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Report No. AN-AGCR-772

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

OPERATION OF ML-1 REACTOR SKID IN GCRE: SAFETY EVALUATIONS REPORT

October 1964

GENERAL TIRE

AEROJET-GENERAL NUCLEONICS

SAN RAMON, CALIFORNIA

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The information herein is regarded as preliminary and subject to further checking, verification and analysis.

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OPERATION OF ML-1 REACTOR SKID IN GCRE: SAFETY EVALUATION REPORT

October 1964

Approved by: R.H. Chesworth
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Supervising Representative
Contract AT(10-1)-880

AEROJET-GENERAL NUCLEONICS
A SUBSIDIARY OF AEROJET-GENERAL CORPORATION

GENERAL TIRE

Report No. AN-AGCR-772

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

OPERATION OF ML-1 REACTOR SKID IN GCRE SAFETY EVALUATION REPORT*

ABSTRACT

The operation of the ML-1 reactor skid in the modified GCRE facility, utilizing the GCRE reactor coolant circulating and heat removal systems, is described. An evaluation of the safety considerations associated with this mode of operation indicates that the consequences of the maximum credible accident are less severe than those previously approved for operation of the ML-1 reactor at the ML-1 test site or for operation of the GCRE-I reactor in the GCRE facility. Work described was performed by AGN under Contract AT(10-1)-880 with the U.S. Atomic Energy Commission.

*Published by Aerojet-General Nucleonics, San Ramon, California

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

OPERATION OF ML-1 REACTOR SKID IN GCRE

SAFETY EVALUATION REPORT

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

OPERATION OF ML-1 REACTOR SKID IN GCRE

SAFETY EVALUATION REPORT

I. INTRODUCTION

The GCRE Facility at the National Reactor Testing Station in Idaho was provided under the Army Gas-Cooled Reactor Systems Program (AGCRSP) to permit the operational evaluation of a test reactor of the gas-cooled, water moderated concept. The information developed by this evaluation was to provide the basis for the design of the reactor for a mobile, gas-cooled, nuclear power plant (ML-1) being developed under the AGCRSP for the Military Establishment. Additional information concerning the AGCRSP and the ML-1 program is presented in Appendix A.

The GCRE facility consisted basically of a water-filled pool in which the test reactor was operated and a dual-cycle heat removal system which transferred the energy generated in the reactor to atmosphere in a cooling tower. Provision was made to circulate the coolant gas and to control the temperature of the gas in the circulating loop at the desired operating values. A detailed description of the facility has been published in other documents[1,2]*. Operation of the test reactor was discontinued in April 1961. In January 1962, the decision was made to deactivate the GCRE and, subsequently, the reactor and immediately associated equipment were removed from the pit. The control room and the heat removal equipment were "mothballed" in anticipation of subsequent use in the AGCRSP.

*Numbers in parentheses identify references listed at the end of this report.

The reactor skid for the ML-1 power plant was delivered to the NRTS in February 1961 for operational testing. Since that time only limited testing of the power plant has been possible, primarily because of delays in the delivery of the power conversion equipment and difficulties with the performance of the turbine-compressor sets. The rotating turbomachinery is still considered to be highly developmental equipment and, as a consequence, despite the availability of several machines, a relatively high probability exists that extended operation of the ML-1 power plant may not be possible for some time.

The availability of the GCRE facility has suggested the possibility of operation of the ML-1 reactor skid in that installation during periods when power conversion equipment is not available. A study to determine the feasibility of such a plan indicated that, with relatively minor modification, the GCRE facility could be used to support the test operation of the ML-1 reactor skid and provide meaningful information to the AGCRSP.

This document presents an evaluation of the safety of the operation of the ML-1 reactor skid in the modified GCRE facility. A comparison is drawn between the safety of the projected mode of operation (hereinafter identified as ML-1/GCRE) and the previously approved operation of the ML-1 reactor at the ML-1 test site and of the GCRE-I reactor.

II. SUMMARY

It is proposed that the ML-1 reactor skid be operated in the modified GCRE test facility. Modifications to the facility for this program include provisions for installation of the ML-1 reactor skid in the GCRE reactor pit and for connection of the skid to the existing circulating gas system, the provision of radiation shielding over the pit, the adaptation of the GCRE pool water heat exchanger to serve as a moderator water cooler for the ML-1 reactor skid, physical modification to the GCRE control room to permit the installation of the ML-1 control cab in that room, instrumentation and control circuit modifications required for the safe operation of the ML-1 reactor and minor structural and heating and ventilating modifications. The planned mode of operation of the ML-1 reactor contemplates a maximum reactor power of 2.2 Mw(t)—design power for the ML-1 reactor is 3.3 Mw(t); a nominal system pressure of 200 psia (ML-1 system pressure is 300 psia); and a coolant flow rate of 17.8 lb/sec (the ML-1 flow rate is 26.7 lb/sec). The coolant gas will enter the reactor at 800^{o}F and exit at 1200^{o}F; these values are identical with the design values for ML-1 power plant operation.

The maximum credible accident for the ML-1/GCRE is the same as that defined for the ML-1 operating at the ML-1 test site. This accident results from a simultaneous failure of all six reactor control rods and the coolant gas circulation system. In this event, the nuclear excursion would be terminated by a partial meltdown of the core and, assuming the circulating coolant system failure involved a rupture of the gas ducting, the release of fission product activity to the atmosphere. The safety evaluation shows that the direct radiation doses associated with the maximum credible ML-1/GCRE accident are less than those calculated for the maximum credible ML-1 or GCRE-I accident but that,

under certain credible but highly improbable conditions, the inhalation dose could be considerably higher than that predicted in the event of either the maximum credible ML-1 or GCRE-I accidents.

III. DESCRIPTION OF THE ML-1/GCRE OPERATION

A. CONFIGURATION

The projected ML-1/GCRE operation contemplates the placement of the ML-1 reactor skid in the GCRE reactor pit (maintained in the dry condition) on a suitable footing. The reactor skid is coupled to the circulating gas ducts by a technique which is identical with that used to couple the reactor skid to the power conversion skid in the ML-1 power plant. In this configuration, reactor coolant is circulated through the GCRE heat removal system and oil-fired gas preheater by the existing GCRE compressor. The moderator water is circulated by the pump on the reactor skid through the relocated pool water heat exchanger where energy is transferred to the GCRE cooling water system. The existing reactor skid-to-power conversion skid cables will connect the reactor skid to a special junction box located in the GCRE work area from which point the existing 100-foot sections of power plant-to-control cab cables will be routed to the ML-1 control cab located in the GCRE control room. This arrangement will permit normal operation of the ML-1 reactor skid from the ML-1 control cab and will retain all existing scram circuits, interlocks, and controls. Special cables will connect the GCRE facility interlocks and scram circuits to the ML-1 control cab to provide fully integrated protection in the event of non-standard conditions.

An auxiliary shield tank, filled with borated water, will be placed over the reactor to attenuate the relatively high neutron and gamma fluxes which exist above the reactor during power operation. The reactor pit will be covered with concrete shielding blocks to control radiation levels in the GCRE test facility to acceptable values. A forced air ventilation system is provided for the reactor pit; this air will be discharged through the existing building exhaust system and 150-foot high stack. The ML-1/GCRE operating configuration concept is shown in Figure 1. Details of the GCRE facility modification are presented in Appendix B.

B. OPERATING PARAMETERS

The nominal operating conditions for the ML-1/GCRE and the ML-1 power plant are compared in Table 1. The ML-1/GCRE state point diagram is shown in Figure 2; for comparison, the ML-1 power plant state point diagram is shown in Figure 3.

TABLE 1 - COMPARISON OF ML-1/GCRE AND ML-1 OPERATING PARAMETERS

Parameters	ML-1/GCRE	ML-1
Reactor power, Mw(t)	2.2	3.3
Reactor inlet temperature, ^{o}F	800	800
Reactor outlet temperature, ^{o}F	1200	1200
Coolant mass flow rate, lb/sec	17.8	26.7
Reactor inlet coolant pressure, nominal, psia	200	300

The primary limits on ML-1/GCRE operation are established by the allowable gas pressure in the GCRE gas preheater and the heat rejection capacity of the GCRE cooling system. Special tests at reactor powers higher than 2.2 Mw(t) are possible if the reactor inlet gas temperature is lowered. The loop pressure can be increased to 300 psia at reduced preheater outlet temperatures. Neither of these modes of operation is contemplated for ML-1/GCRE.

C. OPERATING PROCEDURES

The operation of the ML-1 reactor and immediately associated equipment will be controlled by an operator located in the ML-1 control cab which will be placed in the GCRE control room. The operation of the heat removal system and other GCRE equipment will be controlled by an operator seated at the GCRE console. Communication between the two operators will be possible at normal voice levels. A third operator will be available in the control room to record data displayed on the instrumentation racks and to make periodic inspections of the reactor test area. The entire operation will be under the supervision of a qualified supervisor normally stationed in the control room.

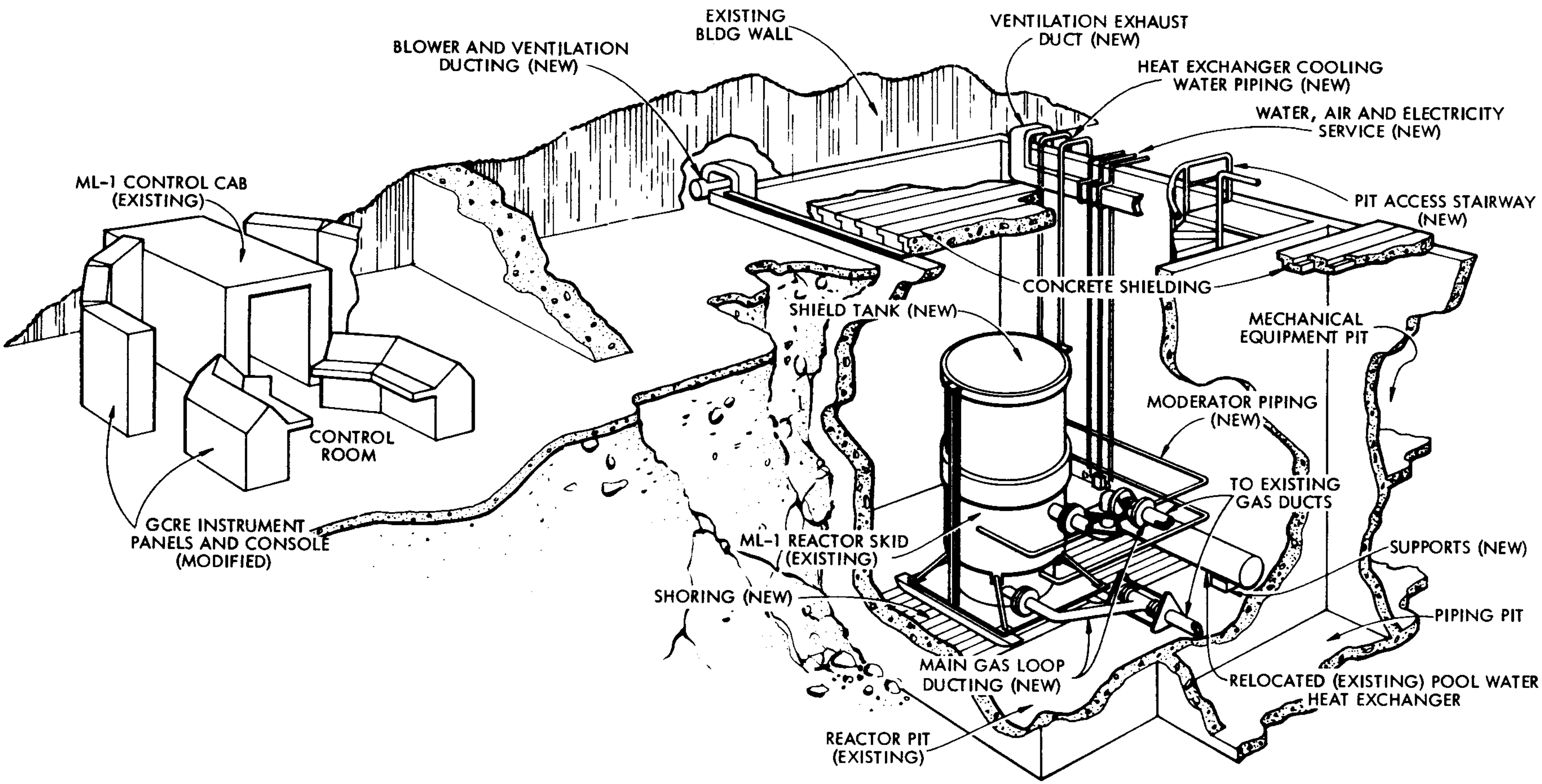

FIGURE 1. ML-1/GCRE OPERATING CONFIGURATION CONCEPT

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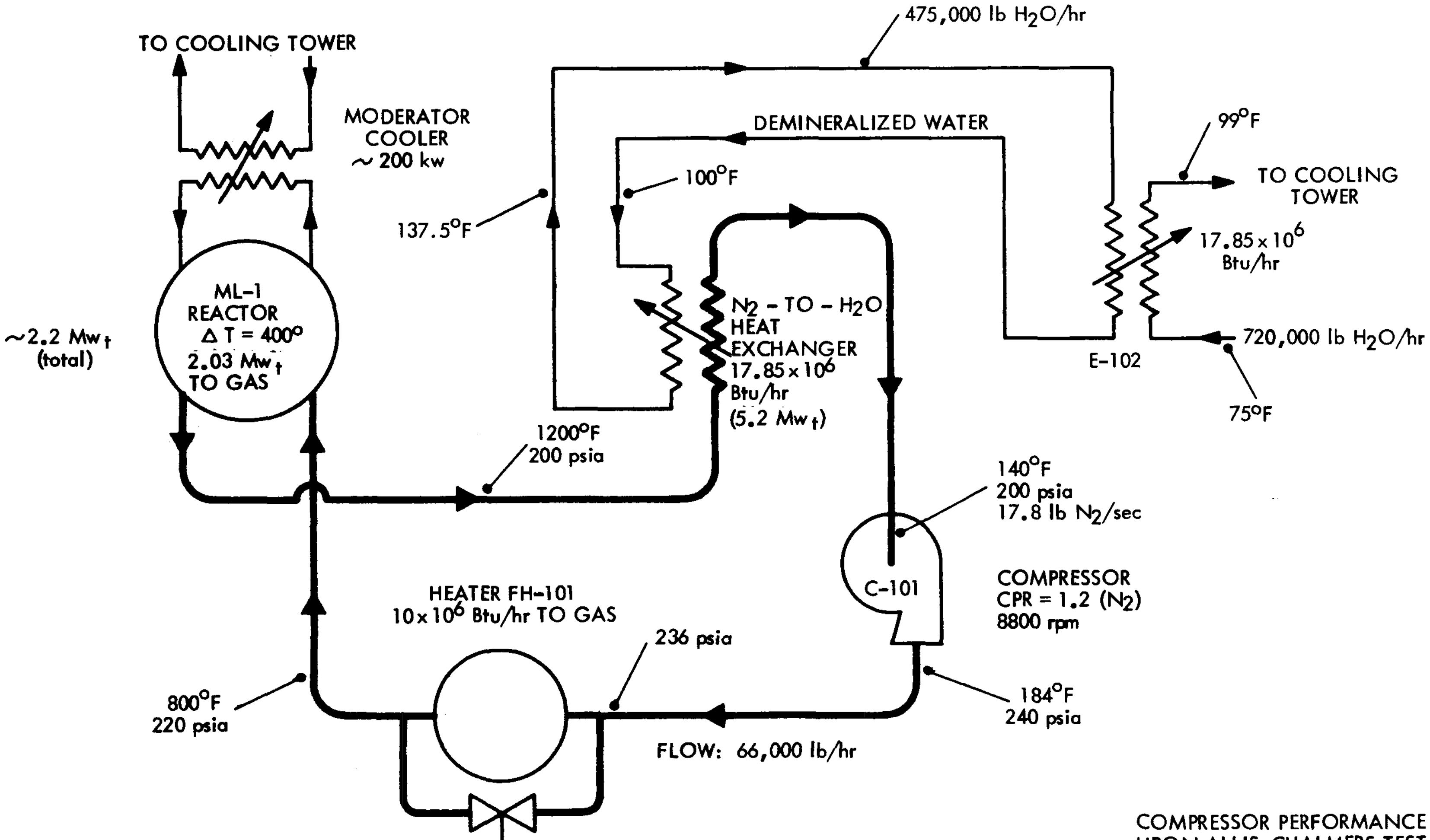


COMPRESSOR PERFORMANCE BASED UPON ALLIS-CHALMERS TEST REPORT - DH 3M COMPRESSOR - SERIAL NO. 5308, AIR DATA CONVERTED TO N_2

FIGURE 2. ML-1/GCRE STATE POINT DIAGRAM

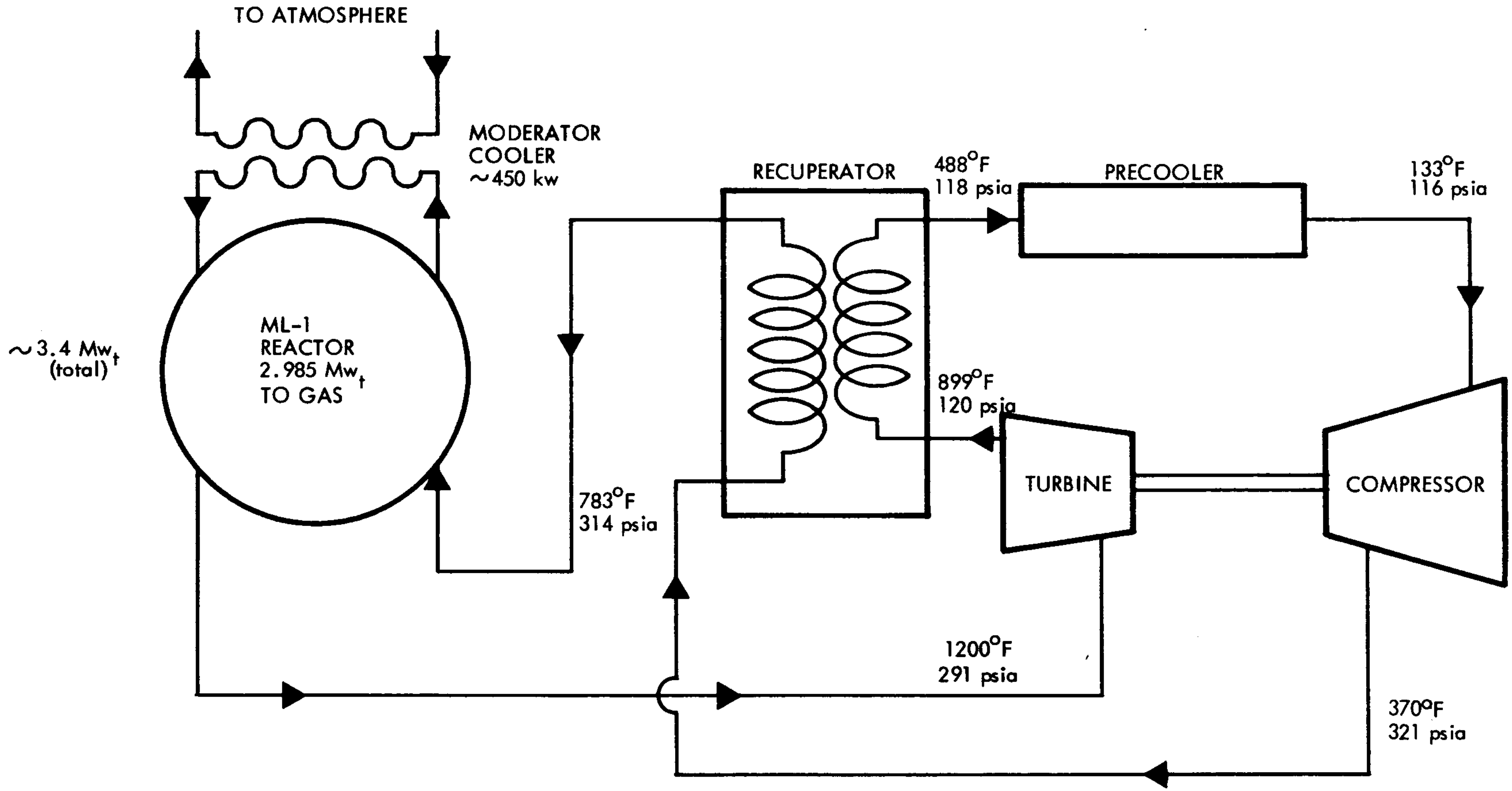


FIGURE 3. ML-1 POWER PLANT STATE POINT DIAGRAM

Standard operating procedures will be used for all ML-1/GCRE operations. Procedures exist for the operation of the GCRE cooling loop which, with minor modification, will be satisfactory for ML-1/GCRE operation. Similarly, standard operating procedures currently exist for the operation of the ML-1 reactor without a power conversion skid. Minor modification of these procedures will be required for ML-1/GCRE operation.

A training program will be conducted to review equipment characteristics and operating procedures with all operators and supervisors prior to the initiation of ML-1/GCRE operations.

