A gross beta sample is also dried down and counted. A further 100cc is distilled down for liquid scintillation analysis. Identified and unidentified radioactivities are then compared to MPCs as established in 10 CFR 20 and the allowable volume of release is calculated, using a dilution factor of 100,000 gallons per day. This is a conservative estimate of the water flow from the University into the Winspear Avenue trunk line. Tell-tale (ground water) samples may be extracted from pipes inserted into the ground between the waste tanks. This sampling is performed twice per year.

17. Solid radwaste is typically collected, using recyclable containers which are distributed throughout the Facility. These recyclable containers are usually plastic or fiberboard drums, and are used to collect contaminated paper, glassware, disposable cloth, etc. These containers are transferred to a state license with the Radiation Protection Department, which is across the street in the Howe Research Building. The Radiation Protection Department hydraulically compacts the material in the recyclable containers into 55-gallon drums for disposal. Some materials are not compactable, and are loaded into 55-gallon drums directly at the NSTF. When full, these materials are also transferred to Howe Research Building. Occasionally it is necessary to utilize a concrete-lined 55-gallon drum for more active waste. These, again, are eventually transferred to Howe, but typically remain within the NSTF containment until it is nearly time to ship. Some activities at the NSTF entail handling of short-lived radioisotopes only. In such a situation, contaminated glassware, clothing, gloves, etc. are placed into a number of small, specially-marked radwaste containers. When these containers are filled, the material is transferred into a 55-gallon drum which is kept locked shut. When this drum is filled, it is transferred to a low background area, and the material within is checked for contamination. If no contamination is found, the material is disposed of as non-radioactive waste.

18. The Radiation Protection Department (RPD) organizational chart is shown in Figure 1. The Radiation Safety Officer (RSO) establishes policy in consultation with the Radiation Safety Committee (RSC) and implements procedures through the operations of the RPD. (A copy of the RSC Constitution and By-Laws are attached.) The RSO has the authority to immediately discontinue any operation deemed to constitute an undue radiological hazard. The RPD Manager has the responsibility to ensure that the radiation protection program is properly implemented on a day-by-day basis. The RPD Manager reports to the Campus Radiation Safety Officer.

The technical staff (in-house titles "Senior Radiation Safety Monitor" and "Radiation Safety Monitor") are responsible for the day-to-day operations of the RPD. These would include surveys of areas in which radioactive material or radiation-producing equipment are used, enforcing radiation protection regulations, maintaining required records, performing instrument calibrations and quality control of instruments, training of workers in radiation protection, etc. Technical staff have the authority to issue written citations to users who violate regulations. One staff member is assigned exclusively to the NSTF, while three staff members are assigned to the Campus. Technical staff report to the RPD Manager.

19. The RPD's radiation protection program is designed to protect the health and safety of the members of the University community and the public from the potentially harmful effects of radiation, through maintaining both external and internal exposures as low as reasonably achievable. This policy is endorsed by the University Radiation Safety Committee and by the President of the University through the Vice President for Research and Graduate Education. To ensure that radiation exposures are maintained as low as reasonably achievable, the RPD performs periodic inspections of all use areas of radioactive material and radiation-producing equipment for compliance with applicable licenses and regulations.

20. The fixed position radiation and effluent monitors may be summarized as follows:

Monitor	Detector	Efficiency	Range
Primary Water	GM tube	$\sim$ 25%	10 - 10^5CPM
Building Air	2 GM tubes	1.22×10^{-9} μc/cc/CPM	10 - 10^5CPM
Stack Gas	2 GM tubes	1.68×10^{-9} μc/cc/CPM	10 - 10^5CPM
Stack Particulate	End window GM	5×10^{-13} μc/cc/CPM	10 - 10 CPM
Area 1, 2, 3	Scintillator	Not applicable	.1 to 100mr/hour
Bridge	Scintillator	Not applicable	.1 to 100mr/hour
Hot Cell	Scintillator	Not applicable	10 to 10kmr/hour

21. Methods and frequency of instrument calibratins and operational checks are described by Operating Procedure No. 26. I have enclosed a copy of this OP.

22. A number of different dosimetry devices are used in the NSTF personnel monitoring program. Our primary dosimeters consist of commercially available film badges and TLD extremity rings. (Currently provided by Radiation Detection Corporation.) The film badges include normal and extended range films and neutron detection is performed using NDS film. We are investigating the necessity and effectiveness of utilizing track-etch and other types of neutron dosimetry. As a backup, staff members and visitors are issued Victoreen pocket chambers, which are charged and read on an electrometer. A limited number of self-reading pocket chambers are available for special situations such as hot transfers. The Radiation Protection Department also has a Victoreen Model 2810 TLD reader and a supply of lithium fluoride whole body and extremity dosimeters.

Film badges and rings for most staff members are changed on a monthly basis. However, radioisotope processors are issued dosimeters on a weekly basis. Pocket chambers are leak-tested and calibrated on a twice-per-year basis. All dosimetry reports are sent directly to the Radiation Protection Department, where they are reviewed, and a copy is then forwarded for posting to the NSTF.

23. Annual personnel exposure summaries for the NSTF for the last five years are attached.

24. Summaries for quantities of liquid and gaseous waste discharges for the last five years are attached. Solid waste from the NSTF is processed, together with the solid waste from the University campus. Separate records are not maintained. The attached "Summary of Radioactive Waste Disposal" table shows the campus waste

shipments for the past five fiscal years. The NSTF solid waste contribution would be about ten per cent of the indicated values.

25. Attached are the definitions in the Technical Specifications, in alphabetical order.

Table 3.1

Required Instrumentation

Instrument Channel		Min. No. Operating		Function	Set Point	Modes in which Required
Log Count Rate	+ *	1		Indication/Inhibit	< 2 cps;>9800 CPS	Start-Up
Linear Pwr.	+ *	1	a	Indication	-	All
Log Pwr	+ *	1		Indication	-	All
Period	+ *	1		Indication		All
Pwr. Safety	+ *	2		Indication/Scram	120%	All
Pwr. Safety	+ *	1		Reverse	110%	All
Manual Scram	+ *	5		Scram	-	All
D.C. Door Open	+ *	1		Scram	DR < Full Closed	All
Flow	+ *	1		Indication/Scram	68 lps	Forced Conv.
Flapper Open	+ *	1		Scram	<250 kw	Forced Conv.
Water Level Low	*	1		Scram	6.13m Over Fuel	All
" " "	*	1		Annunciation	6.43m " "	All
" " High	*	1		Annunciation	6.74m " "	All
Pool Temp.	+	1		Scram	52°C	Forced Conv.
Core Out. Temp.	+	1		Annunciation	52 °C + ΔT	Forced Conv.
Recorders Inoperative	*	3		Inhibit	--	Start-Up
Conductivity	+	0		Annunciation	200 K Ohms	None
EPF Valve Open	+	0		Annunciation	Valve Open	None
Demin. Temp.	+	0		Annunciation	43°C	None
Suc. Valve Closed	+	1		Disables Pri. Pump	V< Full Open	Forced Conv.
Servo Deviation	+	1		Annunciation/XFer to Manual	± 10%	Servo Control
Blade Pos. - Analog	+	1 of 2		Indication	-	All
Blade Pos. - Digital	*	1 of 2		Indication	-	All
Nitrogen - 16	+	2 of 3		Indication	-	Forced Conv.
Primary Temps	+	2 of 3		Indication	-	All
Core Delta T	+	2 of 3		Indication	-	Forced Conv.

a - Linear power channel and any recorder may be inoperable for short periods while operating.

* - Operability check required prior to operation.

+ - Test and/or calibration required four times/year.

V K Typical Core Loading and Control Rod Effects

The typical core loadings generally contain between 22 and 27 fuel assemblies depending on burn-up and experimental needs. Typical measured rod worths are as follows -

Rod # 1	1.4% delta K/K
# 2	1.5
# 3	3.0
# 4	3.4
# 5	1.5
Pulse	1.8
Sum of 1 to 6	12.6

Typical shutdown margins are in the order of 7 or 8% delta K/K. Typical excess reactivity needs are as follows -

Xenon overide	1.70% delta K/K
Power defect (0-2Mw)	0.35
Burn-up	0.45
Experiments	1.50-3.00
TOTAL	4.00-5.50

VI Personnel and Organization

Omit the position of general manager and change the Nuclear Hazards Committee to Nuclear Safety Committee.

VIII B 1 Loss of Ventilation

Due to the loss of two beam tubes, the release of all Argon-41 to the reactor room would result in a concentration of 4.6×10^{-10} Ci/cc instead of 5.7×10^{10} Ci/cc.

Ar-41 from the dry chamber is not considered. While production of Ar-41 in the facility is calculated to be 1.25mCi/min when in use, the facility is well ventilated and so no significant inventory would be expected to accumulate. (Ref. Letter to NRC dated 4/15/64)

VIII B-4 Loss of Pool Water

The gross loss of all pool water should be considered the worst credible accident. As previously stated, this would not result in loss of fuel integrity, but would result in extremely high radiation levels within the building. The hazard would be short term because the core could be recovered by flooding the lower level of the containment vessel if necessary.

VIII B-5 Maximum Start-up Accident

A re-evaluation of this accident is enclosed in the appendix C. The original analysis was based on erroneous rod speeds and rod worths. Also, the assumption that the reactivity is inserted as a step does not result in a worst case analysis.

	1	2	3	4	5	6
A	FC	28	32	10	13	
B		24	42	22R	5	
C	1	9R	27R		25R	6
D	7	38	11R	8R	18R	2
E	15	17R		4R	37	12R
F		40	3R	23	16	

PULSTAR CORE

LOADING NO. 125

CRITICAL ROD POS.

\# 1 66.70

\# 2 66.70

\# 3 66.70

\# 4 66.70

\# 5 66.70

DATE April 13, 1982

	1	2	3	4	5	6
A		19	29	18	4	
B	10	17		3	12	
C	9	7	34	21	22	
D	26	14		20	36	
E	33	11	30	1	13	
F						

PULSTAR CORE

LOADING NO. 50

CRITICAL ROD POS.

\# 1 58.26

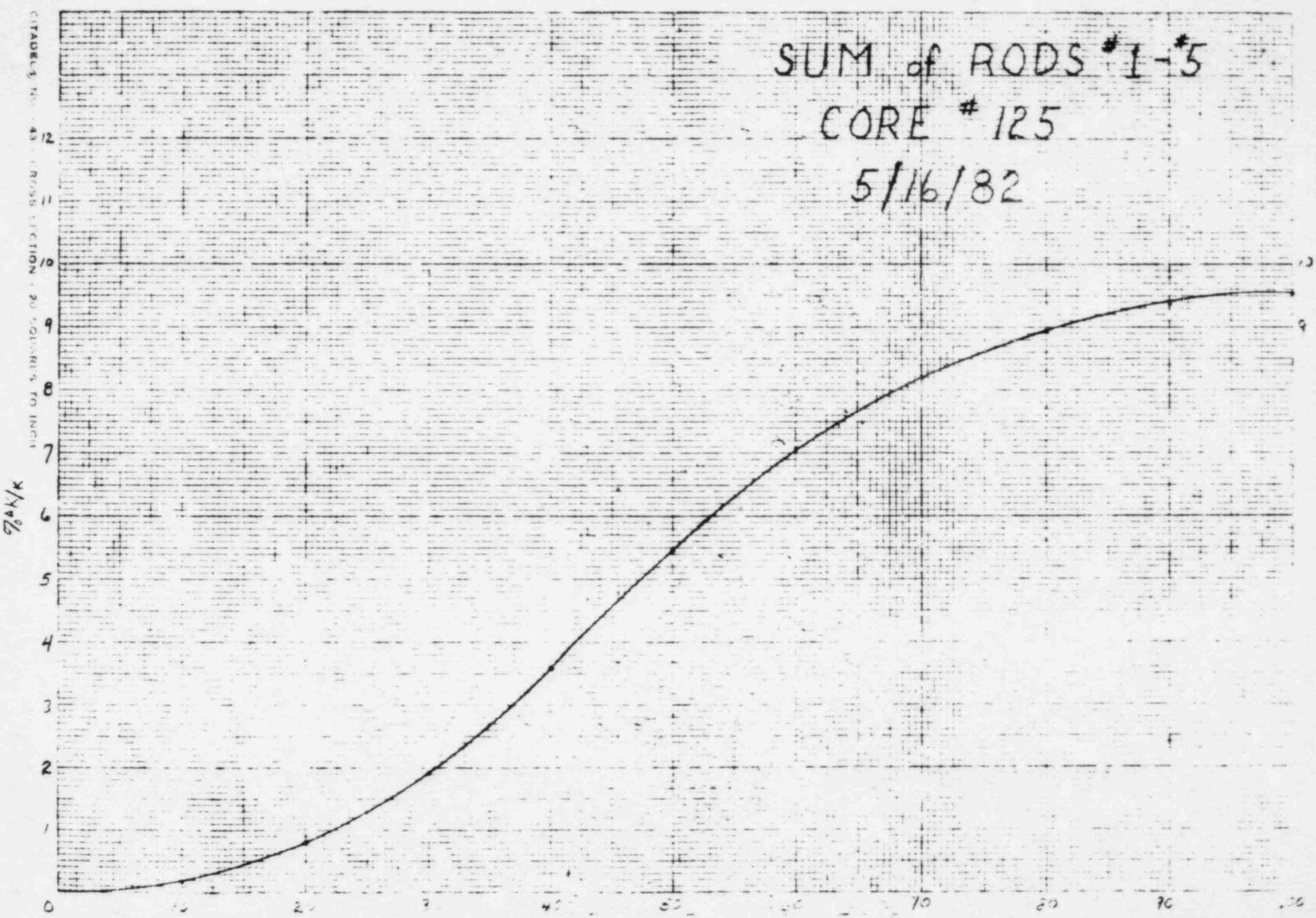
\# 2 58.26

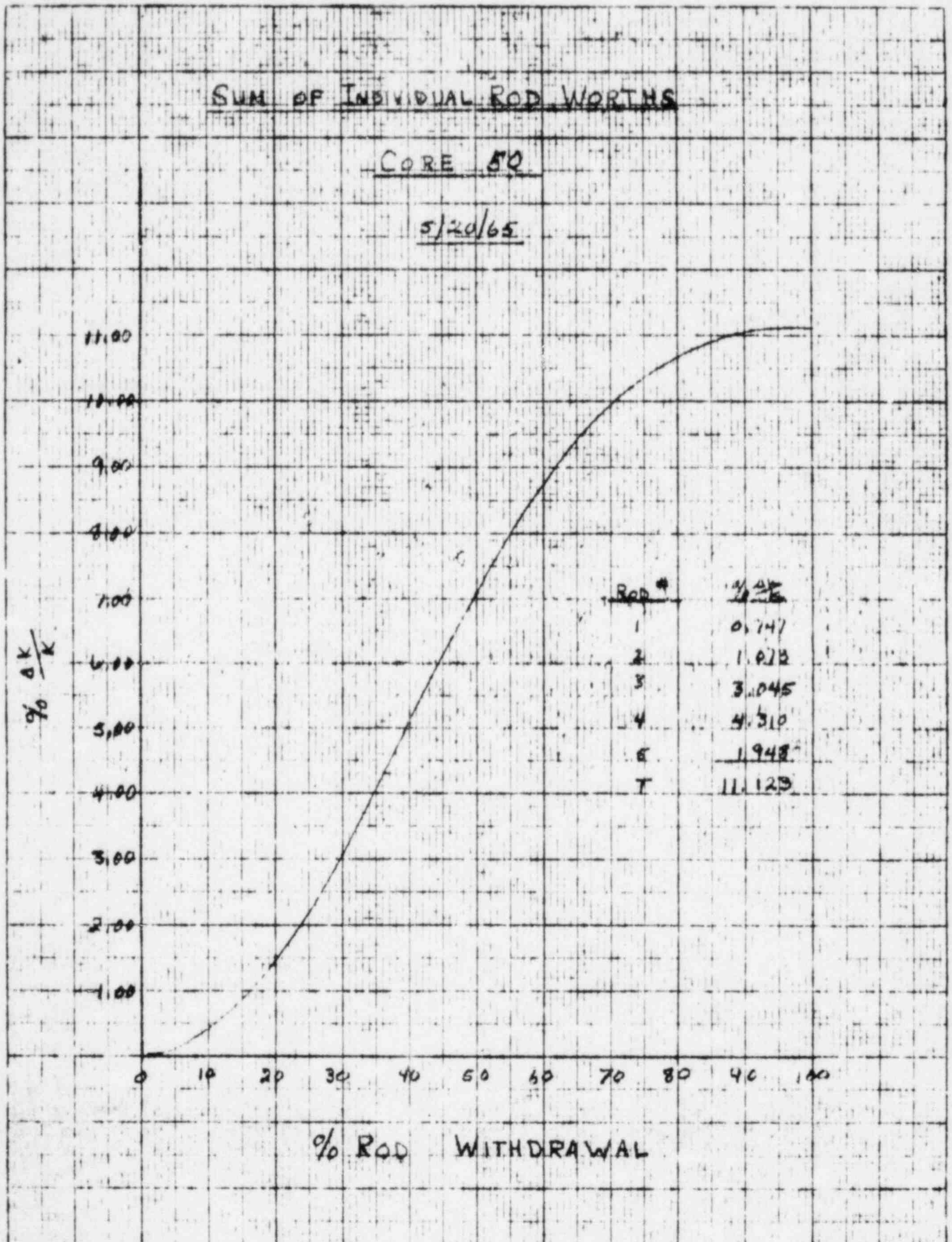
\# 3 58.26

\# 4 58.26

\# 5 58.26

DATE May 20, 1965


SUM of RODS #1 - #5
CORE # 125
5/16/82
%Δk/k
% Withdrawal
0
1
2
3
4
5
6
7
8
9
10
11
12
CITADEL NO. 43 - CROSS SECTION - 20 SQUARES TO INCH


SUM OF INDIVIDUAL ROD WORTHS
CORE 5Q
5/20/65
% Δk/k
11.00
10.00
9.00
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0
10
20
30
40
50
60
70
80
90
100
% ROD WITHDRAWAL
Rod #
% Δk/k
1
0.747
2
1.078
3
3.045
4
4.310
6
1.948
T
11.128

	1	2	3	4	5	6
A	1	26	17	15	31	grid plug
B	21	9	22		13	G8
C	36	4	7	28	8	11
D	20	3		27	25	G7
E	12	14	34	32	33	G6
F	G9	G56	G3	G4	G5	F-C

PULSTAR CORE

LOADING NO. 70 G 8 A

CRITICAL ROD POS.

\# 1 74.50

\# 2 74.50

\# 3 74.50

\# 4 74.50

\# 5 74.50

DATE July 22, 1971

G = graphite reflector

	1	2	3	4	5	6
A						
B						
C						
D						
E						
F						

PULSTAR CORE

LOADING NO. ______

CRITICAL ROD POS.

\# 1 ______

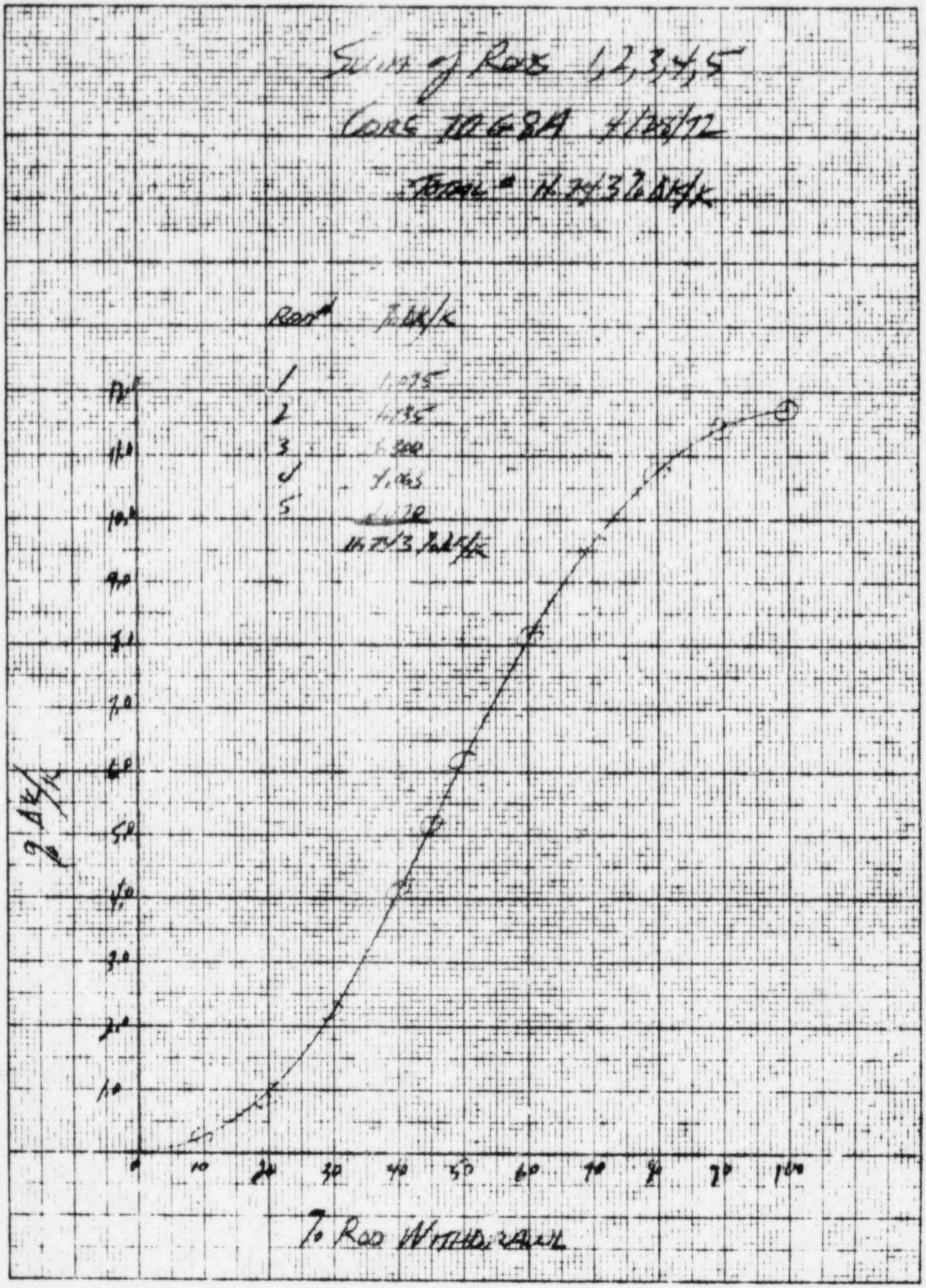
\# 2 ______

\# 3 ______

\# 4 ______

\# 5 ______

DATE ______


Sum of Rods 1,2,3,4,5
Core 10-68A 4/28/72
Total = 14.743 % Δk/k
Rod # % Δk/k
1 1.075
2 4.135
3 6.500
4 7.065
5 2.320
14.743 % Δk/k
% Δk/k
0
10
20
30
40
50
60
70
80
90
100
% Rod Withdrawal