IV. SAFETY EVALUATION

In this section, the various accidents which could occur during ML-1/GCRE operation are evaluated and the results of the accidents are postulated. As would be expected, frequent reference is made to the ML-1 power plant safety analysis[3] and the GCRE-I safety analysis[1]. It is the intent of this presentation to show that the ML-1/GCRE operation is basically safer than operation of the ML-1 power plant and that the maximum credible ML-1/GCRE accident results in conditions no more severe than those approved for operation of the GCRE-I reactor in the GCRE facility.

A. ML-1/GCRE MAXIMUM CREDIBLE ACCIDENT

1. Description

The maximum credible accident during operation of the ML-1 reactor skid in the modified GCRE facility is presumed to occur while the reactor is operating at full power, 2.2 Mw(t). The accident is initiated by a rupture of the reactor coolant circulating ducting which causes a complete loss of coolant flow. A simultaneous failure of all reactor safety scram circuits or mechanisms is also hypothesized. Under these conditions, the reactor power would increase and, in the absence of coolant flow, gross overheating of the fuel elements would occur quite rapidly. Approximately 50% of the core would melt, with a subsequent release of the gaseous fission products associated with that core volume. The partial core melting would result in a decrease in system reactivity which would shut down the reactor.

2. Comparison with ML-1 Maximum Credible Accident

The significant differences between the ML-1 maximum credible accident and that described above for the ML-1/GCRE are as follows:

1) The quantity of fission products released (assuming long term operation) would be about 0.75×10^6 curies in the event of the ML-1/GCRE accident as compared with 1.12×10^6 curies from the ML-1 accident. This situation is the result of the reduced maximum power level for the ML-1/GCRE mode of operation.

2) In the event of the ML-1/GCRE accident, the fission products would normally be contained within the test building (except in the case of coolant duct rupture in the heater; see discussion below) from which location a controlled dispersion to the atmosphere would be accomplished. This situation is significantly different from the ground level "puff" release hypothesized for the ML-1 accident.

3. Results of ML-1/GCRE Maximum Credible Accident

The consequences of the occurrence of the maximum credible ML-1/GCRE accident vary widely depending on the location of the assumed rupture of the reactor coolant circulating ducting. Three possible rupture locations are analyzed; it is assumed that equal probability exists for failure in any of the three locations. In each case, it is assumed that the reactor pit ventilating air supply blower, the building exhaust blower and the mechanical equipment room space heater are shut down immediately following the accident to effectively "seal" the test building from the atmosphere.

1) The worst possible conditions result from a rupture of the piping in the oil-fired coolant heater. In this event, the bulk of the fission product activity would be released through the 28-foot high heater stack. Assuming the worst possible meteorological conditions (inversion), and an immediate shutdown of the control room air conditioning system by the automatic control system, if the fission product activity is released over a period of one minute the dose in the GCRE control room from the radioactive cloud would be 3.1 Rem*.

*The calculations to support these and all other exposures cited in this report appear in Appendix C.

If the inversion condition existed and the control room air conditioning system were not shutdown radioactive gas would be introduced into the control room. In this event, the one minute inhalation (thyroid) dose would be 5400 Rem, the beta skin dose would be 350 Rem and the whole body gamma dose would be 0.37 Rem. This same failure could have occurred during GCRE-I operation with more severe exposure.

2) If the coolant ducting rupture occurs in the mechanical equipment pit, the released fission product material would be largely contained within the test building.[4,5] Assuming a uniformly distributed volume source of gamma radiation, occupants in the control room would receive a dose of about 0.5 Rem in 10 minutes; the dose received by occupants of the counting room (the normal point of evacuation in the event of an accident; see Appendix D) would be 0.05 Rem in 10 minutes. If the building exhaust blower were activated to clear the radioactive gas in the test building, and if the worst possible condition (inversion) existed, the gamma dose in the control room would be about 66 mrem in 10 minutes.

3) If the rupture of the coolant piping occurs in the reactor or piping pits, further containment of the fission product activity would result because of the shielding over these pits. Although it is not possible to estimate the gamma source in the test building, it is obvious that, initially, the source would be much lower than that described for a rupture in the mechanical equipment pit. In addition, proper control of the operation of the pit ventilating air blower and building exhaust blower and the use of dilution air by proper operation of the mechanical equipment room space heater would significantly reduce the rate of radioactive gas discharge from the stack and the corresponding exposure levels in the GCRE.

Because of the lower fission product inventory in the ML-1/GCRE mode of operation and the fact that the radioactive gases would be dispersed at some point above ground level (at least 28 ft; in two of three cases, 150 ft), the resultant gamma and inhalation doses at locations in and adjacent to the NRTS would be significantly lower than those resulting from the ML-1 maximum credible accident.

B. OTHER ACCIDENTS

This section of the evaluation discusses other possible accidents associated with the ML-1/GCRE operation. Where no significant difference exists between the accident as discussed in the ML-1 safety analysis[(3)] and that possible in the ML-1/GCRE operating configuration, such is noted in this section.

1. Abnormal Operation of Reactor Control Circuit

Experimental evaluation has established values for the reactor kinetic constants which are slightly different from those used in the ML-1 safety analysis[(3)] but these differences are so small that the result of the accident described therein is essentially unchanged.

2. Electrical Power Failure

During ML-1/GCRE operation, electrical power is supplied to the facility from the NRTS distribution system. In addition, the 300 kw diesel-driven emergency generator in the GCRE facility will be operated at all times when the reactor is in operation. All critical power loads (moderator circulating pump, nuclear instrumentation, control rod drives, compressor emergency drive motor, etc.) are connected at all times to the emergency generator.

The reactor will not be started up unless NRTS power is available and a momentary or complete loss of this source of power will automatically scram the reactor. In addition, automatic switching is available to transfer the loads normally on the emergency generator to the NRTS power system in the event of failure of the emergency generator; such a failure would also scram the reactor.

In the event both power sources failed, the magnetic coupling in the compressor drive would be de-energized and the compressor, gear set and emergency motor would coast to a stop in about one minute. Some gas would be circulated through the core during the coast down and, at the time of complete flow stoppage, the energy generated in the core will heat the fuel and, by conduction and radiation, the moderator water and other reactor components. Assuming no moderator circulation (the worst case), the maximum fuel element cladding temperature would peak at about 1750°F and moderator boiling would occur at an initial rate of approximately 400 lb/hr (decaying to ~ 60 lb/hr after 24 hours).

The rate of boiling is such that all critical reactor components remain covered with water for about twenty-four hours (assuming no make-up) and, as a consequence, no damage will occur to such components.

Actually, provision exists for remote make-up of moderator water. In addition , it is unlikely that one of the two power sources could not be restored in less than 24 hours and even more unlikely that a source of power to drive the moderator circulating pump could not be placed in operation during the same period.

5. Coolant Loss Accompanying Reactor Scram

The ML-1 safety analysis(3) is essentially applicable except that the reactor scram mechanism is different and the probability of coolant loss failure as a result of compressor failure is lower. In the case of ML-1/GCRE operation, the reactor will be shut down in the event of a coolant flow failure by the scram circuit associated with the flowmeter in the main gas loop (as opposed to the rapid pressure loss shutdown mechanism provided for the ML-1). The coolant circulating compressor at GCRE has operated for more than 4,000 hours without a significant problem. This experience permits the conclusion that the probability of compressor failure with this unit is significantly less than that with the highly-developmental turbine-compressor set in the ML-1 power plant.

6. External Causes of Physical Damage and Transportation Hazards

The ML-1 safety analysis(3) covers the potential hazards associated with the relocation of the fueled ML-1 reactor skid from the ML-1 test site to the GCRE facility. A special ANSOP will be prepared for this move.

7. Sabotage

The ML-1 safety analysis(3) covers the proposed ML-1/GCRE operation.

C. CONCLUSIONS

On the basis of the analysis presented in this section of the report, it is concluded that the direct radiation doses associated with the maximum credible ML-1/GCRE accident are less than those calculated for the maximum ML-1 or GCRE accident except in one case. In the event that the maximum credible accident includes the rupture of the reactor coolant circulating duct in the oil-fired heater, if an inversion condition exists and if the control building air conditioning system is not shut down automatically, significant inhalation doses will occur in the control room. While credible, it is considered that this combination of circumstances has a probability of occurrence very near to zero.

The results of the dosage calculations for the maximum credible ML-1/GCRE accident are presented in Table 2, and the comparison of dosages at selected NRTS sites as a result of the ML-1/GCRE, ML-1 and GCRE-I maximum credible accidents are shown in Table 3.

TABLE 2 - RADIATION DOSAGE AT GCRE CONTROL ROOM
FROM ML-1/GCRE MAXIMUM CREDIBLE ACCIDENT

	One Minute Dose in Control Room, Rem	Ten Minute Dose in Control Room, Rem
Release Through Heater Stack in One Minute		
Cloud gamma dose	3.1	3.1
Submersion gamma dose	0.37	38.0
Submersion skid beta dose	350.	
Inhalation dose		
Thyroid	5400.	
Bone	1760.	
Lung	313.	
G. I. Tract	219.	
Muscle	1.2	
Testes	24.2	
Release Through Ventilation Stack in Ten Minutes		
Cloud gamma dose		0.066
Direct gamma dose from reactor room		0.279
Release Contained in Reactor Building		
Direct gamma dose from reactor room		0.48 (1)

(1) Comparable value for GCRE-I maximum credible accident is 113 Rem.

TABLE 3 - COMPARISON OF DOSAGE AT SELECTED NRTS SITES FROM ML-1/GCRE, ML-1 AND GCRE-I MAXIMUM CREDIBLE ACCIDENTS

Dose Location	ML-1/GCRE	ML-1	GCRE-I
	Thyroid Dose, Rem		
Spert II	17.0	33.0	85.0
Central Facilities	2.9	3.6	27.0
Atomic City	2.2	3.1	21.0
EBR-II	0.2	0.3	8.1
	External Cloud Gamma Dose, Rem		
Spert II	0.5	0.7	3.2
Central Facilities	0.2	0.3	1.7
Atomic City	0.15	0.24	1.5
EBR-II	0.06	0.08	0.6

REFERENCES

1. GCRE Hazards Summary Report, AGCRSP, IDO-28506, December 1958

2. Final Summary Report of the Gas-Cooled Reactor Experiment I, AGCRSP, IDO-28598, October 1963

3. Final Hazards Summary Report for the ML-1 Nuclear Power Plant, AGCRSP, IDO-28560, November 1960

4. N. F. Islitzer, The Transport and Disperson of Iodine-131 From the SL-1 Accident, TID-7641

5. N. F. Islitzer, The Role of Meteorology Following the Nuclear Accident in Southeast Idaho, IDO-19310

APPENDIX A

AGCRSP BACKGROUND INFORMATION

This background of the Army Gas-Cooled Reactor Systems Program includes a short history of the Program, a description of the ML-1 power plant, and a selected bibliography.

A. HISTORY

The purpose of the Army Gas-Cooled Reactor Systems Program (AGCRSP) is to develop a mobile nuclear power plant for military field use. The current primary goal of the Program is the fabrication and test operation of a demonstration model of such a plant.

In 1955, at the request of the USAEC Division of Reactor Development, the Oak Ridge School of Reactor Technology performed a study which established the feasibility of the concept of a mobile, closed-cycle, gas-cooled nuclear power plant. Following this work, the Corps of Engineers Nuclear Power Field Office authorized the Sanderson and Porter Company to evaluate power conversion equipment and to prepare a conceptual design for the projected plant. At the conclusion of the Sanderson and Porter work, responsibility for development of the power conversion equipment for the plant was assigned to the Corps of Engineers and the development of the reactor was assigned to the USAEC.

As a result of the above arrangement, parallel programs were undertaken as follows:

1) The Corps of Engineers directed the Stratos Division of Fairchild Engine and Aircraft Corporation to develop a turbine-compressor set suitable for use in the projected plant. The construction of a test facility (GTTF) to evaluate the power conversion equipment was assigned to Aerojet-General Corporation. (The design of the facility was completed by Sanderson-Porter.)

2) Under the direction of the USAEC, Aerojet studied the feasibility of several concepts for the reactor to be incorporated in the power plant. The water-moderated concept was selected as the basis for development because of the modest extrapolation of technology required. Aerojet was awarded the contract for the design and testing of a reactor based on this concept. At the same time, Aerojet was assigned responsibility for the design of a test facility (GCRE) at NRTS which was constructed under the supervision of the USAEC-ID.

Aerojet was designated as systems contractor and performance specifications for the demonstration power plant were evolved in June 1959. Fabrication of the reactor skid was completed in April of 1961, fabrication of the power conversion skid was completed in June 1962, and the power plant first operated as a unit (ML-1) in September 1962. Following a modification and checkout, the ML-1 power plant operated successfully for 101 hours in February and March 1963, for over 600 hr in April and May 1964 and for over 100 hr in September 1964.

B. THE ML-1

The ML-1 is a closed cycle, gas-cooled nuclear power plant developed under the AGCRSP to demonstrate the feasibility of such a plant for military field use. During the design and construction, every reasonable effort was made to incorporate features into the plant which would be directly usable in the design of a field unit. However, since evaluation of the performance of the plant was a major requirement for the ML-1, a significant amount of additional instrumentation was provided. The physical arrangement of the equipment is such that the "prototype" components are easily identified as the:

1) <u>Reactor Skid</u> - a 15 ton unit containing the nuclear reactor and associated shielding and controls;

2) <u>Power Conversion Skid</u> - a 15 ton unit containing the power conversion equipment; and,

3) <u>Control Cab</u> - a 2-1/2 ton unit containing all the instruments and controls for operation of the field plant.

The ML-1 reactor consists of a calandria-type pressure vessel with appropriate inlet and exit gas ducts and plenums. Sixty-one pin-type $BeO-UO_2$ fuel elements are located in the tubes of the calandria. The demineralized water moderator surrounds the calandria and the six semaphore-type control rods operate in the moderator spaces between the calandria tubes. The entire reactor structure is supported inside a nine-foot diameter tank which contains heavy metal shields to permit relocation of the reactor within 24 hours after shutdown from extended operation, and a drainable (borated water) shield to attenuate radiation during reactor operation.

The plant working fluid (99.5 vol% nitrogen, 0.5 vol% oxygen) enters the reactor at $800^{\circ}F$ and approximately 300 psia. The gas is heated to $1200^{\circ}F$ in a single pass over the hot surfaces of the fuel elements. The moderator water is maintained at a temperature of $180^{\circ}F$; energy deposited in the moderator is removed in an water-to-air heat exchanger mounted on top of the power conversion skid. Provision is made for circulation, filtration and demineralization of the moderator water and for circulation and cooling (by exchange with the moderator water) of the shield water.

The hot gas leaving the reactor is expanded in a gas turbine which drives the compressor and alternator. The gas leaving the turbine passes through a regenerative heat exchanger (recuperator), through the system heat sink (air-to-air precooler) and to the compressor suction. The compressor discharges through the recuperator to the reactor inlet, thus completing the closed (modified Brayton) cycle. A lubrication system (including provision for recovery and removal of lubricating oil from the working fluid which leaks past the turbine compressor seals), the electrical switch gear, and miscellaneous power conversion hardware are mounted on the power conversion skid. An a-c, two-speed motor is coupled to the turbine-compressor shaft to provide starting power to the set.

The following auxiliary systems are provided for the ML-1:

1) A deoxygenation system which removes dissolved oxygen from a bypass stream to maintain the moderator system oxygen content below 0.7 ppm.

2) An emergency cooling system which automatically injects a supply of coolant gas into the reactor in the event of a complete stoppage of working fluid flow.

3) A working fluid makeup system to compensate for normal leakage and to provide for initial charging of the system.

4) Waste gas storage facilities to accommodate the charge of radioactive gas in the loop in the event of an emergency.

C. BACKGROUND BIBLIOGRAPHY

The following bibliography provides information on sources of additional background to the material presented in this report. These documents trace the technical evolution of the AGCRSP from inception but do not, in general, document programmatic decisions. Such activity may be inferred from the technical approaches pursued and from the general background information presented in the reports.

The following reports were published by Aerojet-General Nucleonics, San Ramon, California under the Army Gas-Cooled Reactor Systems Program.

DOCUMENT NO.	TITLE	CLASSIFICATION
IDO-28505	GCRE Semiannual Report, 1 November 1956 Through 30 June 1957, 20 February 1958	CRD
IDO-28506	GCRE-I Hazards Summary Report, December 1958, with three addenda, March 1958, February 1960, May 1960	U
IDO-28519	GCRE Semiannual Progress Report, 1 July Through 31 December 1957, 26 June 1958	CRD
IDO-28526	GCRE Semiannual Progress Report, 1 January Through 30 June 1958, 17 October 1958	CRD
IDO-28533	AGCRSP Semiannual Progress Report, 1 July Through 31 December 1958, 28 February 1959	CRD
IDO-28537	Preliminary Hazards Summary Report for the ML-1 Nuclear Power Plant, 22 April 1959	U

DOCUMENT NO.	TITLE	CLASSIFICATION
IDO-28542	AGCRSP Semiannual Progress Report, 1 January Through 30 June 1958, July 1958	U
IDO-28549	AGCRSP Semiannual Progress Report, 1 July Through 31 December 1959, 22 December 1959	U
IDO-28550	The ML-1 Design Report, 16 May 1960	U
IDO-28555	ML-1 Transportability Studies, 23 March 1960	U
IDO-28558	AGCRSP Semiannual Progress Report, 1 January Through 30 June 1960, 11 July 1960	U
IDO-28560	Final Hazards Summary Report for the ML-1 Nuclear Power Plant, with four supplements, 5 August 1960	U
IDO-28567	AGCRSP Semiannual Progress Report, 1 July Through 31 December 1960, 17 December 1960	U
IDO-28573	AGCRSP Semiannual Progress Report, 1 January Through 30 June 1961, 10 August 1961	U
IDO-28581	AGCRSP Semiannual Progress Report, 1 July Through 31 December 1961, 31 January 1962	U
IDO-28590	AGCRSP Semiannual Progress Report, 1 January Through 30 June 1962, 24 August 1962	U
IDO-28597	AGCRSP, Study of the GCRE Tube Bundle Failure, 14 December 1962	U
IDO-28602	AGCRSP Semiannual Progress Report, 1 July Through 31 December 1962, 22 February 1963	U
IDO-28607	AGCRSP Quarterly Progress Report, 1 January Through 31 March 1963, 15 May 1963	U
IDO-28612	AGCRSP Quarterly Progress Report, 1 April Through 30 June 1963, 15 August 1963	U
IDO-28617	AGCRSP Quarterly Progress Report, 1 July Through 30 September 1963, 15 November 1963	U
IDO-28621	AGCRSP Quarterly Progress Report, 1 October Through 31 December 1963, 27 January 1964	U
IDO-28626	AGCRSP Quarterly Progress Report, 1 January Through 31 March 1964, 15 May 1964	U
IDO-28632	AGCRSP Quarterly Progress Report, 1 April Through 30 June 1964, 15 August 1964	U

APPENDIX B

GCRE FACILITY MODIFICATION DETAILS

A detailed description of the modifications to the GCRE facility provided for ML-1/GCRE operation is presented below:

1) Reactor Coolant Ducting: A support frame (see Figure B-1) is provided in the reactor pit to:

- Support the coolant ducts and associated expansion joints
- Position the ducts for proper alignment of the flanges with mating units on the reactor skid
- Position the reactor skid for proper mating of coolant duct and moderator piping flanges
- Support the instrumentation (thermocouple extensions, gas sampling tubing, etc.)

Stainless steel ducting is provided to connect the reactor skid with the existing main gas loop (GCRE) ducting at the wall of the reactor pit. These ducts include expansion joints to compensate for thermal growth and are internally insulated to minimize stresses in the pipe walls and flanges. The arrangement of these ducts is shown in Figures B-2, B-3 and B-4.

2) Moderator Cooling System: The existing pool water heat exchanger has been relocated to a position on the reactor pit floor (Figure B-2). The cooling water service for this unit has been extended from the original to the new location and piping is provided to connect the exchanger to the circulating moderator system on the reactor skid (Figure B-5).

3) Radiation Shielding: A 9-foot diameter by 6-foot high auxiliary shield tank is provided to attenuate radiation streaming from the top of the reactor skid. This tank (Figure B-6) is supported on a structure (Figure B-7) which straddles the reactor skid so that no weight load is imposed on the integral reactor shield. The arrangement of the tank in the pit, with the skid in position, is shown on Figure B-2.

The reactor and piping pits are covered at floor level with concrete shielding approximately 22-inches-thick. This shield is made up of a series of concrete beams to permit removal for access to the pit (Figure B-8).