OP #31 1/82

Figure 2

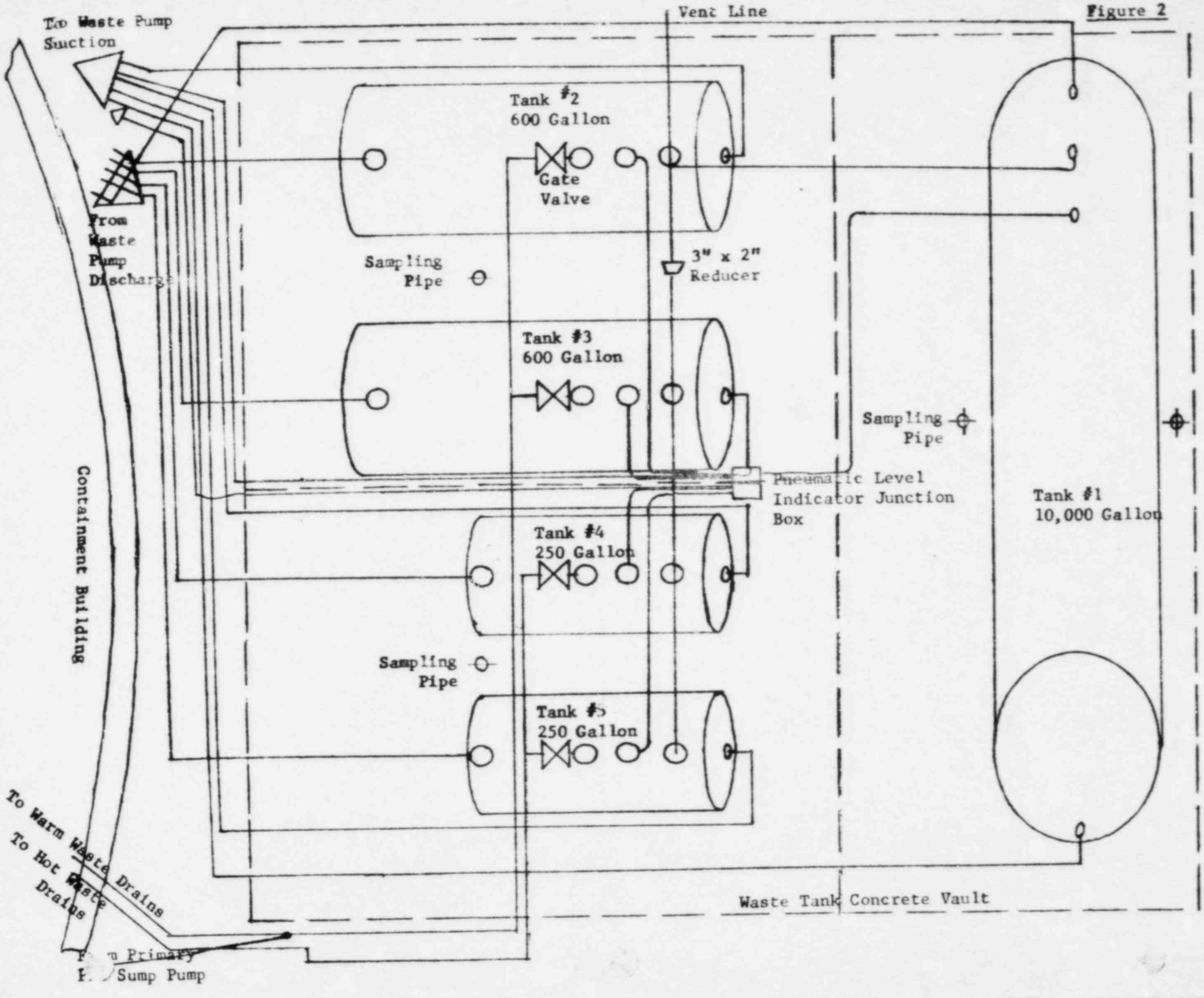
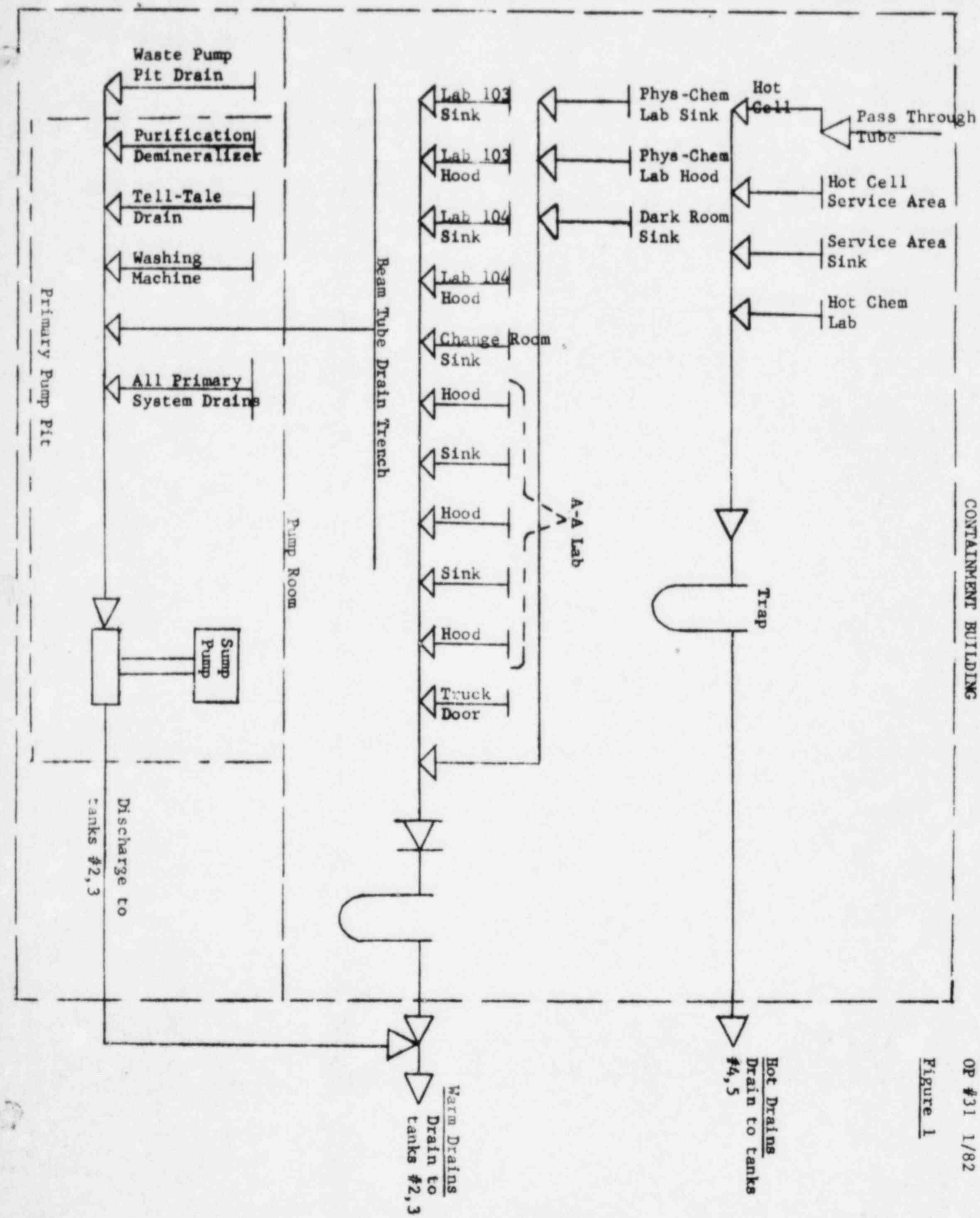


Figure 1


CONTAINMENT BUILDING
Primary Pump Pit
Pump Room
Beam Tube Drain Trench
Waste Pump Pit Drain
Purification Demineralizer
Tell-Tale Drain
Washing Machine
All Primary System Drains
Sump Pump
Discharge to tanks #2,3
Lab 103 Sink
Lab 103 Hood
Lab 104 Sink
Lab 104 Hood
Change Room Sink
Hood
Sink
Hood
Sink
Hood
A-A Lab
Truck Door
Phys-Chem Lab Sink
Phys-Chem Lab Hood
Dark Room Sink
Hot Cell
Pass Through Tube
Hot Cell Service Area
Service Area Sink
Hot Chem Lab
Trap
Warm Drains
Drain to tanks #2,3
Hot Drains
Drain to tanks #4,5

ORGANIZATIONAL CHART

SUNY/BUFFALO RADIATION PROTECTION DEPARTMENT

PRESIDENT
STEVEN SAMPLE

VP FOR RESEARCH AND GRADUATE EDUCATION
DR. DONALD RENNIE

RADIATION SAFETY OFFICER
DR. ALAN K. BRUCE
ASSOCIATE PROFESSOR, BIOLOGY

RADIATION PROTECTION MANAGER
PR-3 1123/34035
MARK A. PIERRO

CAMPUS RADIATION SAFETY

TECHNICAL SPECIALIST
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Monitor:PR-2 1981/34036
KATHLEEN E. OWENS

TECHNICAL SPECIALIST
Senior Radiation Safety
Monitor: PR-2 1981/32399
DONALD J. SHERMAN

TECHNICAL ASSISTANT
Radiation Safety Monitor
PR-1 1980/34037
NANCY A. HUTCHISON

NSTF SAFETY

TECHNICAL ASSISTANT
Radiation Safety
Monitor: PR-1 1980/34030
JAMES P. GRIFFIN

CLERICAL

RADIATION SAFETY RECORDS CLERK
SG-5 1126/34034
M. KAREN THOMAS

TEMPORARY SERVICE

State University of New York at Buffalo

ADMINISTRATIVE CHAIN OF COMMAND AND RESPONSIBILITY FOR RADIATION SAFETY

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                +---------------------+
                |                     |
                |      President      |
                |                     |
                +---------------------+
                           |
                +---------------------+
                |   Vice-President    |
                |  for Research and   |
                | Graduate Education  |
                +---------------------+
                           |
                           |
+-------------------+      |      +--------------------------------------------+
|                   |------+------|                                            |
| Radiation Safety  |             |         Radiation Safety Committee         |
|      Officer      |             |                    RSC                     |
|                   |             |--------------------------------------------|
|       RSO         |             | Radioisotope |Nuclear Safety|    X-Ray     |
|                   |             |  Committee   |  Committee   |  Committee   |
|                   |             |     RIC      |     NSC      |     XRC      |
+-------------------+             +--------------------------------------------+
          |                              |              |              |
+-------------------+             +-----------------------------+      |
|       RPD         |             |     RPD      |     NSTF     |      |
|     Manager       |-------------|  Operating   |  Operating   |      |
|                   |             |  Committee   |  Committee   |      |
|-------------------|             +-----------------------------+      |
|Radiation Protection|                   |              |              |
|    Department     |                    |              |              |
|       RPD         |                    |              |              |
+-------------------+                    |              |              |
          |                              |              |              |
          +------------------------------+--------------+--------------+
                                         |              |              |
                                  +--------------------------------------------+
                                  |    Campus    |   Nuclear    |    Campus    |
                                  |              |   Science    |              |
                                  | Radioisotope |     and      |    X-Ray     |
                                  |              |  Technology  |              |
                                  | Laboratories |   Facility   |Installations |
                                  +--------------------------------------------+
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II. ADMINISTRATIVE ORGANIZATION OF THE RADIATION SAFETY PROGRAM

STATE UNIVERSITY OF NEW YORK AT BUFFALO

RADIATION SAFETY COMMITTEE

CONSTITUTION

December 14, 1979

ARTICLE I: NAME

The official name of the committee shall be the "Radiation Safety Committee" of the State University of New York at Buffalo, hereinafter referred to as "RSC".

ARTICLE II: PURPOSE

The purposes of the RSC are (1) to review and approve policies and procedures pursuant to the safe use of radioactive materials and radiation-producing equipment; (2) to review and approve unusual experiments including reactor modifications and the overall safety of operation of the reactor; (3) to assure compliance of campus activities with applicable state and federal regulations; and (4) to carry out other duties as may be prescribed by licenses, specifications and manuals.

ARTICLE III: RADIATION SAFETY OFFICER

The campus Radiation Safety Officer (RSO) is an individual who is qualified by training and experience in radiological health to establish and administer a radiation protection program under authorization by the President of SUNY at Buffalo and who shall serve as Director of the Radiation Protection Department. The RSO shall establish policy in consultation with the Radiation Safety Committee (RSC) and implement procedures through operations of the Radiation Protection Department. The campus RSO shall have the authority to immediately discontinue any operation deemed to constitute an undue radiological hazard.

ARTICLE IV: MEMBERSHIP AND OFFICERS

The Radiation Safety Committee shall consist of the full membership of its subcommittees, the Nuclear Safety Committee and the Radioisotope Committee and such other subcommittees as provided in the By-Laws. The Chairpersons and Secretaries of each subcommittee shall serve in rotation as Chairperson and Secretary of the RSC.

ARTICLE V: MEETINGS

a. The RSC shall meet at least annually to review the activities of the subcommittees and to take action on matters of mutual concern.

b. Meetings may be called by the Committee Chairperson, the campus RSO or at the request of any committee member.

c. Minutes of all meetings shall be distributed to all members.

ARTICLE VI: QUORUM

A quorum of any meeting of the RSC shall consist of at least five (5) members including the Chairperson or his designee, the campus RSO or his designee and another member whose specialized knowledge is appropriate to the agenda of the meeting. A quorum for consideration of matters affecting the Nuclear Science and Technology Facility (NSTF) must have a majority of individuals present who are not members of the NSTF staff.

ARTICLE VII: BY-LAWS

a. The By-Laws of the RSC shall include the rules of procedure.

b. Proposed amendments to the By-Laws may be made by any committee member. Proposed amendments must be discussed at a meeting of the committee prior to submission to the committee for approval. Proposed amendments must be approved by two-thirds of the members attending.

STATE UNIVERSITY OF NEW YORK AT BUFFALO

RADIATION SAFETY COMMITTEE

BY-LAWS

Adopted December 14, 1979
* Amended January 7, 1981

1. The Committee Chairperson shall notify the members of the agenda not later than 48 hours before a regular meeting. For emergency meetings, the agenda shall be included in the meeting notice.

2. Questions before the Committee shall be decided by vote of the members present, the concurrence of a majority of those present being required for approval except that (1) no question requiring specialized knowledge shall be decided unless a member or consultant who is qualified by training and experience in that field is present and (2) no member who is a Principal Investigator on a specific project shall be entitled to vote on Committee decisions relating to that project.

3. Principal Investigators and/or consultants shall be invited to appear before the Committee to discuss proposed procedures as required.

4. The Nuclear Safety Committee (NSC) shall consist of a minimum of six (6) persons having nuclear experience including the campus RSO, the RPD Manager, the NSTF Director, and the NSTF Operations Manager who shall be permanent members of the NSC. Other members shall represent a cross-section of campus users of the NSTF or possess expertise appropriate to the purpose of the NSC. These members shall be appointed by the Director of the NSTF with the approval of the campus RSO and with the advice and consent of the NSC for a term of three (3) years. The NSC shall elect from its appointive membership a Chairperson and a Secretary.

The NSC is a sub-committee of the RSC and is authorized to act for the RSC in all routine decisions and in those policy decisions unlikely to have a major impact on the campus. The NSC has jurisdiction over all matters pertaining to licensing and regulation of the reactor and the NSTF. NSTF Operating Procedure 41 dealing with procedures for reactor experiments and experimental limitations, and Radiation Protection Manual, Volume II shall serve as guidelines for NSC review and approval of submitted experiments and shall, as such, be a part of these By-Laws. Suggested modifications of OP 41 and Volume II shall be subject to review and approval.

A quorum of the NSC must have at least five (5) members present and must have a majority of non-NSTF members present. The NSC shall meet at least twice a year at approximately six (6) month intervals. Items 1-3 inclusive of the RSC By-Laws are applicable to NSC meetings.

5. The NSTF Operating Committee shall consist of the NSTF Director, the NSTF Operations Manager, and the RPD Manager. Ad hoc Operating Committee members may be designated by the NSTF Director. The NSTF Operating Committee is a sub-committee of the NSC and is authorized to act for the NSC with regard to routine aspects of the NSTF operation.

6. The Radioisotope Committee (RIC) shall consist of a minimum of eight (8) persons having radioisotope experience including the campus RSO,

the RPD Manager, and a representative of the Joint Radioisotope Committee of the Affiliated Hospitals who shall be permanent members of the RIC. Other members shall represent a cross-section of campus users of radioisotopes or possess a source of expertise appropriate to the purpose of the RIC. These members shall be appointed by the campus RSO upon advice and consent of the RIC for a term of three (3) years. The RIC shall elect from its appointive membership a Chairperson and Secretary. The RIC is a sub-committee of the RSC and is authorized to act for the RSC in all routine decisions and in those policy decisions unlikely to have a major impact on the campus. The RIC has jurisdiction over all matters pertaining to New York State Radioactive Materials License # 1049. The RPD Radiation Protection Manual, Volume I, dealing with the use and disposal of radioisotopes shall serve as a guideline for review and approval of submitted experiments. As such, a copy of the Radiation Protection Manual, Volume I, shall be a part of these By-Laws. Suggested modifications shall be subject to review and approval.

A quorum of the RIC must have at least five (5) members present and must have a majority of non-RPD members present. Items 1-3 inclusive of the RSC By-Laws are applicable to RIC meetings.

7. The RPD Operating Committee shall consist of the campus RSO, the RPD Manager, the Chairperson, and the Secretary of the RIC. Ad hoc Operating Committee members may be designated by the campus RSO. The RPD Operating Committee is a sub committee of the RIC and is authorized to act for the RIC with regard to routine aspects of RPD operations including user approval for routine applications.

*8. The X-Ray Committee (XRC) shall consist of a minimum of five (5) persons having experience with radiation-producing equipment including the campus RSO and the RPD Manager who shall be permanent members of the XRC. Other members shall represent a cross-section of campus users of radiation-producing equipment or possess a source of expertise appropriate to the purpose of the XRC. These members shall be appointed by the campus RSO upon the advice and consent of the XRC for a term of three (3) years. The XRC shall elect from its appointive membership a Chairperson and Secretary. The XRC is a sub-committee of the RSC and is authorized to act for the RSC in all routine decisions and in those policy decisions unlikely to have a major impact on the campus. The XRC has jurisdiction over all matters pertaining to the New York State Department of Health's Certificate of Registration. The RPD Radiation Protection Manual, Volume V, dealing with the use of radiation-producing equipment shall serve as a guidline for review and operation of such equipment. As such, a copy of the Radiation Protection Manual, Volume V, shall be a part of these By-Laws. Suggested modifications shall be subject to review and approval.

A quorum of the XRC must have at least three (3) members present. Items 1-3 inclusive of the RSC By-Laws are applicable to XRC meetings.

OP #26 4/82

PULSTAR REACTOR (R-77)

OPERATING PROCEDURE

Effluent and Area Monitor Systems

I. Introduction

The remote monitoring system collects information regarding radiation intensity in various working areas of the containment building, as well as detecting and measuring the radioactive content of air and water effluents. All area monitors provide indication, measurement, and alarm at the monitoring location, and also remotely present the same information on the auxiliary rack in the control room. All effluent monitors provide indication and visual alarm (flashing red light) at the monitoring location. They provide indication, and both audible and visual alarm on the control room auxiliary rack. Strip chart recorders record all monitor readings. Monthly operability checks and quarterly calibrations are performed by the Operations Department personnel. The yearly calibrations are performed by the Radiation Protection Department personnel. Records for all checks and calibrations are forwarded for review and maintained by the Radiation Protection Department. Copies of the records are also maintained by the Operations Department.

II. Monitor Functions

A. Radiation Monitors

The radiation monitors measure either the general radiation level in an area and display it in mR/Hr or measure the activity at some specific location and display it in counts/minute. All area monitors have both yellow and red alarm lights. In all cases, the red alarm means that some pre-determined level has been exceeded. For all monitors, the yellow alarm in the control room indicates an instrument failure. Locally, the effluent monitors have a normally green light to indicate proper operation. An instrument failure is indicated when the local green light goes out, or the remote yellow light comes on. Alarms are accompanied by a bell or other audible signal in the control room. Area monitors have a local bell and red light for high alarm. Effluent monitors have a red light only. Signals can be silenced from the control room. The monitor characteristics are shown in Table I.

B. Monitor Geometries

1. The neutron deck area monitors have their detectors located adjacent to the indicating unit.

2. The hot cell monitor has its detector located in the cell on the wall above the window. The ratemeter is in the hot cell service area.

3. The bridge monitor detector is clamped to the bridge structure just over the pool. The ratemeter is on the side of the bridge.

4. The primary water detector is located in a shield strapped to the primary pipe just upstream of the primary pump. The ratemeter is located just outside the pump room door.

5. A side stream of air is extracted from the stack exhaust duct in the basement. The stream first enters the fixed filter stack particulate monitor. From there it goes into the stack gas monitor and then is returned to the duct. Two ratemeters are in the same area.

6. A side stream of air is drawn from the 36" building duct on the control deck. The stream first passes through the building air monitor and then through a fixed filter.

C. Additional Monitors

1. A scintillation area monitor is provided for the pump room. The detector is in the pump room and the ratemeter is just outside the door. This monitor does not have a readout in the control room.

III. Action By Reactor Operator In Alarm Situations

A. Monitors 1, 2, 3, or 5 (Area Monitors)

If the cause cannot be immediately determined, notify the shift supervisor and Radiation Protection Department for investigation. Announce condition over intercom to warn personnel in the area.

B. Monitor 6 (Bridge) and 11 (Building Air)

Monitor 6 or 11.

LOW ALARM (Instrument Failure) - when the cause cannot be immediately determined, notify shift supervisor and Radiation Protection Department for investigation.

HIGH ALARM - when the cause cannot be immediately determined, notify shift supervisor and the Radiation Protection Department. Increase the red alarm set point to an interim high alarm set point equal to twice the original red alarm set point.

If the interim alarm set point is reached, shut down the reactor and evacuate the building according to procedure.

C. Both Monitors 6 and 11

If the cause cannot be immediately determined to be direct radiation, scram the reactor, actuate the evacuation alarm, evacuate the facility, and notify Radiation Protection Department. Verify that the Pratt dampers are closed; back up manually if necessary.

D. Monitor 7 (Primary Water)

LOW ALARM (Instrument Failure) - notify shift supervisor and Radiation Protection Department for investigation.

HIGH ALARM - notify shift supervisor and Radiation Protection Department. If the monitor is off scale, shutdown the reactor and secure the primary and secondary cooling systems.

E. Monitor 8 (Stack Gas)

LOW ALARM (Instrument Failure) - verify instrument failure according to the following procedure:

1. Halt all experiments or operations that could lead to higher than normal releases to the stack.
2. Check to see if the low alarm set point is set properly (see (section IV.B.6).
3. Observe count rate meter reading at the monitor - verify below low alarm set point.
4. Check to see if the flow rate is adjusted to 5 CFM.
5. Perform an external pulser check - verify correct response on count rate meter.
6. Using an external β-γ source, position the source so to check the operability of each CM detector (see section IV.B.3).
7. Notify shift supervisor and Radiation Protection Department of the results of the above tests.
8. Initiate repairs if necessary.
9. If repairs cannot be made, shut down the reactor until repairs are complete.

HIGH ALARM - notify shift supervisor and Radiation Protection Department.

F. Monitor 9 (Stack Particulate)

LOW ALARM (Instrument Failure) - verify instrument failure according to the following procedure:

1. Halt all experiments or operations that could lead to higher than normal releases to the stack.

2. Check to see if the low alarm set point is set properly (see section IV.B.6).

3. Observe count rate meter reading at the monitor - verify below low alarm set point.

4. Perform an external pulser check - verify correct response on count rate meter.

5. Using an external β-γ source, position the source so to check the operability of the GM detector (see section IV.B.2).