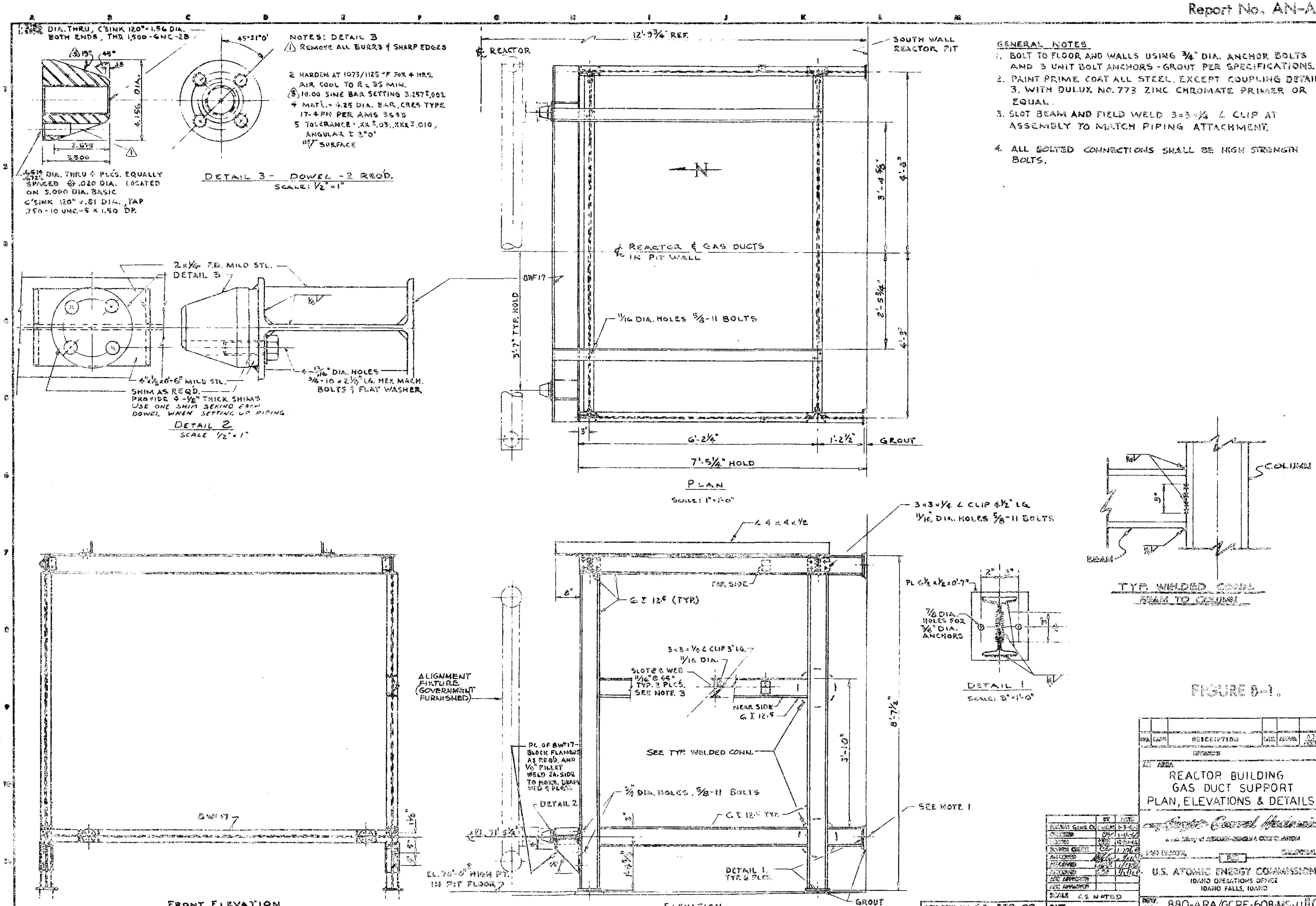
4) Heating and Ventilating System: Provision is made to supply fresh air to the reactor pit and to exhaust both the reactor and piping pit to the existing building ventilating system. The reactor pit is swept with 8500 cfm of either fresh outside air or recirculated building air (Figure B-9). The openings where the ducts penetrate the reactor pit concrete shielding are shielded with specially designed blocks (Figure B-8).

5) Structural Modifications: A concrete pad is provided on the floor of the reactor pit to provide a level surface for the reactor skid and to support the auxiliary shield tank (Figure B-10).

The existing mobile work bridge and rails are removed from the reactor pit and a spiral staircase is installed for access to the pit floor. The upper section of this staircase is removable to permit installation of the concrete shielding blocks.

The exterior doorway in the north wall of the GCRE Control Building and the doorway between the north lobby and the control room are modified to permit movement of the ML-1 control cab into the control room.

6) Instruments and Controls: Provision is made for the placement of the ML-1 control cab in the GCRE control room. The connecting cables shown in Figure B-11 provide for integration of all appropriate GCRE scrams and interlocks into the ML-1 scram logic system as well as for the recording of all pertinent ML-1 data on instrumentation provided in the GCRE instrument racks.


GENERAL NOTES
1. BOLT TO FLOOR AND WALLS USING 3/4" DIA. ANCHOR BOLTS AND 3 UNIT BOLT ANCHORS - GROUT PER SPECIFICATIONS.
2. PAINT PRIME COAT ALL STEEL EXCEPT COUPLING DETAIL 3, WITH DULUX NO. 773 ZINC CHROMATE PRIMER OR EQUAL.
3. SLOT BEAM AND FIELD WELD 3×3×1/4 ∠ CLIP AT ASSEMBLY TO MATCH PIPING ATTACHMENT.
4. ALL BOLTED CONNECTIONS SHALL BE HIGH STRENGTH BOLTS.
NOTES: DETAIL 3
1 REMOVE ALL BURRS & SHARP EDGES
2 HARDEN AT 1075/1125 °F FOR 4 HRS. AIR COOL TO Rc 35 MIN.
3 10.00 SINE BAR SETTING 3.257±.002
4 MAT'L - 4.25 DIA. BAR, CRES TYPE 17-4PH PER AMS 5643
5 TOLERANCE: .XX ±.03, .XXX ±.010, ANGULAR ± 2°0'
DETAIL 3 - DOWEL - 2 REQ'D.
SCALE: 1/2" = 1"
DETAIL 2
SCALE 1/2" = 1"
PLAN
SCALE: 1" = 1'-0"
FRONT ELEVATION
ELEVATION
DETAIL 1
SCALE: 3" = 1'-0"
TYP. WELDED CONN. BEAM TO COLUMN
SOUTH WALL REACTOR PIT
℄ REACTOR & GAS DUCTS IN PIT WALL
ALIGNMENT FIXTURE (GOVERNMENT FURNISHED)
SEE TYP. WELDED CONN.
SEE NOTE 1
GROUT
12'-9 3/4" REF.
7'-5 1/4" HOLD
6'-2 1/4"
8'-7 1/2"
3'-10"
FIGURE B-1.
REACTOR BUILDING
GAS DUCT SUPPORT
PLAN, ELEVATIONS & DETAILS
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
SCALE AS NOTED
AGN DWG. NO. 63-278-00
880-ARA/GCRE-608-MS-11

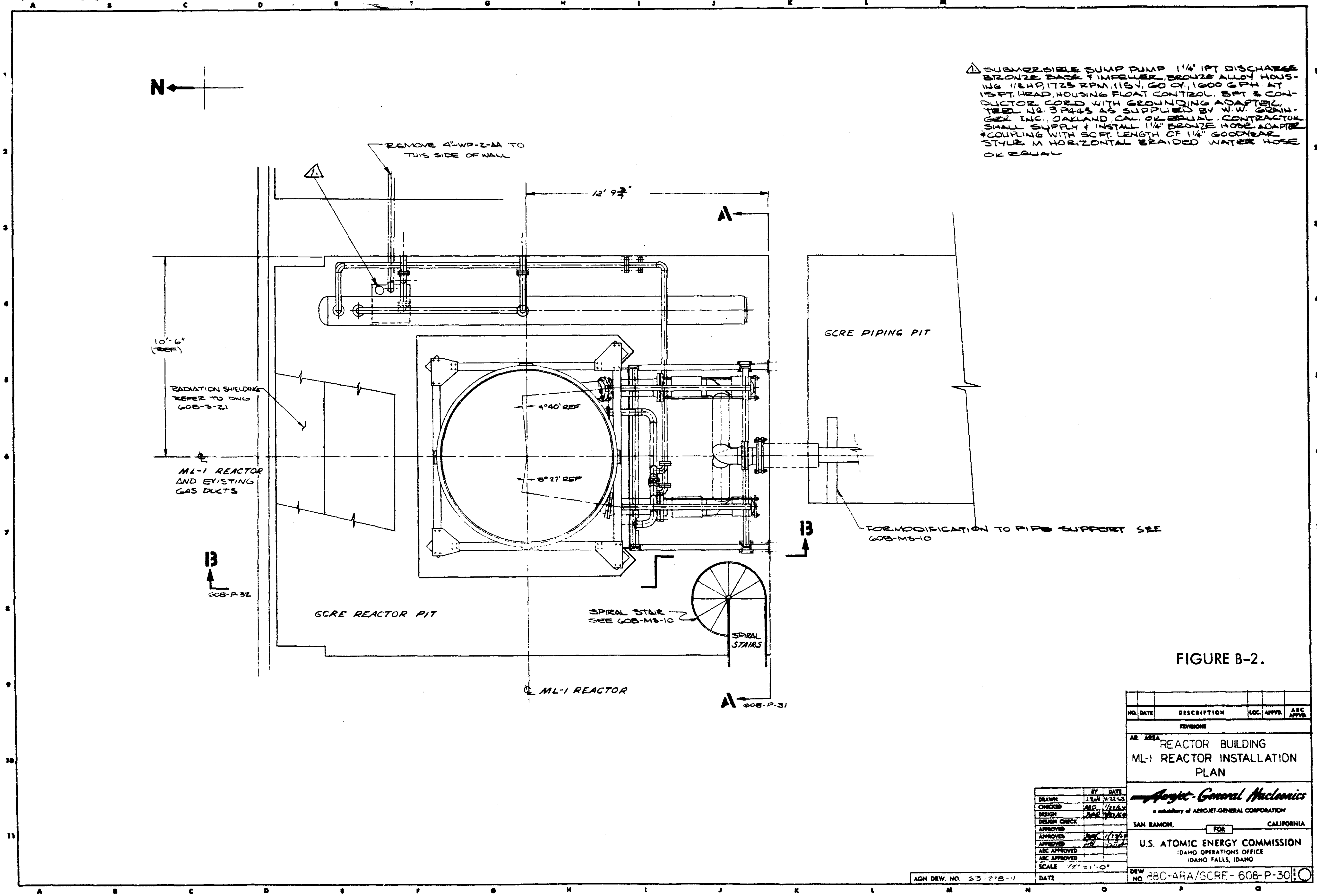

SUBMERSIBLE SUMP PUMP 1¼" IPT DISCHARGE BRONZE BASE & IMPELLER, BRONZE ALLOY HOUSING 1/3 HP, 1725 RPM, 115V, 60 CY., 1600 GPH AT 15 FT. HEAD, HOUSING FLOAT CONTROL, 8 FT 3 CONDUCTOR CORD WITH GROUNDING ADAPTER, TEEL NO. 3P443 AS SUPPLIED BY W.W. GRAINGER INC., OAKLAND, CAL. OR EQUAL. CONTRACTOR SHALL SUPPLY & INSTALL 1¼" BRONZE HOSE ADAPTER & COUPLING WITH 50 FT. LENGTH OF 1¼" GOODYEAR STYLE M HORIZONTAL BRAIDED WATER HOSE OR EQUAL
N
REMOVE 4"-WP-2-AA TO THIS SIDE OF WALL
12' 9¾"
A
10'-6" (REF)
RADIATION SHIELDING REFER TO DWG 608-S-21
GCRE PIPING PIT
4°40' REF
8°27' REF
ML-1 REACTOR AND EXISTING GAS DUCTS
FOR MODIFICATION TO PIPE SUPPORT SEE 608-MS-10
B
608-P-32
GCRE REACTOR PIT
SPIRAL STAIR SEE 608-MS-10
SPIRAL STAIRS
℄ ML-1 REACTOR
A 608-P-31
NO. DATE DESCRIPTION LOC. APPVD. AEC APPVD.
REVISIONS
AR AREA REACTOR BUILDING
ML-1 REACTOR INSTALLATION PLAN
Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA
FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
BY DATE
DRAWN J. Roll 11-22-63
CHECKED
DESIGN
DESIGN CHECK
APPROVED
APPROVED
APPROVED
AEC APPROVED
AEC APPROVED
SCALE ½" = 1'-0"
DATE
AGN DRW. NO. 63-278-11
DRW NO 880-ARA/GCRE-608-P-30

FIGURE B-2.

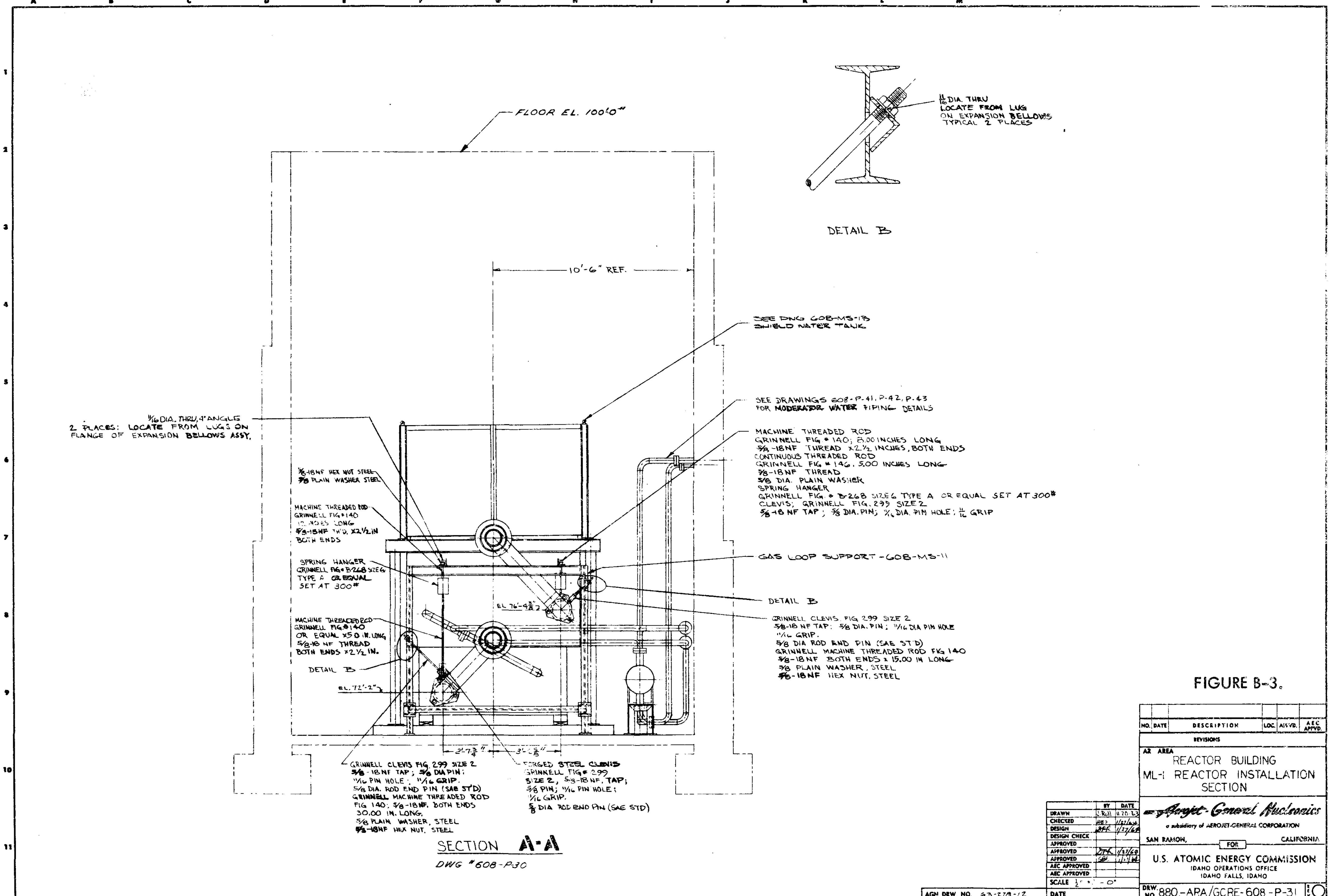

FLOOR EL. 100'-0"
DETAIL B
10'-6" REF.
SEE DWG 608-MS-13 SHIELD WATER TANK
SEE DRAWINGS 608-P-41, P-42, P-43 FOR MODERATOR WATER PIPING DETAILS
MACHINE THREADED ROD GRINNELL FIG # 140; 8.00 INCHES LONG 5/8-18NF THREAD x2½ INCHES, BOTH ENDS
CONTINUOUS THREADED ROD GRINNELL FIG # 146, 5.00 INCHES LONG 5/8-18NF THREAD
5/8 DIA. PLAIN WASHER
SPRING HANGER GRINNELL FIG. # B-268 SIZE 6 TYPE A OR EQUAL SET AT 300#
CLEVIS; GRINNELL FIG. 299 SIZE 2 5/8-18 NF TAP; 5/8 DIA. PIN; 11/16 DIA. PIN HOLE; 11/16 GRIP
GAS LOOP SUPPORT-608-MS-11
DETAIL B
GRINNELL CLEVIS FIG 299 SIZE 2 5/8-18 NF TAP; 5/8 DIA. PIN; 11/16 DIA PIN HOLE 11/16 GRIP. 5/8 DIA ROD END PIN (SAE STD) GRINNELL MACHINE THREADED ROD FIG 140 5/8-18NF BOTH ENDS x 15.00 IN LONG 5/8 PLAIN WASHER, STEEL 5/8-18NF HEX NUT, STEEL
11/16 DIA. THRU, 4" ANGLE 2 PLACES: LOCATE FROM LUGS ON FLANGE OF EXPANSION BELLOWS ASSY.
5/8-18NF HEX NUT STEEL 5/8 PLAIN WASHER STEEL
MACHINE THREADED ROD GRINNELL FIG # 140 10 INCHES LONG 5/8-18NF TH'D. x2½ IN BOTH ENDS
SPRING HANGER GRINNELL FIG # B-268 SIZE 6 TYPE A OR EQUAL SET AT 300#
EL 76'-4 3/8"
MACHINE THREADED ROD GRINNELL FIG # 140 OR EQUAL x 5.0 IN. LONG 5/8-18 NF THREAD BOTH ENDS x2½ IN.
DETAIL B
EL. 72'-2"
GRINNELL CLEVIS FIG 299 SIZE 2 5/8-18NF TAP; 5/8 DIA PIN; 11/16 PIN HOLE; 11/16 GRIP. 5/8 DIA. ROD END PIN (SAE STD) GRINNELL MACHINE THREADED ROD FIG 140; 5/8-18NF, BOTH ENDS 30.00 IN. LONG. 5/8 PLAIN WASHER, STEEL 5/8-18NF HEX NUT, STEEL
FORGED STEEL CLEVIS GRINNELL FIG # 299 SIZE 2, 5/8-18NF. TAP; 5/8 PIN; 11/16 PIN HOLE; 11/16 GRIP. 5/8 DIA ROD END PIN (SAE STD)
SECTION A-A
DWG # 608-P-30
11/16 DIA. THRU LOCATE FROM LUG ON EXPANSION BELLOWS TYPICAL 2 PLACES
NO. DATE DESCRIPTION LOC. APPVD. AEC APPVD.
REVISIONS
AREA
REACTOR BUILDING
ML-1 REACTOR INSTALLATION SECTION
Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA
FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
DRAWN CHECKED DESIGN DESIGN CHECK APPROVED APPROVED APPROVED AEC APPROVED AEC APPROVED
SCALE
DATE
AGN DRW. NO.
DRW. NO. 880-ARA/GCRE-608-P-31

FIGURE B-3.