6. Check to see if GM tube is properly seated within the tube holder.

7. Check to see if the flow rate is adjusted to 5 CFM.

8. If the monitor is still in low alarm, notify shift supervisor and Radiation Protection Department.

9. Initiate repairs, if necessary.

10. If repairs cannot be made, shut down the reactor until repairs are complete.

HIGH ALARM - notify shift supervisor and Radiation Protection Department.

IV. Monthly Instrument Operability Check

Perform in first half of the month, preferably.

A. Radiation Monitors

Area Monitors 1, 2, and 3
Bridge Monitor
*Hot Cell Monitor
Pump Room Monitor

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*If conditions in the Hot Cell prohibit completing any of the required checks, indicate on the proper form the reason such checks were not performed. Arrange to have any checks that were not performed completed as soon as practicable after such conditions in the Hot Cell no longer exist.

1. Inform the reactor operator that a test is in progress of the instrument to be checked.

2. Use a gamma ray emitting source to check operation.

3. For each detector, position the source so that the radiation level is equal to or greater than the high alarm set point. Verify that an alarm condition exists at both the machine and in the control room. Verify that the alarm threshold is consistent with Table II, and record on the check sheet.

4. To check the instrument failure set point, raise the pointer above the meter indicator. Verify that an alarm condition exists at both the machine and in the control room.

5. When checking the bridge monitor high alarm, ensure that warning lights outside the control deck and gamma deck air locks turn on.

6. When checking the hot cell high level alarm, ensure that the hot cell door lock functions.

7. For each monitor, verify that the ratemeter agrees with the follow meter and chart recorder in the control room. Note - the pump room monitor does not have a follow meter in the control room.

Table II

MONITOR	LOW ALARM SET POINT	HIGH ALARM SET POINT
Neutron Deck #1, #2, #3	0.1 mR/Hr	5 mR/Hr
Hot Cell	10.0 mR/Hr	100 mR/Hr
Bridge	0.1 mR/Hr	30 mR/Hr
Pump Room	0.1 mR/Hr	30 mR/Hr

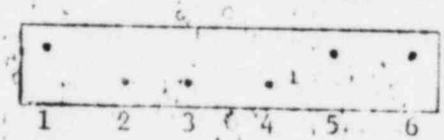
B. Effluent Monitors

Stack Particulate
Stack Gas
Building Air
Primary Water
Building Particulate

General

Since all the new effluent monitors (just like the old ones) have G.M. tube detectors, operability, i.e., the functioning of the system as a whole, can be verified by observing the meter in question. With the improved range of these instruments, i.e., 10 CPM vs. 50 CPM on the old ones, a properly operating channel should read something even with the reactor shut down, as during normal periods of operation. Long-term shutdowns, periods of weeks or months, might require a check source to verify operability. Any upscale deflection that can be shown to be a response to radiation constitutes verification of operability.

Alarm setpoints can not be observed with the instrument set up for normal operation. Switches must be manipulated for this purpose. Normal switch positions are as follows. A small screwdriver is necessary.



Switches 1 through 4 determine what is displayed on the meter. Switches 5 and 6 determine reset mode of Low and High level alarms. Up for automatic reset, down for manual.

NOTE: Only one of switches 1 through 4 should be "Up" at any one time.

Switch 1 up with 2, 3, and 4 down is normal, and provides a display of indicated count rate. Move this switch down to "Off" prior to closing one of the other switches.

Switch 2 displays the Low alarm set point. Switch 3 is not connected. Switch 4 displays the High alarm set point.

1. Inform the reactor operator that a test is in progress of the instrument to be checked.

2. For the stack particulate and the primary water monitors, position an appropriate source to check the operability of the detector and high alarm at both the machine and in the control room.

3. The stack gas and building air have two GM detectors connected in parallel within the sampling tank. Position an appropriate source to check the operability of each detector. Record the count rates for Left, Both, and Right Detectors on the check form.

4. Make sure the high alarm set points for each monitor are adjusted to the points specified on the card posted on the monitor.

5. To check low alarm on all effluent monitors, disconnect the detector cable at the back of the instrument. This causes the instrument to go down scale and trip the alarm.

6. To verify or change an alarm setpoint, proceed as follows:
   a) Move Switch 1 down.
   b) Move either Switch 2 or Switch 4 up, depending on which set point is in question. This displays the corresponding setpoint. To change either set point, rotate the corresponding alarm set point potentiometer while that alarm is displayed.
   c) Return to normal operation by pushing the alarm switch down and the ratemeter switch up. The alarms will trigger at whatever level is set in. Normal setpoint for Low on all channels is 10 CPM. The High alarms are set as indicated on each module on attached tape.

7. Local/Remote Meter Coordination and Recorder Calibration: proceed as above (6 a) and b) to coordinate meters; Remote meter calibration is accomplished by manipulation of the Low and High level alarm set point potentiometers.
   a) In Low mode, set pointer at 20 CPM. Verify with a person in the control room that this is indicated. If needed, adjust the mechanical zero on the remote meter and the electrical zero potentiometer in the three-pen recorder.

   b) In High mode, set pointer at 10^4 CPM. Control room meter and recorder should read the same. If not, set meter cal. with cal-pot (left side) to indicate same as local meter. This affects the recorder so if it is off, adjust the right hand pot to set the recorder. Recheck low end.

   NOTE: These adjustments are all interactive, so they must be rechecked until they read as close as possible. They should prove to be quite accurate. Return all switches to their NORMAL position.

8. Make sure the low alarm set points are adjusted to the proper levels indicated below:

MONITOR	LOW ALARM SET POINT
Building Air	10 CPM
Stack Gas	10 CPM
Stack Particulate	10 CPM
Primary Water	10 CPM
Building Particulate	10 CPM

9. For each monitor, verify that the ratemeter agrees with the follow meter and chart recorder in the control room.

10. To determine Stack Gas and Building Air background count rate, observe the three pen chart recorder tracings for a period of time in which the reactor was shutdown. Choose an average reading (in CPM) and post this number on the monitor chassis.

To determine the Stack Particulate background count rate, observe the three pen chart recorder tracings for a period of time several hours after the reactor has been shut down and a new filter installed on the filter housing. Choose an average reading (in CPM) and post number on the monitor's chassis.

Review and record each monitor's background monthly on the monitor's chassis.

V. Quarterly Instrument Calibration Procedures

Perform in first half of each quarter, preferably.

A. Radiation Monitors

Neutron Deck Monitors 1, 2, and 3.
Bridge Monitor
*Hot Cell Monitor
Pump Room Monitor

1. Inform the reactor operator that a test is in progress of the instrument to be checked.

2. Use a gamma ray emitting source for calibration.

*If conditions in the Hot Cell prohibit completing any of the required calibrations, indicate on the proper form the reason such calibrations were not performed. Arrange to have any calibrations that were not performed completed as soon as practicable after such conditions in the Hot Cell no longer exist.

3. For each detector, position the source such that the radiation level at the detector is equal to the low calibration point. (5 mr/hr for all monitors except Hot Cell, which is 10 mr/hr). This may be accomplished either by placing a calibrated ion chamber type survey meter as close to the detector as possible, or by use of a calibrated source at a measured distance. Adjust the low calibration potentiometer as required. Note - if the background is greater than 5 mr/hr, calibrate at that level.

4. Next, position the source so that the radiation level is increased to 50 mr/hr (100 mr/hr for the Hot Cell). If necessary, adjust the high calibration potentiometer until the reading is ±20% of the actual exposure rate.

5. Repeat steps #3 and #4 until proper indication is reached at each level without further adjustments to the instrument's calibration potentiometer.

6. For each monitor, verify that the ratemeter agrees with the follow meter and chart recorder in the control room.

7. Record the final readings on the calibration form.

B. Effluent Monitors

Stack Particulate
Stack Gas
Building Air
Primary Water
Building Particulate

1. Inform control room that a test is in progress of the instrument to be checked.

2. For each monitor, remove module to Electronic Shop. Calibrate according to procedure in manufacturer's manual (use a scaler and G.M. tube). This procedure compensates for dead time losses at high count rates.

3. For each detector, using a calibrated source in a fixed geometry, compare the count rate of the detector tube with its initial count rate. Replace the detector tube when the count rate decreases by 20 per cent.

VI. Yearly Calibration Procedure

Procedures for each of the required steps listed below will be kept on file in the Radiation Protection Department office.

A. Primary Water Monitor #7

1. Verify that the G.M. tube is operating on a voltage plateau.

B. Stack Gas Monitor #8, Building Air Monitor #11

1. Verify that the G.M. tube is operating on a voltage plateau.

2. Determine detector system sensitivity by injecting a known quantity of radioactive gas according to established procedures.

C. Stack Particulate Monitor #9

1. Verify that the G.M. tube is operating on a voltage plateau.

2. Generate an efficiency vs. energy curve for beta emitters, and establish the detector system sensitivity.

D. Provide sensitivity data to the Operations Department, and post alarm set points on monitors and remote follow meter panel.

VII. Checks Following Repairs

When repairs to monitor systems involve replacements that may affect calibration, recalibration is required. Notify Radiation Protection Department (on backshift, leave message in RPD mailbox) whenever such replacement is made, to determine the appropriate procedures to be followed.

VIII. Record Keeping

A. Maintain records of checks and calibrations as specified in section I.

B. Record all abnormal events or occurrences on the appropriate strip chart recorder. This would include:

1. Releases above normal operating backgrounds.

2. Repairs, maintenance, checks, or calibrations.

3. Instrument failure or malfunction.

4. Causes, if known, for abnormally high levels.

TABLE 1A AREA MONITORS

MONITOR	LOCATION	SENSOR	RANGE	PURPOSE	ALARM			FUNCTION
					LOW	HIGH	BASIS	
AREA 1,2,3	NEUTRON DECK	SCINTILLATOR	0.1-100 mr/hr	GENERAL AREA LEVELS	INSTRUMENT FAILURE	5mr/hr	ARBITRARY	NONE
BRIDGE	REACTOR BRIDGE	SCINTILLATOR	0.1-100 mr/hr	LEVEL OVER POOL	INSTRUMENT FAILURE	30mr/hr	ARBITRARY	CLOSE AIR DAMPERS WITH BLDG. AIR COINCIDENCE
HOT CELL	HOT CELL	SCINTILLATOR	10-10K mr/hr	LEVEL IN HOT CELL	INSTRUMENT FAILURE	100mr/hr	HIGH RADIATION AREA	PROHIBIT CELL ENTRY > 100mr/hr
PUMP ROOM (NOT REQUIRED)	PUMP ROOM	SCINTILLATOR	0.1-100 mr/hr	LEVEL IN PUMP ROOM	INSTRUMENT FAILURE	20mr/hr	ARBITRARY	NONE

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TABLE 1B EFFLUENT MONITORS

MONITOR	LOCATION	SENSOR	RANGE	PURPOSE	ALARMS			FUNCTION
					LOW	HIGH	BASIS	
PRIMARY WATER	PUMP ROOM	GM TUBE	$10-10^5$ CPM	PRIMARY WATER ACTIVITY	INSTRUMENT FAILURE	AS POSTED BASIS / SENSITIVITY	DEPARTURE FROM NORMAL	NONE
BUILDING AIR	FAN ROOM	2 GM TUBES	$10-10^5$ CPM	36" EXHAUST DUCT GAS ACTIVITY	INSTRUMENT FAILURE	AS POSTED BASIS / SENSITIVITY	$2x10^{-6}$uCi/cc RESTRICTED ARGON 41 MPC	WITH BRIDGE TRIPS DAMPERS
BUILDING PARTICULATE	FAN ROOM	GM TUBE	$10-10^5$ CPM	36" EXHAUST DUCT PARTICULATE ACTIVITY	INSTRUMENT FAILURE	AS POSTED BASIS / SENSITIVITY		NONE
STACK GAS	MECHANICAL EQUIPMENT ROOM	2 GM TUBES	$10-10^5$ CPM	UNDERGROUND EXHAUST DUCT ACTIVITY GASEOUS	INSTRUMENT FAILURE	AS POSTED BASIS / SENSITIVITY	$1x10^{-6}$uCi/cc ARGON 41 EQUIVALENT	NONE
STACK PARTICULATE	MECHANICAL EQUIPMENT ROOM	GM TUBE	$10-10^5$ CPM	UNDERGROUND EXHAUST DUCT ACTIVITY PARTICULATE	INSTRUMENT FAILURE	AS POSTED BASIS / SENSITIVITY	RESTRICTED AREA UNIDENTIFIED GROSS β MPC ($3x10^{-9}$ uCi/cc)	NONE

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EFFLUENT MONITOR CALIBRATION CHECK

PERFORMED BY ____________________ QUARTER ____________

INSTRUMENTS USED ______________________________ DATE ____________

MONITOR	UNIT	COUNTS	SCALER CPM / INDICATED CPM							TAIL PULSER	REMARKS
BUILDING AIR											
BUILDING PART.											
PRIMARY WATER											
STACK GAS											
STACK PART.											