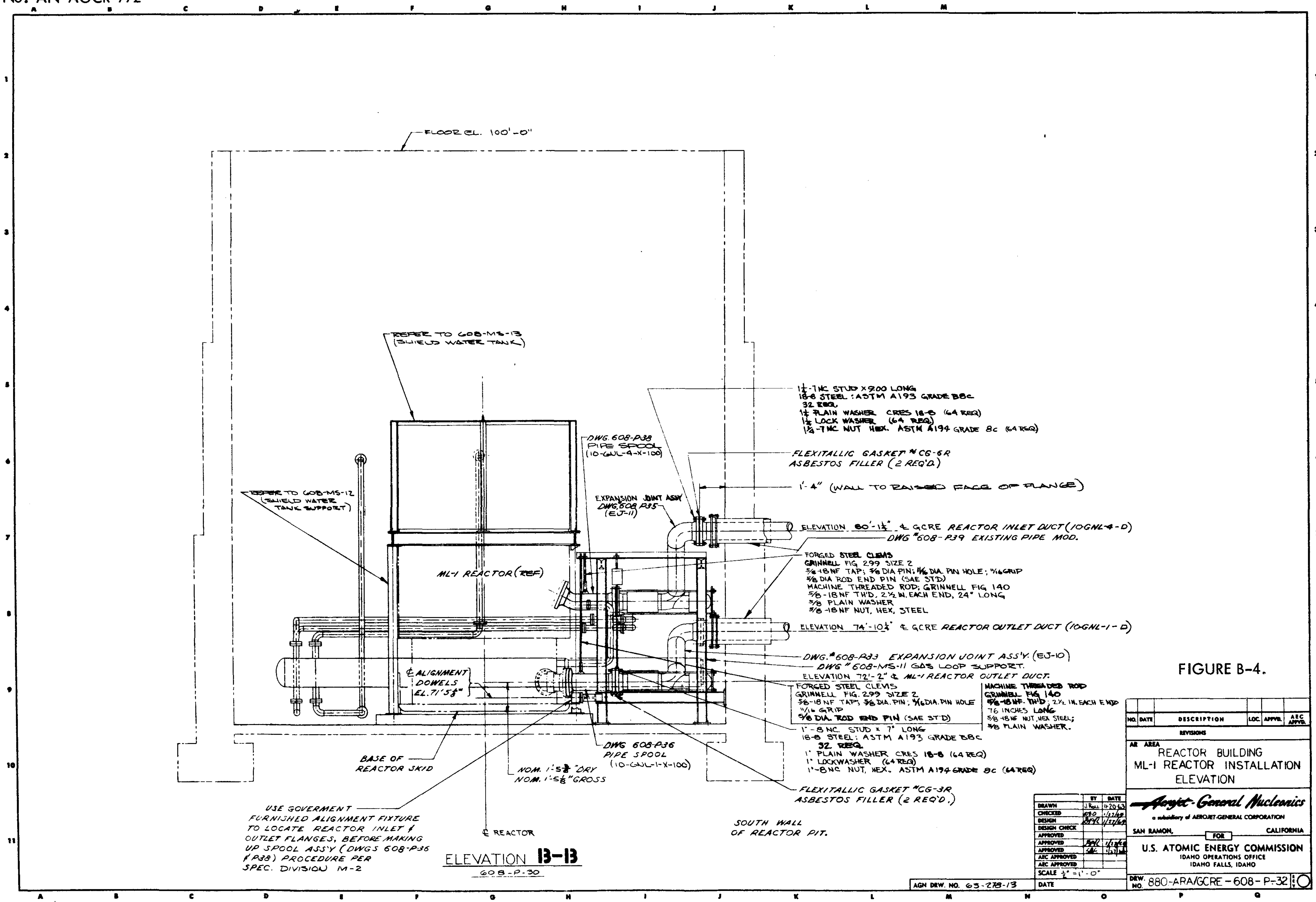

FLOOR EL. 100'-0"
REFER TO 608-MS-13 (SHIELD WATER TANK)
REFER TO 608-MS-12 (SHIELD WATER TANK SUPPORT)
ML-1 REACTOR (REF)
DWG. 608-P38 PIPE SPOOL (10-GNL-4-X-100)
EXPANSION JOINT ASSY DWG. 608 P35 (EJ-11)
1½-7NC STUD X 9.00 LONG 18-8 STEEL: ASTM A193 GRADE B8C 32 REQ.
1½ PLAIN WASHER CRES 18-8 (64 REQ)
1½ LOCK WASHER (64 REQ)
1½-7NC NUT HEX. ASTM A194 GRADE 8C (64 REQ)
FLEXITALLIC GASKET # CG-6R ASBESTOS FILLER (2 REQ'D.)
1'-4" (WALL TO RAISED FACE OF FLANGE)
ELEVATION 80'-1½" ℄ GCRE REACTOR INLET DUCT (10GNL-4-D)
DWG # 608-P39 EXISTING PIPE MOD.
FORGED STEEL CLEVIS GRINNELL FIG 299 SIZE 2
5/8-18NF TAP; 5/8 DIA PIN; 11/16 DIA. PIN HOLE; 11/16 GRIP
5/8 DIA ROD END PIN (SAE STD)
MACHINE THREADED ROD; GRINNELL FIG 140
5/8-18NF THD, 2½ IN. EACH END, 24" LONG
5/8 PLAIN WASHER
5/8-18NF NUT, HEX, STEEL
ELEVATION 74'-10¼" ℄ GCRE REACTOR OUTLET DUCT (10-GNL-1-D)
DWG. # 608-P33 EXPANSION JOINT ASS'Y. (EJ-10)
DWG # 608-MS-11 GAS LOOP SUPPORT.
ELEVATION 72'-2" ℄ ML-1 REACTOR OUTLET DUCT.
FORGED STEEL CLEVIS GRINNELL FIG. 299 SIZE 2
5/8-18NF TAP; 5/8 DIA. PIN; 11/16 DIA. PIN HOLE 11/16 GRIP
5/8 DIA. ROD END PIN (SAE STD)
MACHINE THREADED ROD GRINNELL FIG 140
5/8-18NF THD; 2½ IN. EACH END 76 INCHES LONG
5/8-18NF NUT, HEX STEEL;
5/8 PLAIN WASHER.
1"-8NC. STUD x 7" LONG 18-8 STEEL: ASTM A193 GRADE B8C 32 REQ.
1" PLAIN WASHER CRES 18-8 (64 REQ)
1" LOCKWASHER (64 REQ)
1"-8NC NUT, HEX. ASTM A194 GRADE 8C (64 REQ)
FLEXITALLIC GASKET # CG-3R ASBESTOS FILLER (2 REQ'D.)
℄ ALIGNMENT DOWELS EL. 71'5⅝"
BASE OF REACTOR SKID
DWG 608-P36 PIPE SPOOL (10-GNL-1-X-100)
NOM. 1'-5⅜" DRY
NOM. 1'-5⅝" GROSS
USE GOVERMENT FURNISHED ALIGNMENT FIXTURE TO LOCATE REACTOR INLET & OUTLET FLANGES, BEFORE MAKING UP SPOOL ASS'Y (DWGS 608-P36 & P38) PROCEDURE PER SPEC. DIVISION M-2
℄ REACTOR
SOUTH WALL OF REACTOR PIT.
ELEVATION B-B
608-P-30
NO. DATE DESCRIPTION LOC. APPVD. AEC APPVD.
REVISIONS
AR AREA
REACTOR BUILDING
ML-1 REACTOR INSTALLATION
ELEVATION
Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA
FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
DRAWN
CHECKED
DESIGN
DESIGN CHECK
APPROVED
AEC APPROVED
SCALE ½" = 1'-0"
DATE
AGN DRW. NO. 63-278-13
DWG. NO. 880-ARA/GCRE-608-P-32
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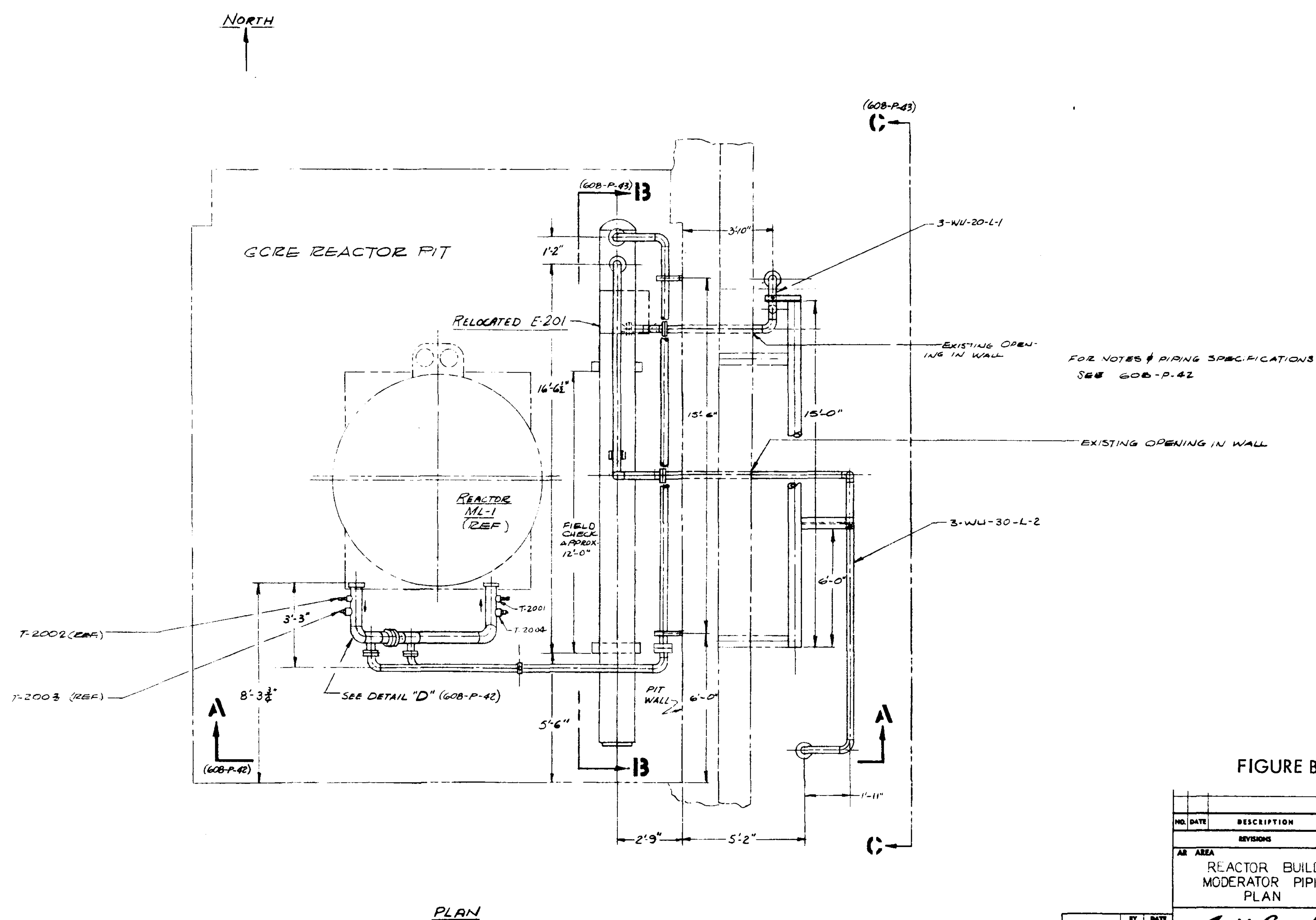
FIGURE B-4.



FOR NOTES & PIPING SPECIFICATIONS SEE 608-P-42

FIGURE B-5

NO.	DATE	DESCRIPTION	LOC.	APPVD.	AEC APPVD.
		REVISIONS			

AR AREA

REACTOR BUILDING MODERATOR PIPING PLAN

Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA

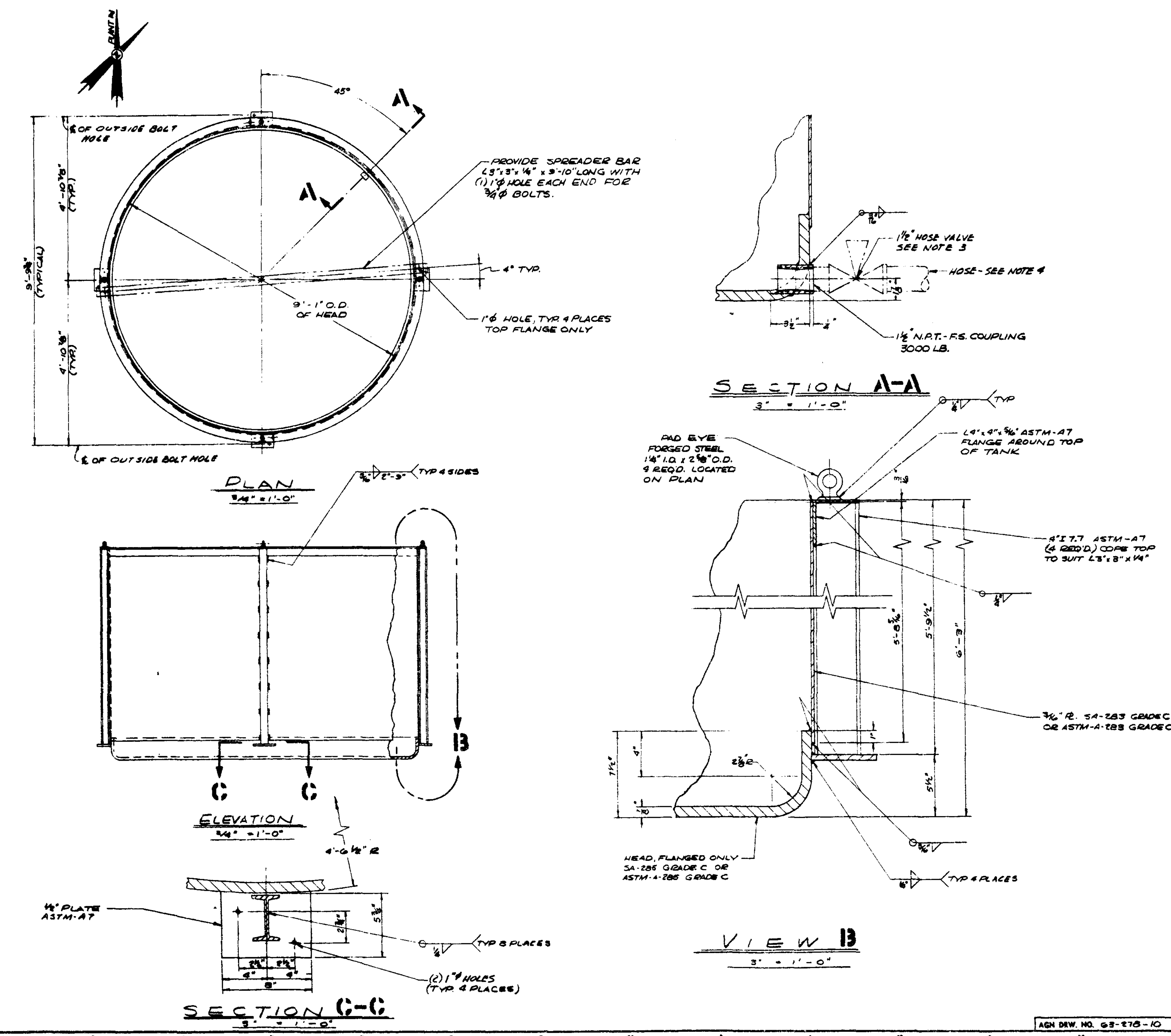
FOR

U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO

SCALE 1/2" = 1'-0"

AGN DRW. NO. 63-278-22

DRW. NO. 880-ARA/GCRE-608-P-41



GENERAL NOTES

1. TANK TO BE NON-CODE, BUT SHALL BE FABRICATED PER ASME BOILER & PRESSURE VESSEL CODE, SECTION VIII, UNFIRED PRESSURE VESSELS
2. LONGITUDINAL SHELL JOINTS TO BE DOUBLE BUTT WELDED
3. CONTRACTOR SHALL SUPPLY 1 1/2" x 3" LONG, STD. WEIGHT BRASS NIPPLE & 1 1/2" BRONZE HOSE GATE VALVE, WEDGE DISC, WITH CAP & CHAIN, CRANE # 451, SCREWED, (HOSE THREAD TO BE AMERICAN STD. FIRE HOSE). INSTALL NIPPLE & VALVE AFTER TANK IS MOUNTED ON SUPPORTS
4. CONTRACTOR SHALL SUPPLY A 35 FT. LENGTH OF 1 1/2" RUBBER COVERED, SINGLE JACKET FIRE HOSE, GOODYEAR OR EQUAL, WITH 1 1/2" BRASS HOSE COUPLING TO MATE WITH HOSE VALVE
5. ENTIRE TANK TO BE SANDBLASTED TO WHITE METAL & PAINTED AS FOLLOWS:
   a. INTERIOR SHALL HAVE 3 COATS OF PHENOLINE AS MFG. BY THE CARBOLINE COMPANY & APPLIED PER THEIR RECOMMENDATIONS
      (1) PRIME COAT, 300 ORANGE WITH MICA FILLER, 8 MILS THICK
      (2) SECOND COAT, 302 BLACK, 8 MILS THICK
      (3) FINISH COAT, 300 WHITE, 6 MILS THICK
   b. EXTERIOR SHALL HAVE PRIME COAT OF ZINC CHROMATE PRIMER, OR EQUAL
6. BOLT SPREADER BAR TO FLANGE WHEN LIFTING OR LOWERING TANK.

FIGURE B-6.

NO.	DATE	DESCRIPTION	LOC.	APPVD.	AEC APPVD.

REVISIONS

AR AREA

REACTOR BUILDING
SHIELD WATER TANK
PLAN, ELEVATIONS & DETAILS

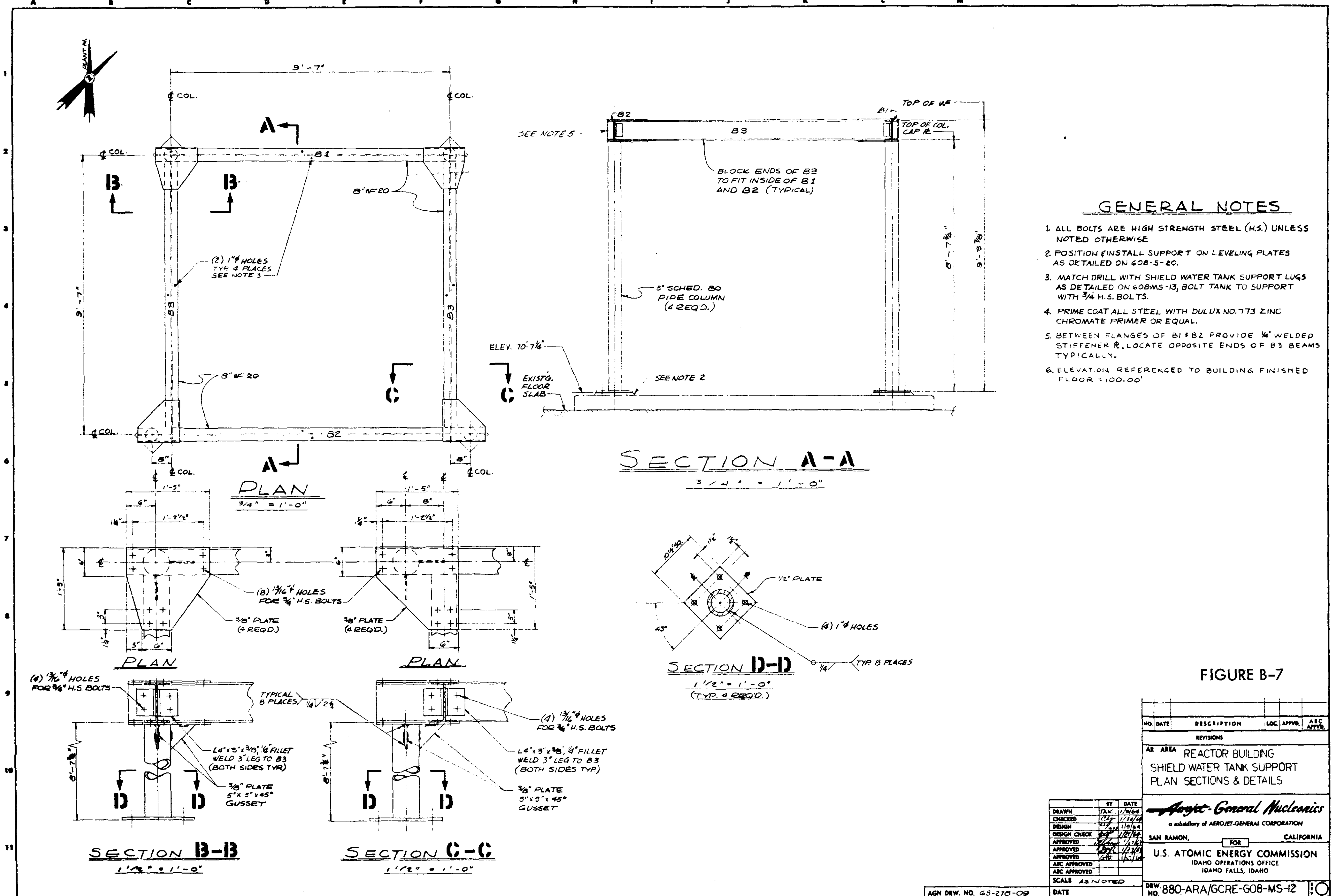
Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA
FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO

	BY	DATE
DRAWN	[illegible]	[illegible]
CHECKED	CLJ	[illegible]
DESIGN	[illegible]	[illegible]
DESIGN CHECK	[illegible]	[illegible]
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	[illegible]
AEC APPROVED		
AEC APPROVED		
SCALE AS NOTED		
DATE		

AGN DRW. NO. 63-278-10

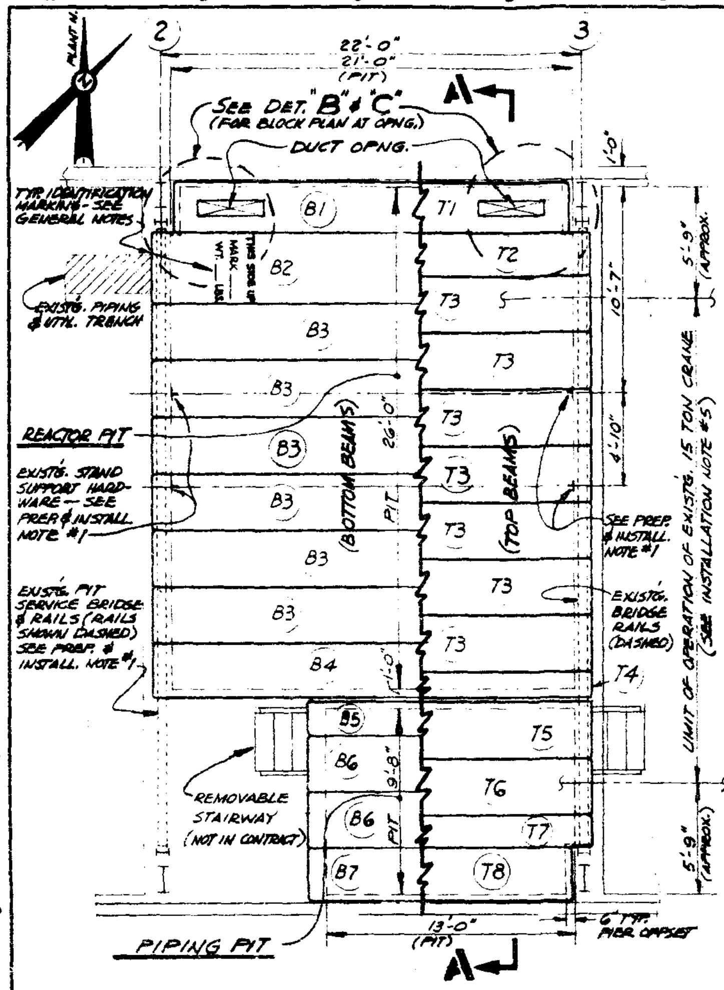
DRW. NO. 880-ARA/GCRE-608-MS-13

IDO-64-2540

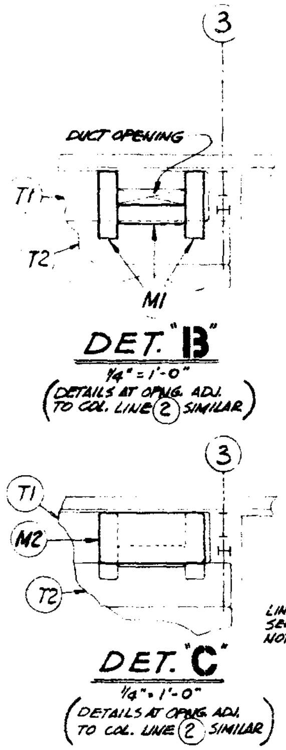


GENERAL NOTES
1. ALL BOLTS ARE HIGH STRENGTH STEEL (H.S.) UNLESS NOTED OTHERWISE
2. POSITION & INSTALL SUPPORT ON LEVELING PLATES AS DETAILED ON 608-S-20.
3. MATCH DRILL WITH SHIELD WATER TANK SUPPORT LUGS AS DETAILED ON 608MS-13, BOLT TANK TO SUPPORT WITH 3/4 H.S. BOLTS.
4. PRIME COAT ALL STEEL WITH DULUX NO. 773 ZINC CHROMATE PRIMER OR EQUAL.
5. BETWEEN FLANGES OF B1 & B2 PROVIDE 1/4" WELDED STIFFENER PL. LOCATE OPPOSITE ENDS OF B3 BEAMS TYPICALLY.
6. ELEVATION REFERENCED TO BUILDING FINISHED FLOOR = 100.00'
PLAN
3/4" = 1'-0"
SECTION A-A
3/4" = 1'-0"
SECTION D-D
1 1/2" = 1'-0"
(TYP. 4 REQ'D.)
SECTION B-B
1 1/2" = 1'-0"
SECTION C-C
1 1/2" = 1'-0"
BLOCK ENDS OF B3 TO FIT INSIDE OF B1 AND B2 (TYPICAL)
5" SCHED. 80 PIPE COLUMN (4 REQ'D.)
8" WF 20
(2) 1"Ø HOLES TYP 4 PLACES SEE NOTE 3
SEE NOTE 5
SEE NOTE 2
TOP OF WF
TOP OF COL. CAP PL
ELEV. 70'-7 1/4"
EXIST'G. FLOOR SLAB
(8) 13/16"Ø HOLES FOR 3/4" H.S. BOLTS
3/8" PLATE (4 REQ'D.)
1/2" PLATE
(4) 1"Ø HOLES
TYP. 8 PLACES
(4) 13/16"Ø HOLES FOR 3/4" H.S. BOLTS
L4"x3"x3/8", 1/4" FILLET WELD 3" LEG TO B3 (BOTH SIDES TYP.)
3/8" PLATE 5"x5"x45° GUSSET
AR AREA REACTOR BUILDING
SHIELD WATER TANK SUPPORT
PLAN SECTIONS & DETAILS
Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA
FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
SCALE AS NOTED
AGN DRW. NO. 63-278-09
DRW. NO. 880-ARA/GCRE-608-MS-12

FIGURE B-7

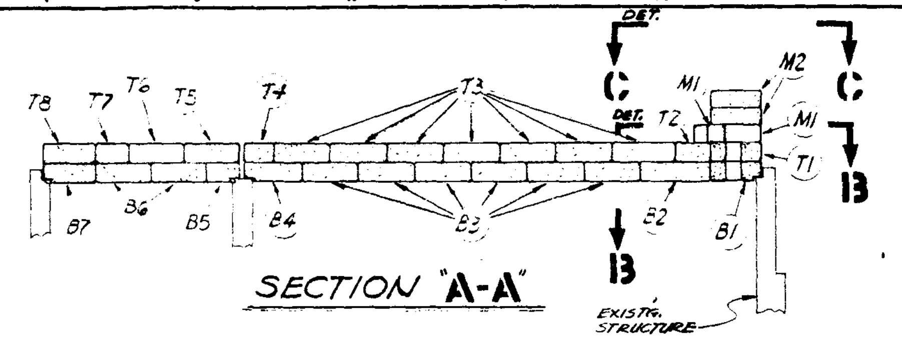


PLAN AT REACTOR & PIPING PITS
1/4" = 1'-0"

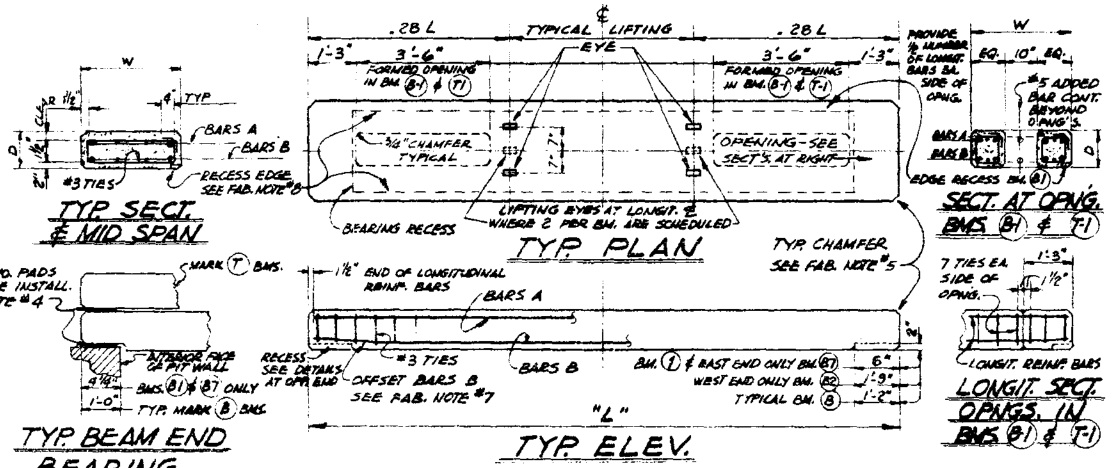


DET. "B"
1/4" = 1'-0"
(DETAILS AT OPNG. ADJ. TO COL. LINE (2) SIMILAR)

DET. "C"
1/4" = 1'-0"
(DETAILS AT OPNG. ADJ. TO COL. LINE (2) SIMILAR)



SECTION "A-A"

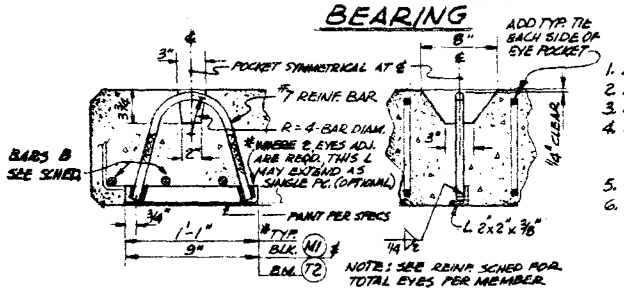


TYP. SECT. @ MID SPAN

TYP. BEAM END BEARING

TYP. PLAN

TYP. ELEV.



TYP. LIFTING EYE DETAIL
1 1/2" = 1'-0"

NOTE: 3"x8" EYE POCKET FOR USE WITH 3 TON (MIN.) CAP. STL. HOOK

GENERAL NOTES

1. DISCREPANCIES
PIT DIMENSIONS USED FOR BEAM DESIGN ARE INDICATED ON PLAN. HOWEVER, THE CONTRACTOR SHALL VERIFY THESE DIMENSIONS WITH THE CONDITIONS AT JOB SITE & SHALL NOTIFY THE CONTRACTING OFFICER OF ANY DISCREPANCIES. THE CONTRACTING OFFICER WILL SUBMIT & BE RESPONSIBLE FOR REVISED BEAM DIMENSIONS, TO CONFORM TO ACTUAL CONDITIONS

2. DESIGN LIVE LOADS
MAX. TOTAL LIVE LOADS TO BE:
1. BOTTOM BEAMS ONLY, IN PLACE 200 P.S.F.
2. BOTTOM & TOP BEAMS, IN PLACE 100 P.S.F.

3. CONCRETE
CONCRETE FOR SHIELDING BEAMS SHALL HAVE AN ULTIMATE COMPRESSIVE STRENGTH OF 3000/PSI AT 28 DAYS. SEE SPECIFICATION "C-2"

4. REINF. STEEL
REINF. STEEL SHALL BE INTERMEDIATE GRADE WITH AN ALLOWABLE TENSILE STRENGTH OF 20000/PSI. SEE SPECIFICATION

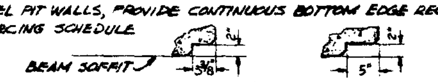
5. MARKING
INDIVIDUAL IDENTIFICATION MARKING TO INCLUDE THE FOLLOWING INFO & SHALL BE PAINTED ON EACH SHIELDING BEAM

INFO	LOCATION
THIS SIDE UP MARK ____ WT. ____ LBS.	TOP SURFACE AT EITHER END
MARK ____ WT. ____ LBS.	EACH END OF BEAM

PAINT BLOCK LETTERS 2" HIGH. SEE THE SPECIFICATION, DIVISION "P-1", PAINTING

FABRICATION NOTES

1. FOR ADDITIONAL INFORMATION SEE SPECIFICATION, DIVISION "C-2"
2. ALL REINF. BARS SHALL BE ROUND & DEFORMED IN ACCORDANCE WITH ASTM A-305-53T
3. LONGIT. BARS A & B EXTEND CONTINUOUS FOR ENTIRE BEAM LENGTH UNLESS NOTED OTHERWISE
4. ALL BEAMS TO HAVE STIRRUP(TIES) THRUOUT THEIR ENTIRE LENGTH. THE FIRST TIE TO BE 1 1/2" FROM END OF BEAM, THEN AS PER SCHEDULE. ALL TIES SHALL BE SECURELY WIRED TO LONGITUDINAL BARS
5. CHAMFER ALL EDGES & CORNERS AS SHOWN, 3/4"
6. RESULTING BEAMS SHALL BE TRUE RECTANGULAR PRISMS WITHOUT TWIST ALONG EITHER PRINCIPAL AXIS. SURFACES & CONFIGURATION SHALL NOT DEVIATE FROM THE FOLLOWING ACCEPTABLE TOLERANCES: WIDTH "W", & DEPTH "D", ± 1/8"; LENGTH "L", ± 1/4"; PLANE SURFACES, NO DEPRESSIONS OR IRREGULARITIES EXCEEDING ± 1/8" RELATIVE TO A 10 FT. STRAIGHTEDGE; TWIST, θ = ±0.5° MAX. — PRINCIPAL AXIS AT END SECTION
7. MAX. OFFSET IN HORIZONTAL (LONGITUDINAL) BARS TO BE 1/16"
8. WHERE MARK (B) BEAMS PARALLEL PIT WALLS, PROVIDE CONTINUOUS BOTTOM EDGE RECESSES SEE REMARKS IN THE REINFORCING SCHEDULE



TYPE "X" TYPE "Y"

PREPARATION & INSTALLATION NOTES

1. PRIOR TO INSTALLATION OF SHIELDING BEAMS, CERTAIN EXISTG. ITEMS SHALL BE REMOVED & STORED OR WASTED AS DIRECTED BY THE CONTRACTING OFFICER. THESE ITEMS ARE:
(a) COMPLETE PIT SERVICE BRIDGE INCLUDING HANDRAILS & ACCESSORY ITEMS
(b) APPROX. 60 FT. OF FLOOR MOUNTED BRIDGE TRACK SYSTEM, INCLUDING RAILS, BEAMS, IMBEDDED ANCHORS (BURN FLUSH WITH FLOOR) & GROUT
(c) FOUR (4) STAINLESS STEEL STAND SUPPORTS ATTACHED TO THE PIT WALLS
2. THE INTENT OF THIS WORK IS TO PROVIDE A RADIATION SHIELD USING THE BEAMS & BLOCKS DETAILED. ALL BEAMS IN FINAL POSITION ARE TO ABUT ADJACENT BEAMS OR PIT WALLS WITH THE ABSOLUTE MAXIMUM JOINT SPACE EQUALING 1/4"
3. ALL CONCRETE SURFACES, ON WHICH BEAMS WILL BEAR, SHALL BE SMOOTH AND FREE FROM PROTRUSIONS OR LOCAL IRREGULARITIES GREATER THAN 1/8" RELATIVE TO A 10 FT. STRAIGHTEDGE
4. AT ENDS OF MARK (T) BEAMS, THE END BEARING AREAS OF MARK (B) BEAMS & FOR FULL BEARING AREA OF MARK (M) BLOCKS, PROVIDE 1/8" THICK LINOLEUM BEARING PADS. HOLD PAD DIMENSIONS 1" SHORT OF BEAM ENDS & INTERIOR FACE OF PIT WALL, TYPICALLY. FOR BEAMS (B-1) & (B-7) ONLY, PADS SHALL BE SIZED TO COVER ENTIRE END BEARING AREA
5. CONTRACTOR MAY USE EXISTING 15 TON RATED CAB BRIDGE CRANE FOR PLACEMENT OF SHIELDING BEAMS, HOWEVER, IT SHOULD BE NOTED THAT THE AREA AT THE NORTH END OF THE REACTOR PIT & THE SOUTH END OF THE PIPING PIT ARE NOT WITHIN THE LIMITS OF TRAVEL OF THE LIFTING HOOK ℄. SPECIAL RIGGING PROCEDURES TO SET BEAMS IN THIS REGION SHALL BE APPROVED BY THE CONTRACTING OFFICER. ALL BEAMS ARE DESIGNED TO BE LIFTED WITHOUT SPREADER, PROVIDING SLING CABLE OR CHAIN DOES NOT FORM A LIFTING ANGLE LESS THAN 60° WITH THE HORIZONTAL

REINFORCING SCHEDULE

No. REQD.	MARK	W	D	L	LONGIT. REINF. BARS A	LONGIT. REINF. BARS B	STIRRUPS (TIES) SPACING FROM EACH END	WT. (LBS) EACH BEAM	LIFTING EYES (TOTAL)	REMARKS
1	B-1	2'-7 1/4"	14"	20'-8 1/2"	6-#6	4-#8 / 2-#7	2@6", 1@5", 6@6 1/2", 1@5", 2@6", 12" MAX.	10200	4	SEE SPECIAL SECTION ABOVE FOR OPNGS. IN BM. TYPE "X" RECESS AT NORTH EDGE PER FAB. NOTE #8
1	B-2	3'-7 3/4"	14"	23'-0"	7-#6	B1 6-#8* / B2 2-#8	30@4", 12@6", REMAINDER @ 8" MAX.	15000	4	* B1 BARS FOR ENTIRE SPAN. B2 BARS SPAN 12'-0" AT BM. ℄
6	B-3	2'-10 3/4"	14"	23'-0"	6-#6	B1 4-#8* / B2 2-#8	14@6", REMAINDER @ 12" MAX.	12000	4	* B1 BARS FOR ENTIRE SPAN. B2 BARS SPAN 14'-0" AT BM. ℄
1	B-4	2'-10 3/4"	14"	23'-0"	6-#6	B1 4-#8* / B2 2-#8	SAME AS B-3	12000	4	TYPE "Y" RECESS AT SOUTH EDGE PER FAB. NOTE #8. * B1 BARS FULL SPAN - B2 BARS SPAN 14'-0" AT BM. ℄
1	B-5	1'-8 3/4"	14"	15'-0"	3-#6	4-#6	5@6", REMAINDER AT 12" MAX.	4800	2	TYPE "Y" RECESS AT NORTH EDGE PER FAB. NOTE #8
2	B-6	2'-10 3/4"	14"	15'-0"	4-#5	B1 2-#7* / B2 2-#6*	5@6" REMAINDER @ 12" MAX.	7800	4	* B1 BARS FOR ENTIRE SPAN. B2 BARS SPAN 9'-0" AT BM. ℄
1	B-7	2'-9 1/4"	14"	13'-10 1/4"	4-#5	4-#6	5@6", REMAINDER @ 12" MAX.	6800	4	TYPE "X" RECESS AT SOUTH EDGE PER FAB. NOTE #8
1	T-1	2'-7 1/4"	12"	20'-8 1/2"	6-#6	3-#8 / 2-#7	3@6", 1@5", 6@6 1/2", 1@5", 2@6", 10" MAX.	8500	4	SEE SPECIAL SECTION ABOVE FOR OPNGS. IN BM.
1	T-2	2'-2 1/4"	12"	23'-0"	5-#6	B1 3-#8* / B2 2-#8	SAME AS T-3	7800	4	* B1 BARS FOR ENTIRE SPAN. B2 BARS SPAN 14'-0" AT BM. ℄
7	T-3	2'-10 3/4"	12"	23'-0"	6-#6	B1 4-#8* / B2 2-#8	14@6", REMAINDER @ 10" MAX.	10300	4	* B1 BARS FOR ENTIRE SPAN. B2 BARS SPAN 14'-0" AT BM. ℄
1	T-4	1'-5 1/4"	12"	23'-0"	3-#6	2-#8 / 2-#7	SAME AS T-3	5300	2	SAME AS T-3
1	T-5	2'-10 3/4"	12"	15'-0"	4-#5	B1 2-#7* / B2 2-#6	5@6", REMAINDER @ 12" MAX.	6800	4	SAME AS T-6
1	T-6	2'-10 3/4"	12"	15'-0"	4-#5	B1 2-#7* / B2 2-#6	5@6", REMAINDER @ 12" MAX.	6800	4	* B1 BARS FOR ENTIRE SPAN. B2 BARS SPAN 9'-0" AT BM. ℄
1	T-7	1'-8 3/4"	12"	15'-0"	3-#6	4-#6	5@6", REMAINDER @ 12" MAX.	4100	2	
1	T-8	2'-9 1/4"	12"	13'-10 1/4"	4-#5	4-#6	5@6", REMAINDER @ 12" MAX.	6000	4	
6	M-1	1'-0"	12"	3'-6"	2-#4	2-#4	ENTIRE SPAN @ 10" MAX.	600	2	
4	M-2	2'-7 7/8"	12"	5'-6"	5-#4	5-#4	ENTIRE SPAN @ 10" MAX.	2300	2	

FIGURE B-8.

NO.	DATE	DESCRIPTION	LOC.	APPVD.	AEC APPVD.