UNIT- Module number.
Counts- Actual reading at time calibration performed.
TAIL PULSER- Puts out 3600 CPM, indicated will be very close to this.
REMOTE METER AND RECORDER INDICATIONS WILL BE CHECKED AS PART OF THIS CALIBRATION.

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AREA MONITOR CALIBRATION CHECK

PERFORMED BY ________________ QUARTER ________

INSTRUMENTS USED ________________ DATE ________

SOURCE USED ________________

MONITOR	mr/hr actual	mr/hr METER / mr/hr INDICATED				REMARKS
NEUTRON 1						
NEUTRON 2						
NEUTRON 3						
BRIDGE						
PUMP ROOM						
HOT CELL						

mr/hr ACTUAL is the rate in the vicinity of the detector as read on the meter used for the calibration.
REMOTE METER AND RECORDER INDICATIONS WILL BE CHECKED AS PART OF THIS CALIBRATION.

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AREA AND EFFLUENT MONITOR OPERABILITY CHECK

Performed by________________________________ Date________________

Pulser used ______________________

Source used______________________

MONITOR	ALARMS HIGH YES	ALARMS HIGH NO	ALARMS LOW YES	ALARMS LOW NO	OPERATING PROPERLY YES	OPERATING PROPERLY NO	COMMENTS				
Neutron 1											
Neutron 2											
Neutron 3											
Bridge											
Hot Cell											
Pump Room											
							10k	L	B	R	
Stack Gas											
Stack Particulate											
Building Air											
Building Particulate											
Primary water											

CONTROL FUNCTION CHECK

MONITOR/FUNCTION		CONTROL FUNCTION YES	CONTROL FUNCTION NO	COMMENTS
Bridge	Control Deck Lights			
	Gamma Deck Lights			
Hot Cell Interlock				
Building Air Damper				

OPERATING PROCEDURE #26

Effluent and Area Monitor Systems

Louis J. Henry Jr. — 5/25/82

Administration — Date

Philip M. Orlosky — 4/26/82

Operations — Date

Mark A. Pierro — 4/30/82

Radiation Protection — Date

SUGGESTED DRAFT FORMAT FOR THE REPORTING OF RECORDED PERSONNEL WHOLE BODY EXPOSURES FOR CALENDAR YEAR 1978

Licensee Reporting (Name & Address)	NRC License No(s).
State University of New York at Buffalo Nuclear Science and Technology Facility Rotary Road Buffalo, New York 14214	SNM-273, R-77

☐ IF PERSONNEL MONITORING WAS NOT REQUIRED DURING THE YEAR, CHECK THIS BOX.

OTHERWISE, COMPLETE THE FOLLOWING TABLE:

Annual Whole Body Dose Ranges * (Rems)	Number of Individuals in Each Range
No Measurable Exposure	168
Measurable Exposure Less Than 0.100	128
0.100 -- 0.250	5
0.250 -- 0.500	6
0.500 -- 0.750	3
0.750 -- 1.000	3
1.000 -- 2.000	1
2.000 -- 3.000	1
3.000 -- 4.000	1
4.000 -- 5.000	
5.000 -- 6.000	
6.000 -- 7.000	
7.000 -- 8.000	
8.000 -- 9.000	
9.000 -- 10.000	
10.000 -- 11.000	
11.000 -- 12.000	
> 12.000	

Total number of individuals reported 318 (includes Bio classes)

The above information is submitted for the total number of individuals for whom personnel monitoring was (check one):

☐ required under 10 CFR 20.202(a) of 10 CFR 34.33(a) during the calendar year.

☒ provided during the calendar year.

*Individual values exactly equal to the values separating exposure ranges shall be reported in the higher range.

Report prepared by: Louis G Henry (716)831-2826

1979

TABLE 4

SUNYAB Personnel External Radiation Exposure Summary 1979

ANNUAL WHOLE BODY DOSE RANGE (Rems)	NUMBER OF INDIVIDUALS IN EACH RANGE		
	CAMPUS *	NSTF STAFF **	NSTF PROCESSING
No Measurable Exposure	303	65	0
Measurable Exposure Less Than 0.100	37	56	1
0.100 - 0.250	8	4	0
0.250 - 0.500	0	7	0
0.500 - 0.750	1	0	0
0.750 - 1.000	0	1	1
1.000 - 2.000	0	0	2
2.000 - 3.000	0	0	1
Greater than 3.000	0	0	0
TOTAL NO. OF INDIVIDUALS REPORTED	353	133	5

* Includes Students and X-ray User Accounts

** Includes Special Projects and Public Safety Accounts

TABLE 5

SUNYAB Personnel Internal Radiation Exposure Summary 1979

ANNUAL THYROID DOSE RANGE (Rems)	NUMBER OF INDIVIDUALS IN EACH RANGE
No Measurable Exposure	20
Measurable Exposure Less Than 0.100	15
0.100 - 0.250	2
0.250 - 0.500	7
0.500 - 0.750	5
0.750 - 1.000	0
1.000 - 2.000	0
2.000 - 3.000	1
Greater than 3.000	0
TOTAL NO. OF INDIVIDUALS REPORTED	50

TABLE 1 - A

NSTF Personnel External Radiation

Exposure Summary - Calendar Year 1980

Annual Whole Body Dose Range (Rems)	NUMBER OF INDIVIDUALS IN EACH RANGE					
	NSTF Staff	Isotope Production Staff	Special Projects	Public Safety Officers	Visitors	Tours
No Measurable Exposure	10		14	64	908	150
Measurable Exposure Less than 0.100	5		1	9		
0.100 - 0.250	8					
0.250 - 0.500	4	1				
0.500 - 0.750	1					
0.750 - 1.000	1	1				
1.000 - 2.000	1	2				
2.000 - 3.000						
Greater than 3.000						
Total number of Individuals Reported	30	4	15	73	908	150

NSTF Personnel External Radiation Exposure Summary
Calendar Year 1981

ANNUAL WHOLE BODY DOSE RANGE (REMS)	NUMBER OF INDIVIDUALS IN EACH RANGE					
	NSTF STAFF	ISOTOPE PRODUCTION STAFF	SPECIAL PROJ	PUBLIC SAFETY OFFIC.	VISITORS	TOURS
None measurable	5	0	5	53	827	721
Measurable <0.100	4	0	3	4	0	0
0.100 - 0.250	5	0	0	0	0	0
0.250 - 0.500	3	1	0	0	0	0
0.500 - 0.750	2	2	0	0	0	0
0.750 - 1.000	1	0	0	0	0	0
1.000 - 2.000	0	1	0	0	0	0
2.000 - 3.000	0	0	0	0	0	0
> 3.000	0	0	0	0	0	0
TOTAL No. Reported	20	4	8	57	827	721

NSTF Personnel External Radiation Exposure Summary
Calendar Year 1982

ANNUAL WHOLE BODY DOSE RANGE (REMS)	NUMBER OF INDIVIDUALS IN EACH RANGE			
	NSTF STAFF	ISOTOPE PRODUCTION STAFF	SPECIAL PROJ	PUBLIC SAFETY OFFIC.
None measurable	1	0	4	51
Measurable <0.100	2	0	7	3
0.100 - 0.250	3	0	1	1
0.250 - 0.500	7	1	0	0
0.500 - 0.750	3	C	0	0
0.750 - 1.000	1	0	0	0
1.000 - 2.000	0	3	0	0
2.000 - 3.000	0	0	0	0
> 3.000	0	0	0	0
TOTAL No. Reported	17	4	11	55

TABLE V

SUMMARY OF 1978 AIR RELEASES

NUCLIDE	TOTAL Ci RELEASED	MAX CONCENTRATION AT POINT OF RELEASE (uCi/ml)	AVG ANNUAL CONCENTRATION (uCi/ml)	% OF PERMISSIBLE LIMIT *
Routine Releases				
Powerhouse Stack Ar-41	2.09×10^{2}	6.2×10^{-5}	2.60×10^{-6}	5.5×10^{-1}
Building Stack Ar-41	8.85	1.1×10^{-6}	2.64×10^{-7}	2.3×10^{-2}
Building Stack Cs-138	6.18×10^{-2}	1.4×10^{-8}	7.60×10^{-10}	2.1×10^{-4}
Non-Routine Releases				
Ar-41	2.4	9.0×10^{-5}	2.95×10^{-8}	6.3×10^{-3}
Au-198	1.25×10^{-5}	5.6×10^{-8}	1.54×10^{-13}	1.7×10^{-7}
Br- 82	3.90×10^{-4}	8.1×10^{-8}	4.80×10^{-12}	1.0×10^{-6}
Pd-109	1.40×10^{-4}	4.8×10^{-8}	1.72×10^{-12}	7.4×10^{-7}
Cs-138	4.49×10^{-5}	6.6×10^{-10}	5.52×10^{-14}	1.6×10^{-8}

* Based on Technical Specifications

TABLE VI

SUMMARY OF 1978 LIQUID WASTE RELEASES *

NUCLIDE	TOTAL Ci RELEASED	MAX CONCENTRATION AT POINT OF RELEASE (uCi/ml)	AVG ANNUAL CONCENTRATION (uCi/ml)	% OF MAX PERMISSIBLE CONCENTRATION
Ag-110m	9.3×10^{-3}	1.57×10^{-5}	5.37×10^{-8}	5.97×10^{-3}
Co-58	2.3×10^{-3}	3.69×10^{-6}	1.31×10^{-8}	3.28×10^{-4}
Co-60	7.2×10^{-3}	1×10^{-5}	4.19×10^{-8}	4.19×10^{-5}
Cs-134	4.6×10^{-5}	7.46×10^{-8}	2.68×10^{-10}	8.93×10^{-5}
I-131	5.4×10^{-4}	1.14×10^{-6}	3.12×10^{-9}	5.2×10^{-3}
La-140	7.9×10[illegible]	1.68×10^{-7}	4.6×10^{-10}	6.57×10^{-5}
Mn-54	1.3×10^{-3}	1.14×10^{-6}	7.26×10^{-9}	1.82×10^{-4}
Sb-124	1.5×10^{-2}	3.14×10^{-5}	8.61×10^{-8}	1.16×10^{-1}
Unidentified Beta	1.5×10^{-3}	3×10^{-6}	7.5×10^{-7}	8.33×10^{-1}