REVISIONS

AREA

REACTOR BUILDING RADIATION SHIELDING PLAN, SECTIONS & DETAILS

Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA

FOR

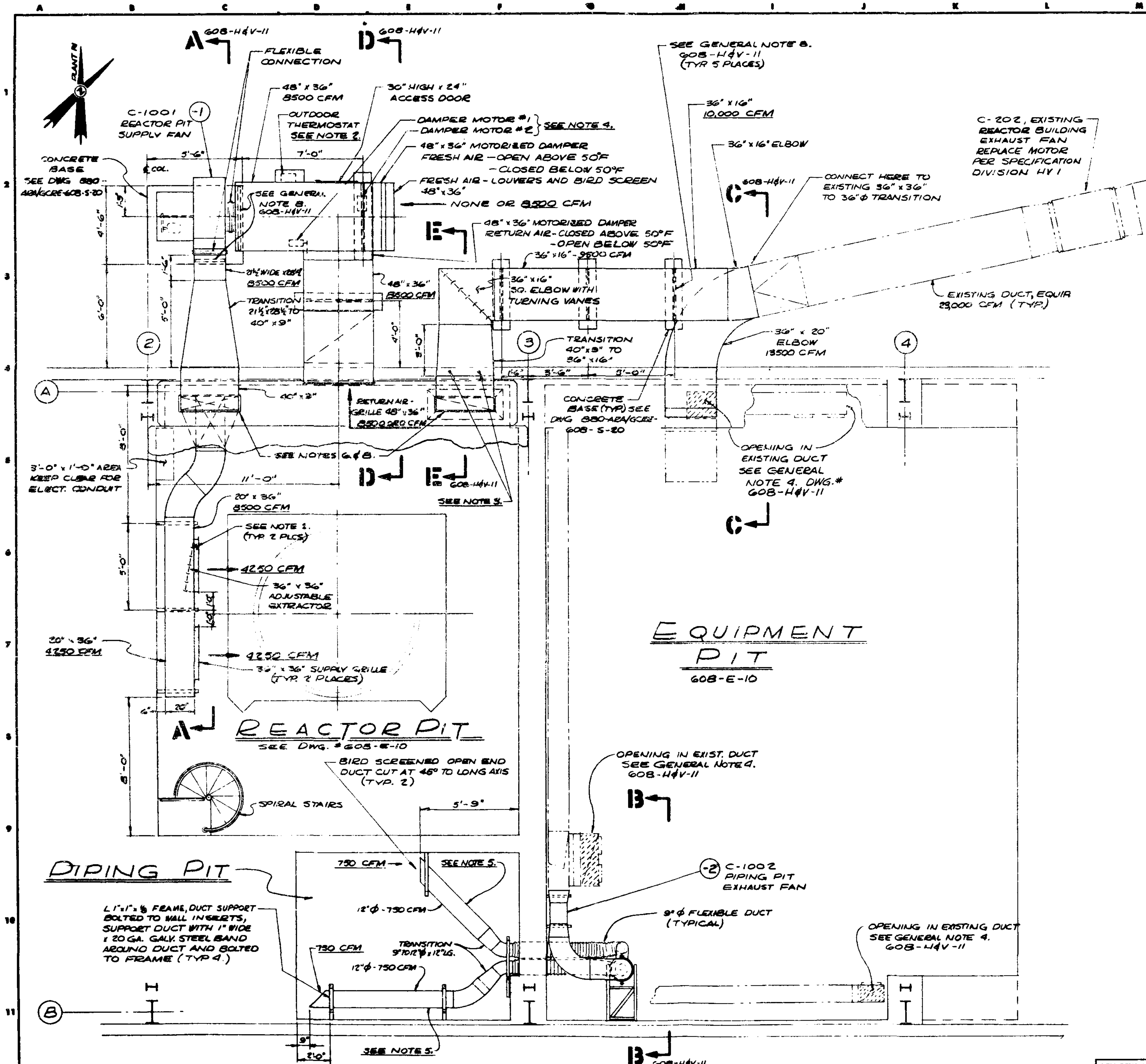
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO

	BY	DATE
DRAWN		1-17-64
CHECKED		1-21-64
DESIGN		1-21-64
DESIGN CHECK		
APPROVED		
APPROVED		
APPROVED		
AEC APPROVED		
AEC APPROVED		
SCALE	AS NOTED	
DATE		

AGN DRW. NO. 63-278-26

DRW. NO. 880-ARA/GCRE-608-S-21 | 0

11.1-64-2542



NOTES

1. 36" x 36" DUCT 6" LONG FORMED WITH SM FLANGE EACH END. SM SCREW ON 3' CENTERS TO MAIN DUCT.
2. OUTDOOR THERMOSTAT SHALL BE INSTALLED IN A LOUVERED WEATHERPROOF ENCLOSURE WHICH SHALL ALSO HOUSE A DISCONNECT SWITCH AND BE MOUNTED ON FAN SUCTION PLENUM WALL, WITH FRONT PANEL HINGED AND WITH CATCH LOCK.
3. MOUNT 2 PLUGGED NIPPLES, ½"-20NF x 1" LONG THREAD, IN TOP OF TRANSITION BETWEEN WALL AND CONCRETE SHIELDING BEAMS. LABEL SAME T-4001 & T-4002 (FOR PROJECT SUPPLIED AND INSTALLED THERMOSTATS).
4. MOUNT DAMPER MOTOR INSIDE AND CONNECT TO DAMPER WITH LINKAGE.
5. ROUTE DUCTS TO CLEAR ALL PIPES AND OBSTRUCTIONS.
6. TOP SHIELDING BLOCKS LEFT OUT OF VIEW FOR CLARITY. SEE DWG. 880-ARA/GCRE-608-S-21 FOR BLOCK ARRANGEMENT.
7. FOR GENERAL NOTES SEE DWG. 880-ARA/GCRE-608-H&V-11
8. FOR EXACT SHIELDING BLOCK PENETRATION LOCATIONS SEE DWG. 880-ARA/GCRE-608-S-21.

EQUIPMENT LIST

ITEM #	AMT.	DESCRIPTION
-1	1	REACTOR PIT SUPPLY FAN # C-1001 WHEEL DIA-27", CAPACITY-8500 CFM, ACTUAL STATIC PRESSURE -1.25" W.C., ELEVATION-4800 FT., EQUIV. STATIC PRESS. FOR FAN SELECTION -1.55" W.C., FAN WHEEL SPEED -907 RPM, BELT DRIVE FOR MIN.RPM = 805±25 FOR MAX. RPM = 1000±25, MOTOR-5HP, 1750 RPM, 440V-3Ø-60~, FAN DISCHARGE-BOTTOM HORIZONTAL, ROTATION-COUNTER CLOCKWISE, MODEL & SIZE - #1027 VENTILATING SET, MFG. - WESTINGHOUSE, OR APPROVED EQUAL.
-2	1	PIPING PIT EXHAUST FAN # C-1002 DUCT DIA. -12", CAPACITY-1490 CFM & STATIC PRESS. - ¾" (UNDER STD. CONDITIONS), MOTOR-½HP, 3400 RPM, 440V-3Ø-60~, MODEL & SIZE - 70123 MFG. - ILG, OR APPROVED

FIGURE B-9

NO.	DATE	DESCRIPTION	LOC.	APPVR.	AEC APPVR.
		REVISIONS			

AR AREA
REACTOR BUILDING VENTILATION MODIFICATION PLAN

	BY	DATE
DRAWN	TAK	1/14/64
CHECKED	FHS	1/23/64
DESIGN	FHS	1/16/64
DESIGN CHECK		
APPROVED		
APPROVED		
APPROVED		
AEC APPROVED		
AEC APPROVED		
SCALE	3/8" = 1'-0"	
DATE		

Aerojet-General Nucleonics
a subsidiary of AEROJET-GENERAL CORPORATION
SAN RAMON, CALIFORNIA

FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO

AGN DRW. NO. 63-278-05

DWG. NO. 880-ARA/GCRE-608-H&V-10 0

11.1-64-2543

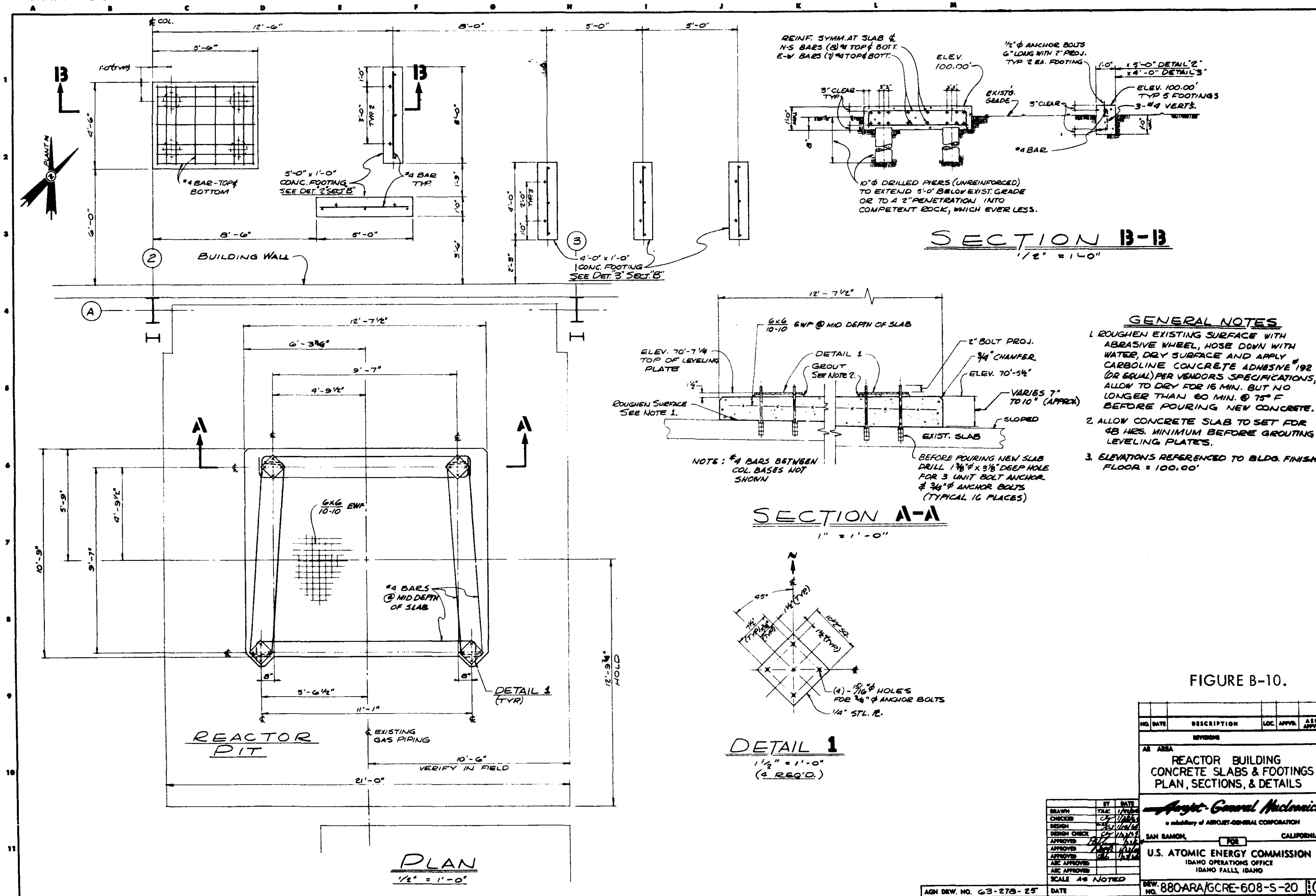


FIGURE B-10.

11.1-64-2544

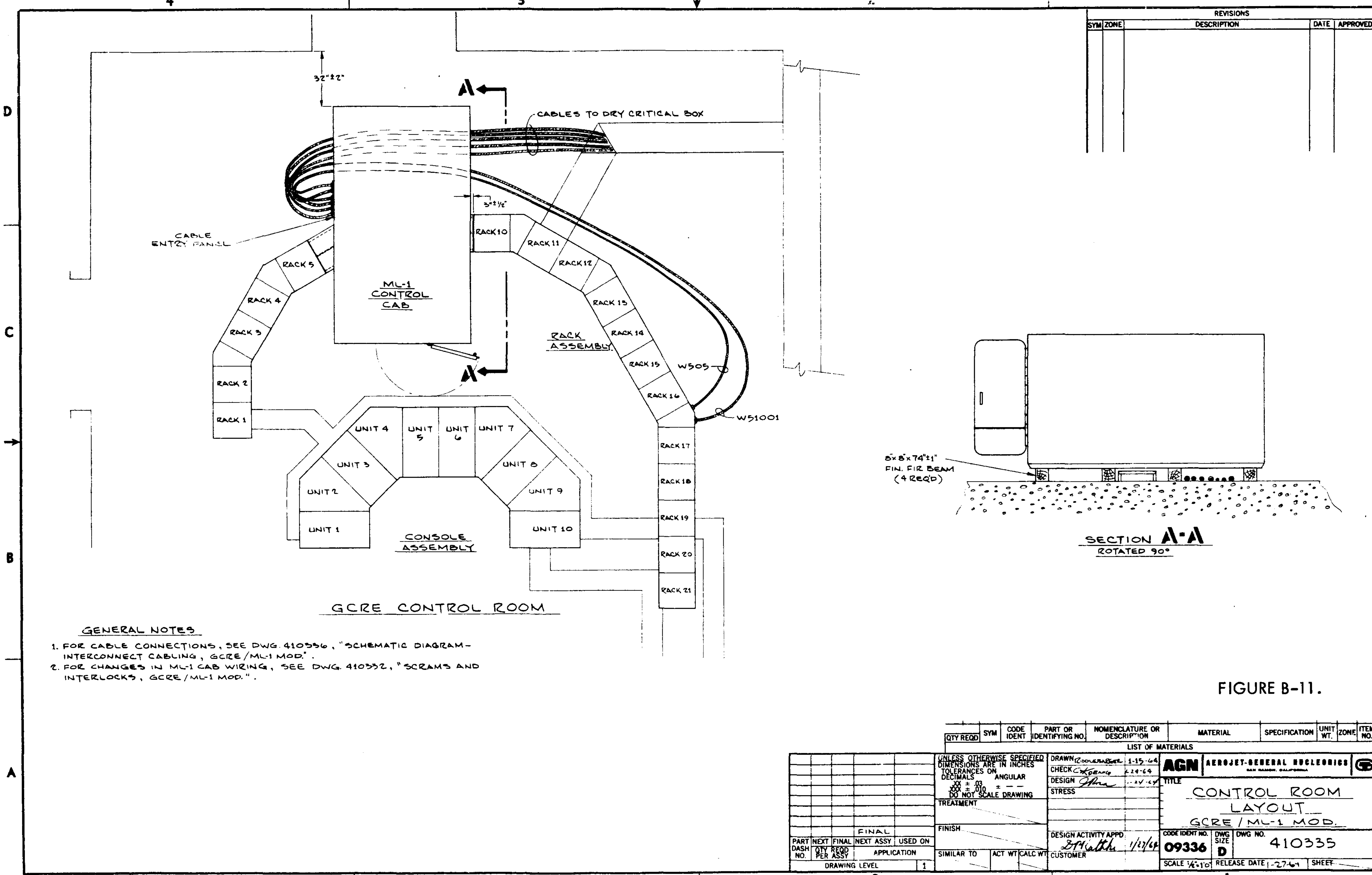


FIGURE B-11.

11.1-64-2545

APPENDIX C

DOSAGE CALCULATIONS

This appendix presents the major calculations performed to determine the dosage rates associated with the ML-1/GCRE maximum credible accident. Repetitive calculations are not shown.

A. GAMMA SOURCE STRENGTH

The gross gamma source term from instantaneous fission is given by the expression:[(C-1)*]

$$S(t) = 2.78t^{-1.23} - 2.41t^{-1.45} \tag{1}$$

where:

S is the gamma source term in Mev/sec/fission, and

t is the decay time in seconds

The gross gamma source term from a reactor system with a finite operating time is given by:

$$S_o(T_o, T_s) = \int_o^{T_o} S(\tau)(dt) \tag{2}$$

where:

S_o is the gamma source term in Mev/fission

T_o is the operating time, sec.

T_s is the time after shutdown, sec, and

τ has been set equal to the time parameter of Eq. 1 so that

$$\tau = T_o + T_s - t \tag{3}$$

*References are listed at the end of this appendix.

Substitution of Eq. 1 and Eq. 3 into the integral of Eq. 2 yields

$$\int_0^{T_o} S(\tau)dt = 2.78\int_0^{T_o} (T_o + T_s - t)^{-1.23}dt - 2.41\int_0^{T_o} (T_o + T_s - 1)^{-1.44}dt$$

$$= \left[12.05\ T_s^{-0.23} - (T_o + T_s)^{-0.23}\right] - 5.35\left[T_s^{-0.45} - (T_o + T_s)^{-0.45}\right] \tag{4}$$

Substitution of the term in Eq. 4 into Eq. 2 and introducing the equivalence between reactor power and fissions/sec yields the following expression for the operating gamma source term:

$$S_o\ (T_o, T_s) = P\left(3.76\left[T_s^{-0.23} - (T_o + T_s)^{-0.23}\right] - 1.67\left[T_s^{-0.45} - (T_o + T_s)^{-0.45}\right]\right) \times 10^{17} \tag{5}$$

where P is the reactor operating power in Mw.

For the ML-1/GCRE, P = 2.2 Mw and Eq. 5 becomes:

$$S_o\ (T_o, T_s) = \left(8.27\left[T_s^{-0.23} - (T_o + T_s)^{-0.23}\right] - 3.67\left[T_s^{-0.45} - (T_o + T_s)^{-0.45}\right]\right) \times 10^{17} \tag{5a}$$

The excursion source term is derived from Eq. 1 and is given by:

$$S_E(T_s) = E(8.67\ T_s^{-1.23} - 7.52\ T_s^{-1.45}) \times 10^{16} \tag{6}$$

where E is the excursion energy in Mw-sec.

For the ML-1/GCRE, E = 125 Mw-sec(C-2) so that Eq. 6 becomes:

$$S_E(T_s = (1.08\ T_s^{-1.23} - 0.94\ T_s^{-1.45}) \times 10^{19} \tag{6a}$$

The total gamma source term for the ML-1/GCRE maximum credible accident is the sum of Eq. 5 and 6, or:

$$S_T(T_o, T_s) = S_o(T_o, T_s) + S_E(T_s) \tag{7}$$

Assuming an operating time of 3.6×10^7 sec[C-2], the total gamma source for the ML-1/GCRE maximum credible accident becomes:

$$S_T(T_o, T_s) = S_o(3.6 \times 10^7, T_s) + S_E(T_s) \qquad (7a)$$

This analysis considers the dosages in the first ten minutes after the excursion so that direct comparison may be drawn with the dosage values given in Reference C-2. Consequently, the time interval of interest is between $T_s = 0$ and $T_s = 600$ sec.

Integrating Eq. 5a:

$$\int S_o(T_o, T_s)dt = \left(\frac{8.27}{0.77} \left[T_s^{0.77} - (T_o + T_s)^{0.77} \right] - \frac{3.67}{0.55} \left[T_s^{0.55} - (T_o + T_s)^{0.55} \right] \right) \times 10^{17} \Bigg|_o^{600} \qquad (5b)$$

Since $T_o = 3.6 \times 10^7$ and $T_s = 0$ to 600; $T_o \gg T_s$ and Eq. 5b can be written as:

$$\int S_o(T_o, T_s)dt = \left[10.75 \, (T_s^{0.77} - T_o^{0.77} - 6.67 \, (T_s^{0.55} - T_o^{0.55}) \right] \times 10^{17} \Bigg|_o^{600} = \left[10.75 \, (600^{0.77}) - 6.67 \, (600^{0.55}) \right] \times 10^{17} \qquad (5c)$$

or

$$\int S_o(T_s) = 1.245 \times 10^{20} \text{ Mev} \qquad (5d)$$

Similarly, the excursion source term may be determined by integrating Eq. 6a from 1 sec to 600 sec as follows:

$$\int_1^{T_s} S_E(T_s)dt = \left(\frac{1.08T_s^{-0.23}}{-0.23} - \frac{0.94T_s^{-0.45}}{-0.45} \right) \times 10^{19} \Bigg|_1^{600} \qquad (6b)$$

$$= \left(\frac{2.09}{600^{0.45}} - \frac{2.09}{1} - \frac{4.7}{600^{0.23}} + \frac{4.7}{1} \right) \times 10^{19}$$

or:

$$\int S_E(T_s) = 1.65 \times 10^{19} \text{ Mev} \tag{6c}$$

The total gamma source term is, from Eq. 7,

$$S_T(T_o, T_s) = 1.245 \times 10^{20} + 1.65 \times 10^{19} = 1.41 \times 10^{20} \text{ Mev/10 min} \tag{7b}$$

In order that these calculations may be directly comparable with those of Reference C-2, the release fractions are assumed to be the same as those used in that reference. The activity release associated with the ML-1 maximum credible accident results in a total gamma source which comprises about 50% of the rare gases and about 25% of the halogens. Since these constituents make up about 20% of the total gamma source, about 8% of that source is released into the radioactive cloud. Thus, for this analysis, the gamma source for dosage calculations is:

$$S_T = 1.41 \times 10^{20} \text{ (from Eq. 7b)} \times 0.08 = 1.13 \times 10^{19} \text{ Mev/10 min} \tag{7c}$$

B. DOSE THROUGH CONTROL ROOM WALL

The entire gamma dose is assumed to be a uniformly distributed volume source (radioactive cloud) in the reactor test building. The dose point under consideration is separated from the source by 24 inches of concrete and some distance as shown in Figure C-1. The energy flux from the volume source at the dose point is given by[(C-3)]:

$$\phi = \sum_{i=1}^{2} \frac{B_i S_i}{2 \mu_i} \left[E_2(b_1) - E_2(b_3) \right] \tag{8}$$