TOTAL VOLUME RELEASED IN 1978: 51,822 gallons

TOTAL CURIES RELEASED IN 1978: 3.73×10^{-2}

* After dilution by Sanitary Sewer

TABLE 6

Summary of 1979 Air Releases

NUCLIDE	TOTAL Ci RELEASED	MAX CONCENTRATION AT POINT OF RELEASE (uCi/ml)	CONCENTRATION (uCi/ml)	% OF PERMISSIBLE LIMIT
Routine Releases[1]				
Power House Stack Ar-41	2.54×10^{2}	5×10^{-5}	3.44×10^{-6}	0.67
Building Stack Ar-41	2.85×10^{1}	1.4×10^{-6}	4.23×10^{-7}	0.075
Power House Stack Cs-138	1.25×10^{-1}	1.5×10^{-8}	1.69×10^{-9}	0.0003
Non-Routine Releases[2]				
(Power House Stack)				
Ar-41 (Gas)	1.09	9.24×10^{-5}	1.47×10^{-8}	3.6×10^{1}
Au-198 (Particulate)	4.06×10^{-5}	2.25×10^{-8}	5.49×10^{-13}	5.49×10^{-3}
Br-82 (Particulate)	2.03×10^{-4}	9.18×10^{-7}	2.75×10^{-12}	6.88×10^{-3}
Cd-109 (Particulate)	8.71×10^{-5}	4.84×10^{-7}	1.18×10^{-12}	5.9×10^{-3}

Permissible Limit Based on Technical Specifications

Permissible Limit Based on Maximum Permissible Concentrations

TABLE 7

Summary of 1979 Liquid Waste Releases *

NUCLIDE	TOTAL Ci RELEASED	MAX CONCENTRATION AT POINT OF RELEASE (uCi/ml)	AVG ANNUAL CONCENTRATION (uCi/ml)	% OF MAX PERMISSIBLE CONCENTRATION
Na-22	1.61×10^{-5}	3.09×10^{-8}	9.36×10^{-11}	9.4×10^{-6}
Mn-54	1.67×10^{-3}	1.57×10^{-6}	9.71×10^{-9}	2.4×10^{-4}
Co-58	2.47×10^{-3}	2.25×10^{-6}	1.44×10^{-8}	3.6×10^{-4}
Co-60	1.13×10^{-2}	9.23×10^{-6}	6.57×10^{-8}	6.6×10^{-3}
Zn-65	2.76×10^{-4}	3.83×10^{-7}	1.60×10^{-9}	5.3×10^{-5}
Ag-110m	2.53×10^{-2}	4.80×10^{-6}	1.47×10^{-7}	1.6×10^{-2}
Sb-122	4.12×10^{-3}	1.63×10^{-7}	2.39×10^{-8}	3.0×10^{-3}
Sb-124	6.94×10^{-4}	9.61×10^{-6}	4.03×10^{-9}	5.8×10^{-4}
I-131	3.5×10^{-5}	6.73×10^{-8}	2.03×10^{-10}	3.4×10^{-4}
La-140	1.09×10^{-4}	1.88×10^{-7}	6.34×10^{-10}	9.1×10^{-5}
Unidentified Beta	5.48×10^{-3}	1.05×10^{-5}	3.19×10^{-8}	3.5×10^{-2}

Total Volume Released in 1979 = 24,280 gallons

Total Curies released in 1979 = 5.15×10^{-2}

* After Dilution in Sanitary Sewer

TABLE 3 - A

Summary of NSTF Liquid Waste Releases - 1980

NUCLIDE	TOTAL Ci RELEASED	MAX DAILY CONCENTRATIONS AT POINT OF RELEASE[1]		AVERAGE ANNUAL CONCENTRATION[3]	
		uCi/ml	% of MPC[2]	uCi/ml	% of MPC[2]
Cr-51	9.16×10^{-4}	2.42×10^{-6}	4.8×10^{-3}	6.64×10^{-9}	1.3×10^{-5}
Mn-54	3.11×10^{-3}	4.81×10^{-6}	1.2×10^{-1}	2.25×10^{-8}	5.6×10^{-4}
Co-58	8.76×10^{-3}	1.64×10^{-5}	4.1×10^{-1}	6.35×10^{-8}	1.6×10^{-3}
Co-60	1.82×10^{-2}	2.72×10^{-5}	2.7	1.32×10^{-7}	1.3×10^{-2}
Zn-65	8.94×10^{-4}	1.54×10^{-6}	5.1×10^{-2}	6.48×10^{-9}	2.2×10^{-4}
Ag-110m	2.05×10^{-2}	3.68×10^{-5}	4.1	1.49×10^{-7}	1.7×10^{-2}
Sb-124	3.66×10^{-2}	3.65×10^{-5}	5.2	2.65×10^{-7}	3.8×10^{-2}
I-131	3.73×10^{-2}	9.31×10^{-7}	1.5	2.7×10^{-7}	4.5×10^{-1}
La-140	1.3×10^{-3}	3.28×10^{-6}	4.7×10^{-1}	9.42×10^{-9}	1.4×10^{-3}
Ba-140	5.01×10^{-4}	1.33×10^{-6}	1.7×10^{-1}	3.63×10^{-9}	4.5×10^{-3}
Unidentified Beta	9.89×10^{-3}	1.56×10^{-5}	17.3	7.17×10^{-8}	7.9×10^{-2}

1 After dilution by the Sanitary Sewer flow rate of 100,000 gal/day

2 Based on 10NYCRR16-A, Table 4, Schedule I, Column 2, Maximum Permissible Concentrations (MPC's)

3 Obtained by dividing the total Curies released by the annual sewer volume

TOTAL VOLUME RELEASED IN 1980: 23,027 gallons

TOTAL CURIES RELEASED IN 1980: 1.38×10^{-1} (13.8% of Maximum Permissible Limit)

TABLE 2 - A

Summary of NSTF Air Releases - 1980

NUCLIDE	TOTAL Ci RELEASED	MAX CONCENTRATION AT POINT OF RELEASE uCi/ml	AVERAGE ANNUAL CONCENTRATION uCi/ml	MAX RELEASE RATE Curies/sec	MAX RELEASE RATE % OF PERMISSIBLE LIMIT[1]	MAX ANNUAL RELEASE RATE Curies/sec	MAX ANNUAL RELEASE RATE % OF PERMISSIBLE LIMIT[1]
Routine Releases							
Power House Stack Ar-41	4.81×10^{2}	5.1×10^{-5}	6.51×10^{-6}	1.5×10^{-4}	1.22	1.5×10^{-5}	1.27
Building Stack Ar-41	8.88	5.6×10^{-7}	1.33×10^{-7}	1.6×10^{-6}	1.3×10^{-2}	2.8×10^{-7}	2.3×10^{-2}
Power House Stack Cs-138	8.53×10^{-1}	5.9×10^{-8}	1.15×10^{-8}	1.7×10^{-7}	2×10^{-3}	2.7×10^{-8}	3×10^{-3}
Non-Routine Releases (Power House Stack)							
Ar-41 (gas)	1.34×10^{1}	9.3×10^{-5}	1.81×10^{-7}	2.7×10^{-4}	2.23	4.2×10^{-7}	3.5×10^{-2}
Au-198 (Particulate)	1.26×10^{-4}	7.0×10^{-7}	1.71×10^{-12}	2.0×10^{-6}	6.7×10^{-2}	4.0×10^{-12}	1.3×10^{-6}
Br-82 (Particulate)	9.00×10^{-6}	1.5×10^{-8}	1.22×10^{-13}	4.3×10^{-8}	3.6×10^{-4}	2.9×10^{-13}	2.4×10^{-8}
Pd-109 (Particulate)	8.8×10^{-6}	1.03×10^{-8}	1.19×10^{-13}	3×10^{-8}	4.9×10^{-4}	2.8×10^{-13}	4.7×10^{-8}
K-42 (Particulate)	4.1×10^{-3}	1.2×10^{-5}	5.55×10^{-11}	3.4×10^{-5}	1.6×10^{-1}	1.3×10^{-10}	6.2×10^{-6}
Cu-64 (Particulate)	5.4×10^{-5}	4.54×10^{-8}	7.31×10^{-13}	1.3×10^{-7}	6.2×10^{-4}	1.7×10^{-12}	8.2×10^{-8}

1 Permissible limit based on technical specifications

TABLE 2-A
Summary of NSTF Air Releases - 1981

NUCLIDES RELEASED	TOTAL Ci	AVE ANN. CONC uCi/cc	MAX. CONC uCi/cc	MAX REL RATE Ci/sec	MAX REL RATE % Lim*	ANNUAL RATE Ci/sec	ANNUAL RATE % Lim*
Routine Releases:							
Building Air							
(Ar-41)	1.2E1	1.79E-7	7.0E-7	1.49E-6	1.1E-2	3.8E-7	3.2E-2
Stack Gas							
(Ar-41)	2.6E2	3.64E-6	5.3E-5	1.49E-4	1.1	8.2E-6	6.9E-1
Stack Part.							
(Cs-138)	2.0E-2	2.86E-10	2.0E-9	5.67E-9	5.5E-5	4.5E-10	7.1E-5
Non-Routine Releases:							
Powerhouse Stack							
Ar-41 (gas)	8.5E-1	1.19E-8	1.9E-4	5.35E-4	3.8	2.7E-8	2.2E-3
Au-198(par)	4.7E-4	6.62E-12	7.6E-7	2.11E-6	7.5E-2	1.5E-11	6.2E-6
K-40 (par)	2.3E-6	3.25E-14	1.4E-8	3.89E-8	9.1	4.4E-13	1.0E-5
I-128 (par)	1.4E-7	1.95E-13	8.2E-8	2.28E-7	2.2E-3	7.4E-14	8.2E-9

* Permissible limit based on Technical Specifications.

TABLE 3-A
Summary of NSTF Liquid Waste Releases - 1981

NUCLIDE	TOTAL Ci RELEASED	MAX. DAILY CONC AT POINT OF RELEASE * uCi/ml	MAX. DAILY CONC AT POINT OF RELEASE * % MPC**	AVE ANNUAL CONC*** uCi/ml	AVE ANNUAL CONC*** % MPC**
H-3	6.33E-3	7.80E-6	7.80E-3	4.59E-8	4.59E-5
Na-22	3.46E-4	5.42E-7	5.42E-2	2.51E-9	2.51E-4
Cr-51	9.29E-4	2.46E-6	1.23E-5	6.73E-9	1.35E-5
Mn-54	4.11E-3	3.81E-6	9.53E-2	2.98E-8	7.45E-4
Co-58	1.00E-2	6.51E-6	1.63E-1	7.25E-8	1.81E-3
Co-60	2.10E-2	1.90E-5	1.90	1.52E-7	1.52E-2
Zn-65	8.89E-4	1.29E-6	4.30E-2	6.44E-9	2.15E-4
Se-75	1.95E-4	5.16E-7	5.73E-3	1.41E-9	1.57E-5
Ag-110m	2.21E-2	2.16E-5	2.4	1.60E-7	1.78E-2
Sb-124	6.97E-3	6.77E-6	9.67E-1	5.05E-8	7.21E-3
I-131	8.35E-4	1.28E-6	2.13	6.05E-9	1.01E-2
La-140	3.86E-4	1.02E-6	1.46E-1	2.67E-9	3.81E-4
Unident. beta	2.02E-2	1.67E-5	18.56	1.46E-7	1.62E-1

* After dilution by the sanitary sewer flow rate of 100,000 g/d
** Based on 10NYCRR16-A, Table 4, Sched I, Col 2, Max Perm Conc.
***Obtained by dividing total Ci released by annual sewer flow

TOTAL VOLUME RELEASED IN 1981: 34,455 gallons

TOTAL CURIES RELEASED IN'1981: 9.43E-2 (9.43% of Max Perm Limit)

Corrected Data for 1981 - mp 9/22/82

State University of New York at Buffalo
Radiation Protection Department
Program UPSTACKS

NSTF ARGON-41 AND CESIUM-138 AIR RELEASES FOR 1981

	POINT OF RELEASE			
	BUILDING AIR	STACK GAS	STACK PARTICULATE	UNITS
Nuclide	Argon-41	Argon-41	Cesium-138	
Total amount	1.1E+01	2.6E+02	3.5E-02	Curies
Maximum values:				
concentration	7.0E-07	5.4E-05	4.0E-09	uCi/cc
rate	1.5E-06	[illegible]E-04	1.1E-08	Ci/sec
limit*	1.2E-02	[illegible].2E-02	3.0E-05	Ci/sec
% of limit*	1.2E-02	1.3E+00	[illegible].8E-02	percent
Annual average:				
concentration	1.7E-07	3.7E-06	4.9E-10	uCi/cc
rate	3.6E-07	8.4E-06	1.1E-09	Ci/sec
limit*	1.2E-03	1.2E-03	3.0E-06	Ci/sec
% of limit*	3.0E-02	7.0E-01	3.7E-02	percent
Monitor sensitivities:				
JANUARY 1 -	1.7E-09	3.0E-09	7.6E-13	uCi/cc-CPM
AUGUST 1 -	2.5E-09	2.7E-09	7.6E-13	uCi/cc-CPM

Additional data:

Period included in calculations	=	365	days
Number of transient releases	=	0	
Time of reactor operation	=	5411.45	hours
Ventillation system ON	=	5733	hours
Stack flow rates:			
Building air	=	7.65E+09	cc/hr
Stack (vent. ON)	=	1.02E+10	cc/hr
Stack (shut-down)	=	4.25E+09	cc/hr

* = Permissible limit based on NSTF Technical Specifications.