In this case, the expression may be rewritten as:

$$\phi = \frac{B S_v}{2 \mu_s} \left[E_2(b_1) - E_2(b_3) \right] \tag{8a}$$

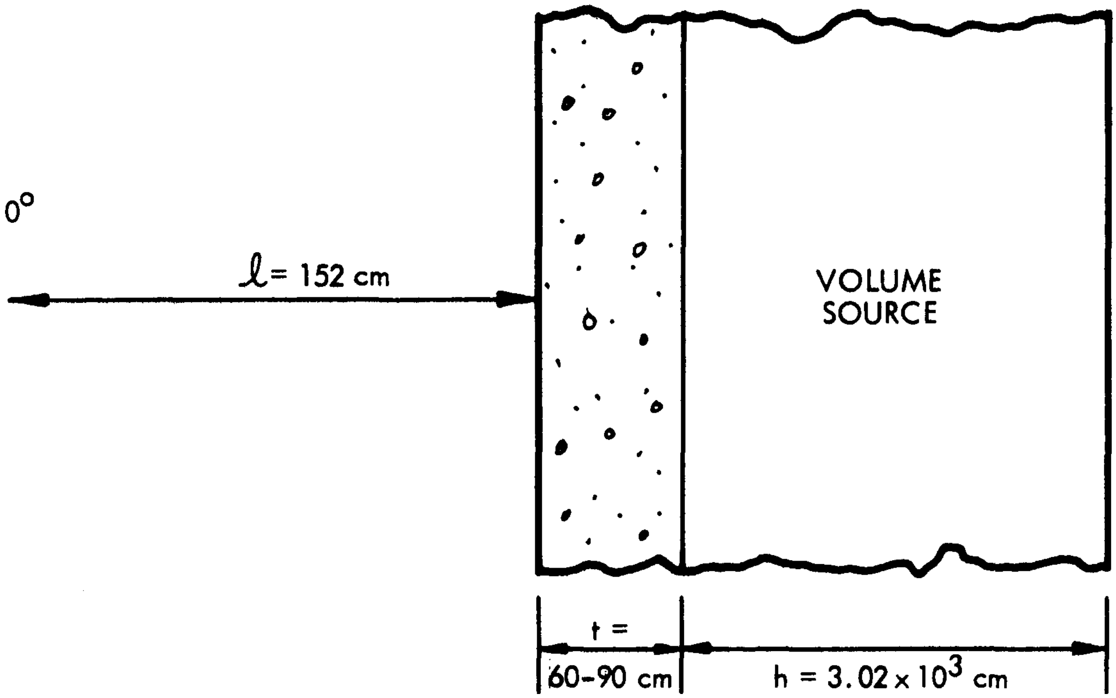


FIGURE C-1. GEOMETRY FOR CALCULATING DOSE THROUGH CONTROL ROOM WALL

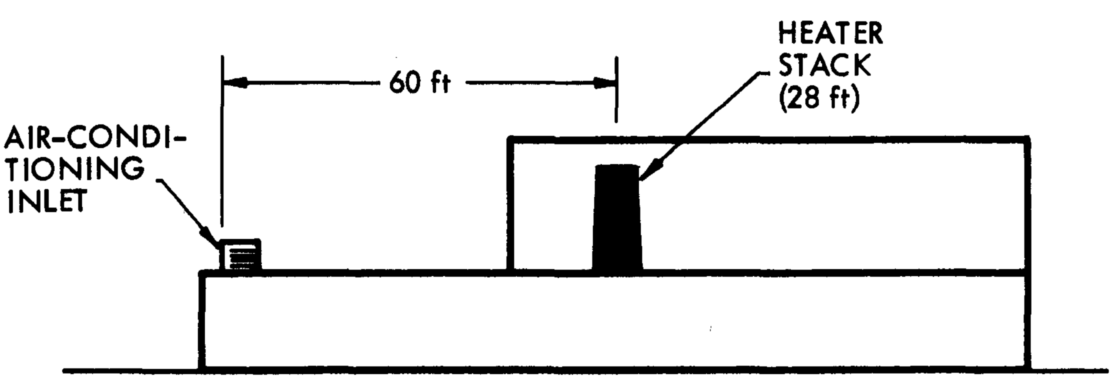


FIGURE C-2. GEOMETRY FOR CALCULATING INHALATION DOSE IN CONTROL ROOM FROM HEATER RUPTURE where ϕ is the energy flux at the dose point in Mev/sec

B is the build-up factor in the concrete

S_v is the gamma source per unit volume of the volume source

$$\left(\frac{1.13 \times 10^{19}}{4.3 \times 10^{9}}\right), \text{ Mev/sec-cm}^3$$

μ_s is the gamma absorption coefficient in air in cm^{-1}

$E_2(b)$ is an exponential function from Reference C-3, and

b_1 and b_3 are as defined below:

$b_1 = \mu_i X_i$ or, in this case, $= \mu_c t$ where

μ_c is the gamma absorption coefficient in concrete in cm^{-1}, and

t is the thickness of the concrete in cm.

$b_3 = \mu_i X_i + \mu_s h$ or, in this case,

$= \mu_c t + \mu_s h$

where μ_c, μ_s and t are as defined above and h is the distance from the concrete to the center of the cloud as shown in Figure C-1 (3.02×10^3 cm). The attentuation by air from the concrete to the dose point (152 cm) is negligible and is not included in the calculation.

In order that the results of these calculations may be directly comparable with those presented in Reference C-2, it is assumed that the total gamma source term is made up of two equal (in terms of total activity) constituents with decay energies of 0.7 and 1.22 Mev respectively. This requires that Eq. 8a be solved twice using the following values:

Item	Case 1 (0.7 Mev)	Case 2 (1.22 Mev)
μ_s, cm^{-1}	8.15×10^{-5}	6.2×10^{-5}
μ_c, cm^{-1}	0.172	0.133
t, cm	60.9	60.9
h, cm	3.02×10^3	3.02×10^3
$b_1 = \mu_c t$	10.47	8.1
$b_3 = \mu_c t + \mu_s h$	10.72	8.29
B	25	14
S_v, Mev/10 Min-cm^3	1.308×10^9	1.308×10^9

The Case 1 solution becomes:

$$\phi_1 = \frac{(25)(1.308 \times 10^9)}{(2)(8.15 \times 10^{-5})} (2.25 \times 10^{-6}) - (1.72 \times 10^{-6})$$

$$= 10.63 \times 10^7 \text{ Mev/10 min-cm}^2$$

The Case 2 solution becomes:

$$\phi_2 = \frac{(14)(1.308 \times 10^9)}{(2)(6.2 \times 10^{-5})} (3.00 \times 10^{-5}) - (2.45 \times 10^{-5})$$

$$= 8.12 \times 10^8 \text{ Mev/10 min-cm}^2$$

The dose rate conversion factors for the energies under consideration are:

for 0.7 Mev, 4.95×10^2 Mev/cm^{2}sec = 1.0 mr/hr
1.22 Mev, 5.40×10^2 Mev/cm^{2}sec = 1.0 mr/hr

The dose observed at the dose point may be calculated as follows:

$$D_1 = \frac{(10.63 \times 10^7)}{(600)(4.95 \times 10^2(10^3)(6)} = 0.0597 \text{ Rem/10 min}$$

$$D_2 = \frac{8.12 \times 10^8}{(600)(5.40 \times 10^2)(10^3)(6)} = 0.418 \text{ Rem/10 min}$$

The total dose is then

$$D = D_1 + D_2 = 0.0597 + 0.418 = 0.478 \text{ Rem/10 min}$$

The dose in the counting room, taking into account the distance between the control room and the counting room and the intervening walls, would be about 0.05 Rem/10 minutes.

C. DOSAGE IN CONTROL ROOM FROM EXHAUSTED RADIOACTIVE CLOUD

1. General

The rare gases and halogens constitute the major portion of the gamma activity released by the ML-1/GCRE maximum credible accident. It is assumed that 50% of the total source of rare gases and 25% of the total source of halogens are released. Based on data presented in Reference C-2, this release will amount to about

$$\frac{2.2}{3.3}(1.12 \times 10^6) \simeq 7.5 \times 10^5 \text{ curies,}$$

when the lower operating level of the ML-1/GCRE is taken into account. This release corresponds, effectively, to about 40% of the volatile core inventory or about 8% of the total core gamma source. These assumptions are consistent with those used in Reference C-2.

During the excursion associated with the maximum credible accident, about 4×10^4 curies of volatile fission products are generated per Mw-sec of excursion energy. For the 125 Mw-sec excursion discussed in Reference C-2, and again assuming about 40% release, the total excursion activity released is

$$(125)(4 \times 10^4)(0.4) \simeq 2.0 \times 10^6 \text{ curies}$$

The total release is then:

$$(7.5 \times 10^5) + (2 \times 10^6) = 2.75 \times 10^6 \text{ curies}$$

2. Cloud Exhausted from 28 Foot (Heater) Stack

Assuming that the distance between the heater stack and the occupant in the control room is 60 ft (as shown in Figure C-2) and assuming the worst possible meteorological condition (inversion) described as follows:

$n = 0.55$

$C = 0.14$

$\bar{u} = 1.34$ m/sec

$h = 28$ ft $= 8.54$ meters,

the unshielded dose at the control room can be taken from the dose nomograph of Reference C-4 for an infinite cloud travel path over the receptor. In the case of the ML-1/GCRE release, the situation is more truly represented by a 1/2 infinite line (cloud path) and the presence of 12 inches of concrete shielding further reduces the dose.

By reference to the nomograph, the unshielded, uncorrected dose from the excursion source is 32 Rem while that from the operating source is 30 Rem. Taking into consideration the 1/2 infinite path and the concrete, these values are reduced to about 1.6 and 1.5 Rem respectively. The total dose to an occupant of the control room from the radioactive cloud resulting from the ML-1/GCRE maximum credible accident if released through the heater stack is then about 3.1 Rem under inversion conditions.

3. Cloud Exhausted from 150 Foot (Ventilation System) Stack

The release associated with the ML-1/GCRE maximum credible accident is assumed to be dispersed in the volume of the reactor test building. The source is (from Section C-1 above) 2.75×10^6 curies and the volume of the test building is about 152,000 ft^3. The exhaust blower will transfer about 22,000 ft^3/min up the stack which will be replaced by non-radioactive air.

Assuming perfect mixing in the building, the rate of release of activity from the stack can be expressed as:

$$\frac{dQ}{dt} = kQ_o \left(\frac{t_o}{t}\right)^n e^{-k(t-t_o)}, \text{ curies/sec} \qquad (9)$$

where Q is the amount of activity released, curies

t is the time after the initial time t_o, sec

k = R/V where R is the stack flow rate, ft^3/min, and

V is the building volume, ft^3

n is the decay parameter for a given fission product (n = 0.2 for long-operation fission products; n = 1.2 for excursion fission products)

The dose from the radioactive release is given by:

$$D_T = \frac{f_r \, D \, k \, Q_o \, t_o^{\,n}}{\bar{u}} \int_{t_o}^{T} \frac{e^{-k(t-t)_o}}{t^n} \, dt, \text{ Rem} \qquad (10)$$

where D_T is the total integrated dose, Rem

f_r is the release fraction

D is the dose from nomograph (Reference C-4), Rem/unit source

$\bar{u}$ is the wind speed, m/sec

T is the time of exposure of the control room occupant, for this analysis 10 min,

and the other factors are as defined for Eq. 9.

Since this analysis treats the exposure to control room occupants in the first ten minutes following the release, it is possible to approximate the value of the integrated term in Eq. 10 by calculating an average release of material from the stack as follows:

$$Q = \frac{Q_o}{T - t_o} \int_{t_o}^{T} e^{-k(t-t_o)}\, dt \tag{11}$$

$$= \frac{Q_o C^{kt_o}}{k\ (T-t_o)} \left(e^{-kt_o} - e^{-kt}\right)$$

$$= 0.583\ Q_o$$

for $k = \frac{R}{V} = 0.00283\ \sec^{-1}$,

$t_o = 1$ sec, and

$T = 600$ sec.

Substituting in Eq. 9, the approximate release rate becomes:

$$\frac{dQ}{dt} = 0.538\ k\ Q_o \left(\frac{t_o}{t}\right)^n \tag{12}$$

and the dose expressed by Eq. 10 becomes:

$$\frac{dD}{dt} = \frac{f_r}{\overline{u}} \left(\frac{D}{Q}\right) \frac{dQ}{dt} \tag{13}$$

or,

$$D_T = \frac{(0.583)\ k\ Q_o\ f_r\ t_o^{\ n}}{\overline{u}} \left(\frac{D}{Q}\right) \int_{t_o}^{T} t^{-n}\, dt, \text{ Mev} \tag{14}$$

Since t_o is always 1 sec,

$$D_T = \frac{(0.583)\ k\ Q_o\ f_r}{\overline{u}\ (n-1)} \left(1 - T^{-n+1}\right), \text{ Mev} \tag{14a}$$

Equation 14a is solved twice to reflect the difference in decay constants (n) between the operating and excursion sources:

$$D_o = \frac{(0.583)(2.83 \times 10^{-3})(7.5 \times 10^5)(0.085)}{(1.34)(0.2 - 1.0)} \left(1 - 600^{-1.2}\right)$$

and including factors of 1/2 for the half-infinite line cloud path and 1/10 for the control room shielding,

$$D_o = 0.47 \text{ mrem in 10 minutes.}$$

Similarly,

$$D_E = \frac{(0.583)(2.83 \times 10^{-3})(2.0 \times 10^{6})(0.085)}{(1.34)(1.2 - 1.0)} (1 - 600^{-2.2})$$

and with the correction factors discussed above,

$$D_E = 65.4 \text{ mrem in 10 minutes.}$$

The total dose is then:

$$\begin{aligned} D_T &= D_o + D_E \\ &= 0.47 + 65.4 \\ &= 65.87 \text{ mrem in 10 minutes.} \end{aligned}$$

D. INHALATION AND WHOLE BODY DOSES FROM COOLANT DUCT RUPTURE IN HEATER AND SUBSEQUENT INTRODUCTION OF ACTIVITY INTO CONTROL ROOM BY AIR-CONDITIONING SYSTEM

The maximum inhalation dose associated with the ML-1/GCRE maximum credible accident occurs in the event the coolant duct rupture associated with the accident occurs in the gas-fired heater. In this case, the released volatile fission products are exhausted from the 28-foot heater stack and, if meteorological conditions are correct, are carried directly to the intake of the control building air-conditioning system. If it is assumed that the procedure which specifies shutdown of the air conditioner in the event of an accident is not implemented, significant radioactivity will be introduced into the control room. This calculation develops the inhalation and whole body doses to an occupant of the control room for the situation described above.

1. Inhalation Dose

The rate of introduction of radioactive gas into the control room from the cloud discharged from the heater stack may be expressed by:

$$RC_o = (0.736) \frac{(0.006 A_i)}{39.6}, \text{ curies/sec} \tag{15}$$

where RC_o is the rate

0.736 is the intake rate of the air-conditioning system, m^3/sec

0.006 is the atmospheric dilution factor between the heater stack and the air-conditioner intake (from Reference C-5).

A_i is the activity in curies of a selected isotope

39.6 is the total volume of the radioactive cloud released, m^3; 700 ft^3 of gas released from the coolant loop in one minute and diluted with the 700 cfm of heater flue gas.

The total active volume of the control room, counting room and air-conditioning system is about 425 m^3. Consequently, the rate of change of activity in the control room is expressed by:

$$\frac{dQ}{dt} = R_oC - \frac{R}{425} Q \qquad (16)$$

where Q is amount of activity

t is time in seconds, and the other terms are as defined for Eq. 15.

Solving Eq. 16 for a boundary condition of Q(o) = 0 (acutally 60 seconds after system rupture), the expression becomes:

$$Q(t) = 425\ C_o\ (1-e^{\frac{-Rt}{425}}) \text{ curies, and} \qquad (17)$$

the concentration is:

$$C(t) = C_o(1-e^{\frac{-Rt}{425}}) \text{ curies/m}^3 \qquad (18)$$

The activity intake for the personnel in the control room during the period from t = 60 to t = 120 after system rupture, assuming uniform mixing in the control building, is:

$$I_i = B \int_o^{60} C(t)_i\ dt \text{ curies} \qquad (19)$$

where B is the breathing rate = $3.48 \times 10^{-4}\ m^3/sec$

Equation 18 is substituted into Eq. 19 as follows:

$$I_i = B\,C_o \left[\left(\int_o^{60} dr\right) - \left(\int_o^{60} e^{\frac{-Rt}{425}}\right)\right] dt$$

$$= B\,C_o \left[60 - \frac{425}{R}\left(1 - e^{\frac{-60R}{425}}\right)\right]$$

$$= (3.48 \times 10^{-4})(4.23 \times 10^{-4}\,A_i)$$

$$= 1.47 \times 10^{-7}\,A_i \text{ curies,} \qquad (19a)$$

where A_i is the released activity of the isotope under consideration.

The operating and excursion inventory of isotopes which are significant contributors to the inhalation dose are listed in Table C-1. These values reflect 10,000 hours of operation at 2.2 Mw and a 125 Mw-sec excursion. The release fractions are as defined in Reference C-2. The activities of the isotopes as a function of time as computed by the RISC* code are shown in Figure C-3.

The dose to a particular body organ is given by:

$$D = \sum_{i=1}^{37} I_i\,K_i \text{ Rem} \qquad (20)$$

where K_i is the dose conversion factor for the organ being considered and the i-th isotope. The inhalation dose for the accident and conditions postulated for this analysis is given in Table C-2.

TABLE C-2 - INHALATION DOSE

Organ	Dose, Rem
Thyroid	5.4×10^3
Bone	1.76×10^3
Muscle	1.16
Lung	3.13×10^2
G.I. Tract	2.19×10^2
Testes	24.2

*Internal communication: AN-COMP-168, 20 July 1963. RISC is Radiological Inhalation Safety Code.

TABLE C-1 - OPERATING AND EXCURSION ISOTOPE INVENTORY AND ACTIVITY RELEASED, c

Isotope	Excursion Inventory, Curies	Operating Inventory, Curies	Release, Curies	Isotope	Excursion Inventory, Curies	Operating Inventory, Curies	Release, Curies
La-143	3.67×10^{3}	1.04×10^{5}	5.41×10^{3}	Ru-106	9.03×10^{-3}	3.95×10^{3}	1.97×10^{1}
Ce-143	1.84	1.10×10^{5}	1.10×10^{3}	Rh-105	5.08	1.65×10^{4}	8.25×10^{1}
Pr-143	1.85×10^{-4}	1.10×10^{5}	1.10×10^{3}	Mo-99	1.84×10^{1}	1.12×10^{5}	1.12×10^{1}
Ce-144	1.69×10^{-1}	6.73×10^{4}	6.72×10^{2}	Zr-95	8.44×10^{-1}	1.15×10^{5}	1.15×10^{2}
Ba-141	2.83×10^{3}	1.03×10^{5}	5.28×10^{3}	Y-93	1.29×10^{2}	1.19×10^{5}	5.96×10^{1}
La-141	2.13×10^{1}	1.10×10^{5}	5.50×10^{3}	Rb-92	4.19×10^{4}	8.43×10^{4}	6.32×10^{3}
Ce-141	1.06×10^{-4}	1.10×10^{5}	5.51×10^{3}	Sr-92	1.01×10^{2}	1.08×10^{5}	5.41×10^{1}
Ba-140	4.08	1.14×10^{5}	5.69×10^{3}	Rb-91	5.00×10^{3}	1.05×10^{5}	5.52×10^{3}
La-140	1.51	1.19×10^{5}	5.96×10^{3}	Sr-91	7.94	1.12×10^{5}	5.62×10^{1}
Cs-137	4.64×10^{-3}	2.89×10^{3}	1.45×10^{2}	Y-91	0	1.12×10^{5}	5.61×10^{1}
I-135	1.90×10^{2}	1.16×10^{5}	2.89×10^{4}	Sr-90	4.78×10^{-3}	2.98×10^{3}	1.49
Te-134	1.97×10^{3}	1.30×10^{5}	3.30×10^{4}	Y-90	0	2.96×10^{3}	1.48
I-134	1.87×10^{2}	1.45×10^{5}	3.62×10^{4}	Kr-89	1.44×10^{4}	6.97×10^{4}	4.21×10^{4}
I-133	6.33×10^{1}	1.19×10^{5}	2.98×10^{4}	Rb-89	8.11×10^{2}	8.80×10^{4}	4.44×10^{3}
Te-132	1.08×10^{1}	7.52×10^{4}	1.88×10^{4}	Sr-89	1.00×10^{-3}	8.82×10^{4}	4.41×10^{1}
I-132	1.76×10^{1}	7.89×10^{4}	1.97×10^{4}	Np-239	0	5.31×10^{2}	5.30
Te-131	1.41×10^{3}	5.32×10^{4}	1.36×10^{4}	Y-92	5.64	1.10×10^{5}	5.50×10^{1}
I-131	3.14×10^{-2}	5.38×10^{4}	1.34×10^{4}	Te-129	1.79×10^{-1}	1.29×10^{4}	3.22×10^{3}
Te-129	3.29×10^{1}	3.67×10^{3}	9.25×10^{2}	TOTAL	7.28×10^{4}	2.99×10^{6}	2.92×10^{5}