State University of New York at Buffalo
Radiation Protection Department
Program UFSTACKS

NSTF ARGON-41 AND CESIUM-138 AIR RELEASES FOR 1982

	POINT OF RELEASE			
	BUILDING AIR	STACK GAS	STACK PARTICULATE	UNITS
Nuclide	Argon-41	Argon-41	Cesium-138	
Total amount	6.0E+00	1.6E+02	2.4E-02	Curies
Maximum values:				
concentration	5.6E-07	5.2E-05	3.0E-09	uCi/cc
rate	1.2E-06	1.5E-04	8.4E-09	Ci/sec
limit*	1.2E-02	1.2E-02	3.0E-05	Ci/sec
% of limit*	10.0E-03	1.2E+00	2.8E-02	percent
Annual average:				
concentration	8.9E-08	2.2E-06	3.3E-10	uCi/cc
rate	1.9E-07	5.1E-06	7.9E-10	Ci/sec
limit*	1.2E-03	1.2E-03	3.0E-06	Ci/sec
% of limit*	1.6E-02	4.3E-01	2.6E-02	percent
Monitor sensitivities:				
JANUARY 1 -	2.5E-09	2.7E-09	5.6E-13	uCi/cc-CPM
FEBRUARY 28 -	1.2E-09	1.7E-09	5.6E-13	uCi/cc-CPM

Additional data:

Period included in calculations	=	365	days
Number of transient releases	=	229	
Time of reactor operation	=	5544.25	hours
Ventillation system ON	=	6261	hours
Stack flow rates:			
Building air	=	7.65E+09	cc/hr
Stack (vent. ON)	=	1.02E+10	cc/hr
Stack (shut-down)	=	4.25E+09	cc/hr

* = Permissible limit based on NSTF Technical Specifications.

State University of New York at Buffalo

NYSDH RADIOACTIVE MATERIALS LICENSE # 1051
Summary of NSTF Liquid Waste Releases - 1982

NUCLIDE	TOTAL Ci RELEASED	MAX. DAILY CONC AT POINT OF RELEASE * uCi/ml	% MPC**	AVE ANNUAL CONC*** uCi/ml	% MPC**
H-3	4.02E-3	1.06E-5	1.06E-2	2.19E-8	2.91E-5
Na-24	1.53E-3	4.05E-6	6.75E-2	1.11E-8	1.85E-4
Cr-51	7.04E-4	9.95E-6	1.99E-2	5.10E-8	1.02E-4
Mn-54	2.98E-3	6.08E-6	1.52E-1	2.16E-8	5.40E-4
Co-58	1.4.E-2	1.90E-5	4.75E-1	1.02E-7	2.55E-5
Co-60	1.31E-2	2.03E-5	2.02	9.49E-8	9.49E-3
Ag-110m	2.76E-2	3.10E-5	3.44	2.00E-7	2.22E-2
Sb-124	8.08E-3	1.66E-5	2.37	6.29E-8	8.99E-3
I-131	4.39E-3	9.44E-6	15.7	3.18E-8	5.30E-2
Unident. beta	3.03E-2	4.05E-5	45.0	2.20E-7	2.44E-1

TOTAL VOLUME RELEASED IN 1982: 20,375
TOTAL CURIES RELEASED IN 1982: 1.14E-1 Ci (11.4% of Max. Perm. Limit)

* After dilution by the sanitary sewer flow rate of 100,000 g/d
** Based on 10NYCRR16-A, Table 4, Sched I, Col 2, Max Perm Conc.
***Obtained by dividing total Ci released by annual sewer flow

State University of New York at Buffalo
Radiation Protection Department

Summary of Radioactive Waste Disposal
Licenses 1049, 1051 and R-77

Fiscal Year	LSV	ANM	DRY	LIQ	Total Volume Cu.ft.	Total Activity Curies	Number of Shipments
1978	473	67	190	15	5625	3.375	5
1979	236	34	95	30	3027	8.38	3
1980	389	120	46	76	4733	3.70	4
1981	243	54	160	57	2849	3.72	3
1982	1	30	30	0	458	0.333	1

1.0 Definitions

1.1 Channel Calibration - A Channel Calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the Channel Test.

1.2 Channel Check - A Channel Check is a qualitative verification of acceptable performance by observation of channel behavior. This verification where possible shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.3 Channel Test - A Channel Test is the introduction of a signal into the channel to verify that it is operating.

1.4 Control Blade - A neutron absorbing blade used to control core reactivity but is not magnetically coupled to its drive unit.

1.5 Control - Safety Blade - A neutron absorbing blade used to control the reactivity of the core. A Control-Safety Blade is magnetically coupled to its drive unit allowing it to perform the function of a safety device when the magnet is deenergized.

1.6 Experiment - An Experiment, as used herein, is any of the following:

a. An activity utilizing the reactor system or its components or the neutrons or radiation generated therein;

b. An evaluation or test of a reactor system operational, surveillance, or maintenance technique;

c. An experimental or testing activity which is conducted within the confinement or containment system of the reactor; or

d. The material content of any of the foregoing, including structural components, encapsulation or confining boundaries, and contained fluids or solids.

1.7 Experimental Facility - An Experimental Facility is any structure or device which is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

1.8 Explosive Material - Explosive Material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosive Hazard in "Dangerous Properties of Industrial Materials" by N.I. Sax, Third Ed. (1968), or is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, 1966, "Identification System for Fire Hazards of Materials", also enumerated in the "Handbook for Laboratory Safety" 2nd Ed. (1971) published by the Chemical Rubber Co.

1.9 Fast Scram - Fast Scram is a rapid reduction of the magnet holding current of the Control Safety Blades until the blades fall by gravity into the reactor core.

1.10 Fuel Assembly - A grouping of fuel elements which is not taken apart during the charging and discharging of a reactor core.

1.11 Fuel Element - The smallest structurally discrete part of a reactor which has fuel as its principal constituent. Same as fuel pin.

1.12 Limiting Condition of Operation (LCO) - Limiting Conditions for Operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. (10 CFR 50.36)

1.13 Limiting Safety System Settings (LSSS or LS^3) - Limiting Safety System Settings are for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be chosen such that automatic protective action will correct the abnormal situation before a safety limit is exceeded. (10 CFR 50.36)

1.14 Measured Value - The Measured Value of a process variable is the value of the variable as indicated by a measuring channel.

1.15 Measuring Channel - A Measuring Channel is the combination of sensor, amplifiers, and output devices which are used for the purpose of measuring the value of a process variable.

1.16 Movable Experiment - A Movable Experiment is one which may be inserted, removed, or manipulated while the reactor is critical.

1.17 Operable - Operable means that a component or system is capable of performing its intended function in its normal manner.

1.18 Operating - Operating means that a component or system is performing its intended function in its normal manner.

1.19 Permanent Experimental Facility - Those experimental facilities that would require considerable effort and planning to remove or alter such as beam tubes, thermal column, etc.

1.20 Potential Reactivity Worth of an Experiment - The Potential Reactivity Worth of an Experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter equipment position or configuration.

1.21 Reactivity Limits - The Reactivity Limits are those limits imposed on reactor core excess reactivity. Quantities are referenced specifically to a cold core (80 - 100°F) with the effect of xenon poisoning on core activity accounted for if greater than or equal to 0.05% delta k/k. The reactivity worth of samarium in the core will not be included in excess reactivity limits. The reference core condition will be known as the cold, xenon-free critical condition.

1.22 Reactor Operations - Reactor Operation means that the control blades installed in the core are not fully inserted, that the console key is in the keyswitch, or manipulations are being conducted in the pool that could affect core reactivity.

1.23 Reactor Safety System - The Reactor Safety System is that combination of safety channels and associated circuitry which forms the automatic protective system for the reactor or provides information which requires manual protective action to be initiated.

1.24 Reactor Secured - The reactor is secured when a shutdown checklist has been completed.

1.25 Reactor Shutdown - The reactor is considered shut down if all control-safety blades are fully inserted, the console key is removed, and no manipulations are being conducted in the pool that could affect core reactivity. When the reactor is shut down, an operator must be in the facility but not necessarily in the control room.

1.26 Readily Available on Call - "Readily Available on Call" shall mean that the licensed senior operator shall ensure that he is within a reasonable driving time (1 hour) from the reactor building. The licensed senior operator shall always keep the licensed operator informed of where he may be contacted.

1.27 Removable Experiment - A Removable Experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

1.28 Reportable Occurrence - A Reportable Occurrence is any of the following:

a. Operation in excess of a safety limit as set forth in section 2.1;

b. Discovery of a safety system setting less conservative than the limiting setting established in the Technical Specifications;

c. Operation in violation of a limiting condition for operation established in the Technical Specifications;

d. A safety system component malfunction or other component or system malfunction which could, or threatens to, render the safety system incapable of performing its intended safety functions;

e. Release of fission products from a failed fuel element;

f. An uncontrolled or unplanned release of radioactive material from the restricted area of the facility in excess of applicable limits;

g. An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials within the restricted area in excess of the limits specified in Appendix B, Table 1 of 10 CFR 20;

h. Conditions arising from natural or man-made events that affect or threatens to affect the safe operation of the facility; or

i. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

1.29 Rundown - Rundown is the automatic insertion of the Control-Safety Blades.

1.30 Safety Channel - A Safety Channel is a measuring channel in the reactor safety system.

1.31 Safety Limit (SL) - Safety Limits are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity. (10 CFR 50.36)

1.32 Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be Secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraint shall exert sufficient force on the experiment to overcome the expected effects of hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which might arise as a result of credible malfunctions.

1.33 Slow Scram - Slow Scram is the shutoff of electrical power to the units providing the magnet holding current with subsequent decay of the magnet holding current until the blades fall by gravity into the reactor core.

1.34 Static Reactivity Worth - As used herein, the Static Reactivity Worth of an experiment is the absolute value of the reactivity change which is measurable by calibrated control or regulating rod comparison methods between two defined terminal positions or configurations of the experiment. For removable experiments, the terminal positions are fully removed from the reactor and fully inserted or installed in the normal functioning or intended position.

1.35 True Value - The True Value of a process variable is its actual value at any instant.

1.36 Unscheduled Shutdown - An Unscheduled Shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not to include shutdowns which occur during testing or check-out operations.

1.37 Unsecured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be Unsecured if it is not and when it is not secured as defined in 1.32 above. (Secured Experiment).