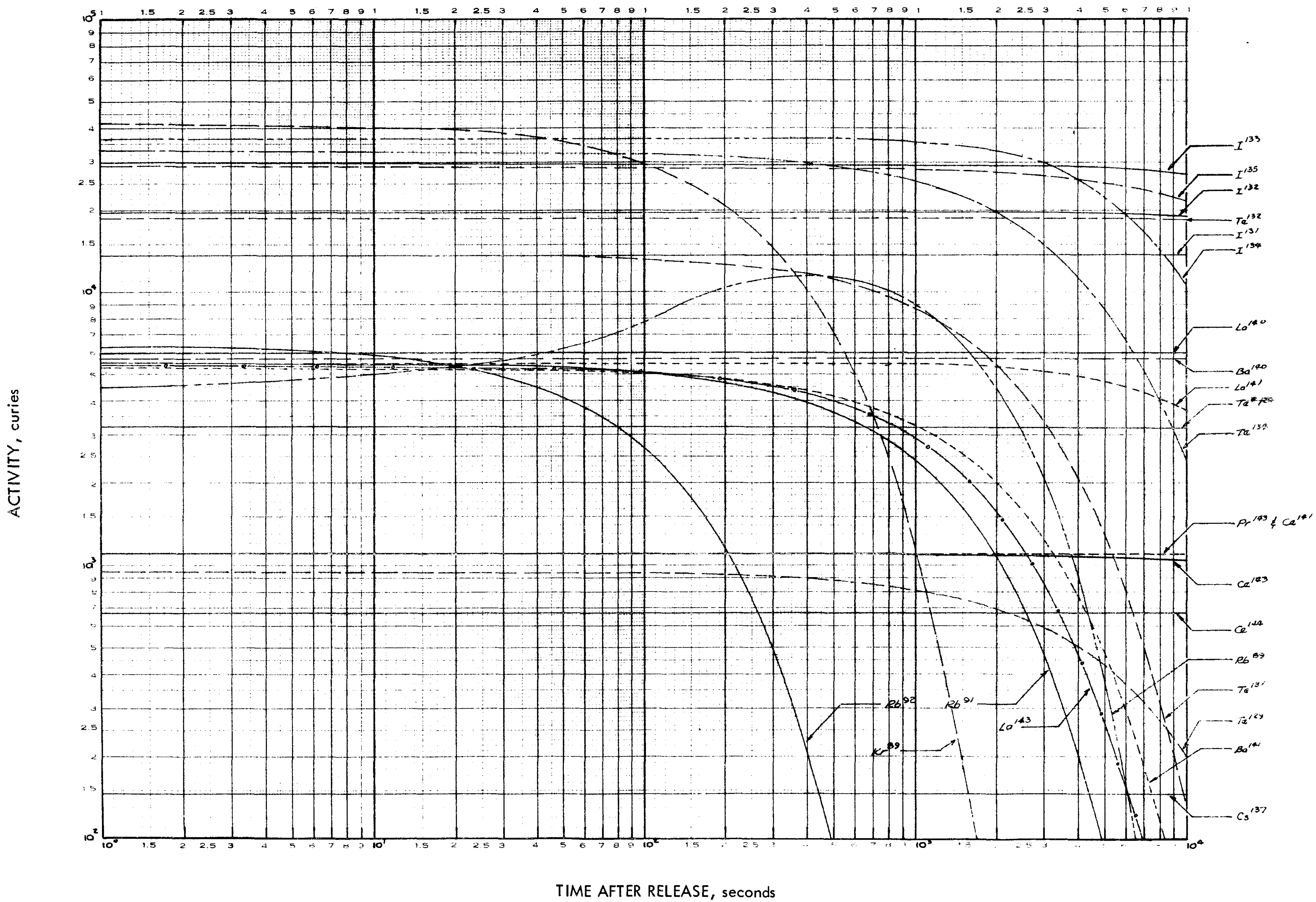


FIGURE C-3a. ISOTOPE ACTIVITY VERSUS TIME (Part a)

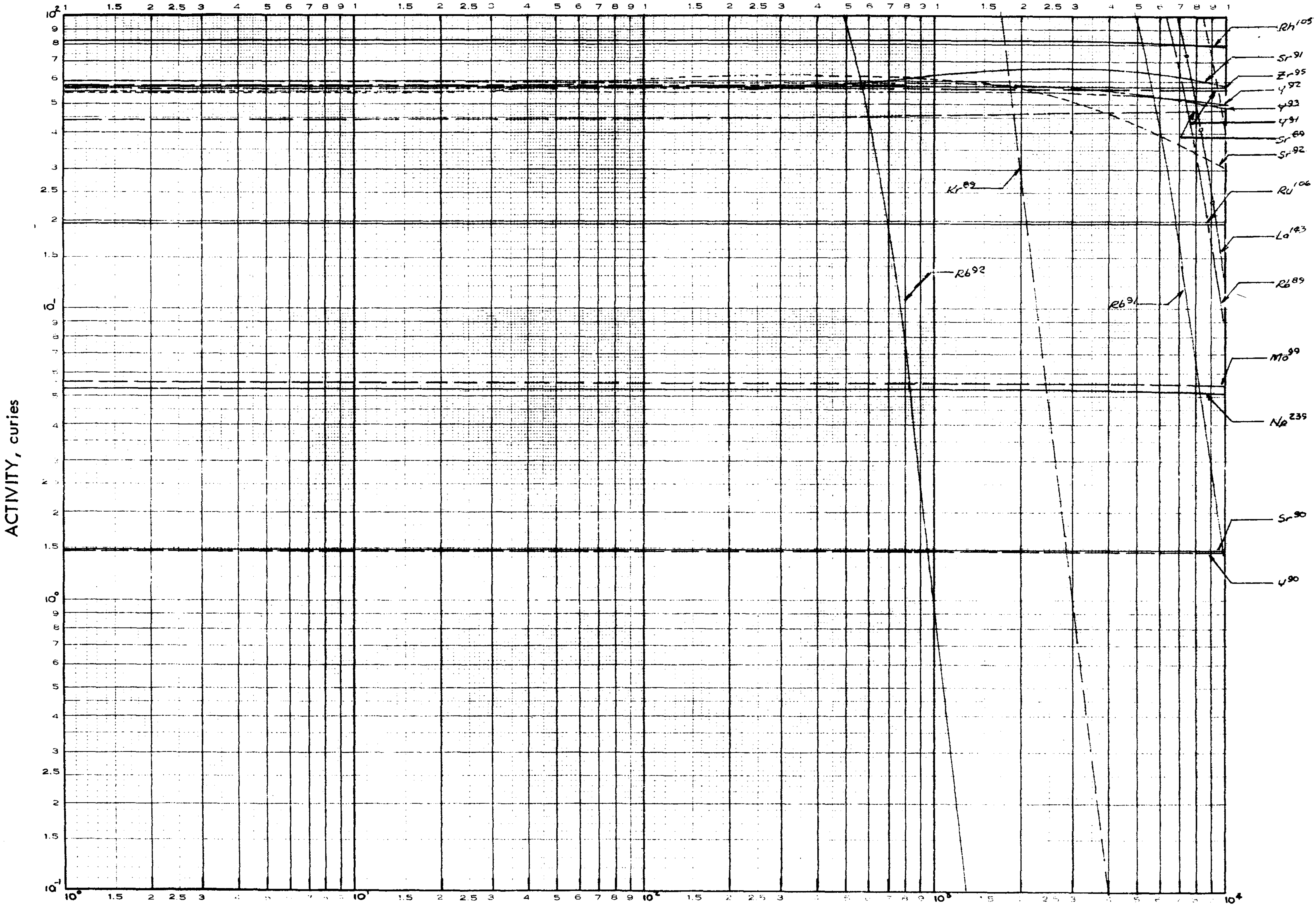


FIGURE C-3b. ISOTOPE ACTIVITY VERSUS TIME (Part b)

2. Beta Skin Dose

The beta skin dose received by personnel in the control room during the one minute period under consideration is given by (C-6):

$$D = 3.07 \int_{0}^{60} C(t)dt \tag{21}$$

$$= (0.307)(4.23 \times 10^{-4})(2.75 \times 10^{6})$$

$$= 356 \text{ rad or } 356 \text{ rem/minute.}$$

3. Whole Body Gamma Dose

The dose to a receptor at the origin of a hemisphere was used to estimate the whole body submersion gamma dose for personnel in the control room. The radius of the hemisphere was chosen equal to the radius of the semi-circle made by the instrument racks (400 cm). The dose for personnel remaining in the control room 2 minutes and 11 minutes after the initial system rupture (1 minute and 10 minute integrated doses) were calculated.

The dose rate is expressed by:

$$D = \int_{v} \frac{K\, S_v(t)\, dV}{4 \pi r^2} \tag{22}$$

where K is the dose conversion factor

S_v is the volume source term

r is the distance from the elemental source term to the receptor, m

It was again assumed that the total source is divided into two equal portions which have energies of 0.7 and 1.22 Mev respectively. Consequently,

$$S_v(0.7) = \frac{C(t)(3.7 \times 10^{10})(0.7)}{(2)(10^6)}$$

$$= 1.3 \times 10^4 \; C(t) \text{ Mev/cm}^3 \text{ and,}$$

$$S_v(1.2) = \frac{C(t)(3.7 \times 10^{10})(1.22)}{(2)(10^6)}$$

$$= 2.25 \times 10^4 \; C(t) \text{ Mev/cm}^3$$

From Eq. 18,

$$C(t) = C_o \left(1-e^{-\frac{Rt}{425}} \right)$$

$$= 44.24 \; (1 - e^{-0.00173t}), \text{ so that}$$

$$S_v(0.7) = 5.75 \times 10^5 (1 - e^{-0.00173t}), \text{ and}$$

$$S_v(1.22) = 9.95 \times 10^5 (1 - e^{-0.00173t})$$

Expanding Eq. 22 yields:

$$D = \frac{K \; S_v(t)}{4\pi} \int_o^{2\pi} d\theta \int_o^{\frac{\pi}{2}} d\phi \int_o^{400} \frac{r^2 \; \sin \phi \; dr}{r^2} \qquad (22a)$$

$$= 200 \; K \; S_v(t) \text{ mr/sec}$$

Using $K(0.7) = 5.62 \times 10^{-7}$ mr/sec/Mev/cm^2-sec, and

$K(1.22) = 1.15 \times 10^{-7}$ mr/sec/Mev/cm^2-sec

and the values of S_v derived above in Eq. 22a:

$$D(0.7) = 64.6(1 - e^{-0.00173t}) \text{ mr/sec, and}$$

$$D(1.2) = 102.5(1 - e^{-0.00173t}) \text{ mr/sec}$$

The dose for the period of time T is, then,

$$D_T = \int_o^T \left[D(0.7) + D(1.2) \right] dt.$$

$$\text{If } i = \int_o^T (1 - e^{-0.00173t}) \; dt$$

$$= T - 578 \; (1 - e^{0.00173t}),$$

When T = 60, i = 2.2 and when T = 600, i = 228 and the dose for one minute is:

$$D_{60} = (2.2)(64.6 + 102.5)$$

$$= 0.368R$$

The dose for ten minutes is:

$$D_{600} = (228)(64.6 + 102.5)$$

$$= 38R$$

REFERENCES

C-1 Zigman, P. and Mackin, J., Early Time Decay of Fission Product Mixtures, USNRDL-TR-400, 11 February 1960

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C-5 Mankee, E. H., Jr., "Methods of Estimating Air Pollution Dispersion of a Relatively Smooth Terrain from Routine Meteorological Observations", 219th National Meeting, American Meteorological Society, 20 June 1963

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STANDARD OPERATING PROCEDURE
ML-1/GCRE OPERATION

PROCEDURE SECTION: 3 SAFETY

TITLE: MASTER EMERGENCY PLANS

SCOPE

This procedure outlines the actions to be executed by personnel in the GCRE in the event of any of several types of emergency situations. The emergency signals, responsibilities, and procedures are defined and described.

Section A of this procedure describes the emergency signals, and defines the responsibilities of various members of the operating staff. The remaining sections outline the procedures for the specific types (by cause) of the emergency situations which are covered by this master plan. These types of emergencies are listed below along with their section letters.

1) Emergency Situations Arising Within the GCRE Area

a.	Radioactive Release	Section B
b.	Fire	Section C

2) Emergency Situations Originating Outside the GCRE Area

a.	Radioactive Release	Section B
b.	Fire	Section C
c.	Aircraft Attack (CONELRAD)	Section D
d.	Security Alert	Section E
e.	Earthquake, Flood, Strike, Riot, etc.	Section F

The procedure which outlines the drills, practices and tests of these emergency plans is ANSOP 3901.

PROCEDURE

A. GENERAL PLANS

1. Emergency Signals: The emergency signals employed at the GCRE area are described below. The detailed procedures for actuating these signals are given in the sections that follow.

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a. FIRE - an INTERMITTENT RINGING of the FIRE GONG system throughout the plant. This signal is always followed by a verbal announcement which identifies the location of the fire and gives such other directions as are required.

b. EVACUATION - a LONG FREQUENCY WARBLING TONE on the plant public address sytem. This signal is always followed by a verbal announcement which specifies the type of emergency and gives general directions for the evacuation.

c. ALERT - a SHORT FREQUENCY WARBLING TONE on the plant public address system. This signal is always followed by a verbal announcement which specifies the type of alert.

d. ALL CLEAR - a STEADY TONE on the plant public address system. This signal is always followed by a verbal announcement which confirms the all clear condition.

2. Responsibility:

a. Each individual regularly assigned to the GCRE area has the responsibility for committing to memory the emergency plans described herein and, when required, for executing the procedures in a calm and efficient manner.

b. The ranking supervisor of the operating organization on duty will automatically assume the title and duties of Emergency Director when an emergency arises. This individual will be responsible for making contacts with the outside agencies, for initiating the various phases of the procedures that follow, and for the control and safety of the personnel in the area. All other personnel will take direction from the Emergency Director.

c. The Manager, AEC-IDO, and more specifically, the Control Group established under his authority, is responsible for executive direction of all activities at NRTS during periods of emergency. The GCRE Emergency Director and consequently all GCRE personnel will accept direction from this authority.

3. Assistance:

a. In the event of an emergency situation originating within the GCRE, the Emergency Director is authorized to request directly such assistance as may be required from the stand-by support organizations, i.e., NRTS Fire Department, NRTS Security Division, NRTS Medical Division and NRTS Site Survey Section, and to request through the NRTS Radio Communication Control Center such other assistance as may be necessary.

b. In the event of an emergency arising outside the GCRE, the Emergency Director will provide such assistance as requested by the Control Center. GCRE-I operations will be suspended if required to provide this requested assistance.

B. RADIOACTIVE RELEASE

1. Originating Within the GCRE Area:

a. Initiation of Alarm:

1) Any member of the GCRE organization who feels that in his judgement, a major release of radioactivity has occurred or is irremediably imminent, will initiate the emergency alarm. This is accomplished by contacting the control room by the fastest means and directing that the alarm be sounded.

2) The control operator sounds the alarm as follows:

Turn the "Evacuation Signal" switch, Section A, console, to LONG. This actuates a long warbling tone on the plant public address system.

b. Emergency Procedure:

1) All personnel on the site will go as expeditiously as possible to the counting room except for the persons noted below:

(a) The control cab operator will:

(1) Scram the reactor

(2) Pick up the operations log

(3) Proceed to the counting room

(b) The GCRE control console operator will:

(1) Record the wind direction and velocity on ANSOL 5752

(2) Proceed to the counting room carrying the ANSOL 5752 sheet

(c) The GCRE control room data taker will:

(1) Shut down the building evacuation blower, the reactor pit supply blower, the mechanical equipment room heater

(2) Shut down the air-conditioner

(3) Proceed to the counting room

(d) The guard will:

(1) Notify patrol headquarters of the alarm by the fastest means

(2) Unlock and open the main gate

(3) Proceed to the counting room

(e) The Emergency Director will:

(1) Interrupt the tone to deliver the following message:

"This is an emergency evauation. All personnel report immediately to the counting room."

(2) Reactivate the evacuation tone

(3) Check that the reactor has been scrammed

(4) Check that blowers have been shut down

(5) Obtain a portable survey meter from the Instrument Shop

(6) Check that the air conditioner has been shut down

(7) Proceed to the counting room

2) The first person arriving in the counting room will immediately turn on the emergency radiation meter stroed in that area by following the procedure posted on the instrument and log the radiation level detected on the log sheet provided.

3) The Emergency Director will attempt to phone the NRTS Radio Communication Control Center, phone 2201 or 2345. If this call can be completed, he will give all the information at his disposal to the dispatcher and stand by on the open line for further instructions.

4) If the call to the Control Center cannot be completed, the Emergency Director will take alternative action as follows:

   (a) Determine the radiation level in the counting room using the portable survey instrument and the installed radiation meter.

   (b) If the radiation level is less than 50 mrem/hr personnel will remain in the counting room until contact is made with and direction received from the NRTS Control Center.

   (c) The H&S Supervisor, or in his absence, the Emergency Director, shall put on emergency protective clothing (stored in the counting room), leave the counting room, make a quick area survey to determine the level and extent of the radiation field and return to the counting room door.

   (d) If this survey indicates that personnel will receive less exposure by evacuating than by remaining in the counting room the Emergency Director will order the evacuation.

   (e) If it is determined that personnel exposure will be less by remaining in the counting room, the Emergency Director will not initiate evacuation.

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(f) If the survey conditions so indicate, the Emergency Director will reactivate the building exhaust blower, space heater or reactor pit blower as indicated to clear the building. Under no conditions will the air-conditioner be reactivated until a survey indicates that no detectable airborne reactivity remains in the test building.

5) If evacuation is directed:

(a) All personnel will put on emergency clothing and equipment, as directed by the Emergency Director.

(b) On signal from the Emergency Director, all personnel will proceed quickly and calmly to the emergency evacuation bus.

(c) The first qualified driver in the bus will start the motor and when all personnel are aboard and checked off the guard's log of personnel, pull out in the direction of Highway 20.

(d) Upon arriving at junction of US-20 and Filmore Blvd., the Emergency Director will advise the driver which route to follow.

(e) Personnel will remain in the bus until dismounting, removal of clothing, survey, and decontamination can be affected at a location specified by the NRTS Control Center.

6) The attempt to phone the Control Center, step b-(3) above, and the radiation survey, step b-(4) above, will be repeated at intervals until the evacuation has been made or until contact has been established with the NRTS Control Center.

2. Radiation Release Originating Outside the GCRE Area:

a. Initiation of Alarm: When directed by the NRTS Control Center, the Emergency Director will:

1) Actuate the alarm by turning the "Evacuation Signal" switch, console, to LONG.

2) Interrupt the tone after 10 seconds to make the following verbal announcement: "This is an evacuation. All personnel proceed to the evacuation bus."

3) Re-institute the evacuation tone.

4) Repeat the verbal announcement after 30 seconds.

b. Emergency Procedure:

1) Upon receipt of the alarm, personnel will act as follows:

(a) Control Cab Operator:

(1) Scram the reactor.

(2) Pick up the operations log.

(3) Proceed to the evacuation bus.

(b) Control Console Operator:

(1) Read and record wind direction and velocity on ANSOL 5752.

(2) Proceed to the evacuation bus with the ANSOL 5752 sheet.

(c) Guard:

(1) Notify patrol headquarters of the emergency evacuation.

(2) Open the main gate.

(3) Proceed to the evacuation bus.

(d) Emergency Director:

(1) Check that the reactor has been scrammed and that the reactor shutdown is apparently procedding normally.

(2) Obtain a portable survey meter from the Instrument Shop.

(3) Proceed to the main gate and check that all personnel get on the evacuation bus. This check will be made on the basis of the guard's log and personal knowledge.

(4) When all personnel are evacuated, proceed to the evacuation bus.

(e) All other personnel:

(1) Secure classified data and lock the files.

(2) Proceed to the evacuation bus.

(3) The first qualified driver aboard will start the motor.

(4) The bus will move out by order from the Emergency Director.

2) The evacuation bus will proceed in accordance with instruction received by the original contact from the Control Center or from the security radio car which may be present at the site. In the absence of any specific instructions, the bus will go to the intersection of Filmore Blvd. and US-20 and then proceed under the direction of the Emergency Director until contacted by a security radio car.

3) Dismounting, survey and further action will be as directed by the Control Center.

C. FIRE

1. Occurring At The GCRE Area:

a. Initiation of Alarm. Any person observing a fire in the GCRE area or its immediate environs will immediately proceed to the nearest fire alarm box and initiate the alarm. NOTE: Fire alarm boxes are located and identified as follows:

Box No. 211 - Guard House

Box No. 212 - Service Building

Box No. 213 - Main entrance lobby - Control Building

Box No. 214 - Wash down area - Control Building

Box No. 215 - Main Control Room

Box No. 216 - North entrance - Control Building

In addition, automatic fire detectors are located and identified as follows:

Alarm No. 221 - Four detectors located in the ceiling of the Main Control Room.

Alarm No. 223 - Six detectors located in the ceiling of the mechanical equipment and electrical equipment rooms.

Alarm No. 224 - Detectors located in the cooling tower.

All the above boxes and automatic detectors connect directly with the NRTS fire alarm system. Actuation of any alarm trips an automatic signal system which rings a single stroke gong. This gong strikes both to announce the fact the alarm has sounded and to identify the location of the fire.

b. Emergency Procedure

1) The person initiating the alarm will:

(a) Immediately contact the control room by the fastest possible means and give the control operator all information at his disposal concerning the location and nature of the fire.

(b) Proceed with "first-aid" fire fighting.

(c) Report to the Emergency Director when the latter arrives at the scene.

2) The control cab operator will:

(a) Scram the reactor.

(b) Contact and notify the Emergency Director of the situation.

(c) Announce over the public address system the location of the fire.

(d) Stand by for further instructions.

3) Fire brigade members will:

(a) Clear work areas of and secure classified information.

(b) Proceed immediately to the scene of the fire and fight the fire under direction of the Emergency Director.

4) The Emergency Director will:

(a) Report to the scene of the fire

(b) Check with the person who initiated the alarm to determine any information concerning the fire which is not obvious.

(c) Direct the fire fighting in accordance with established procedures and training.

(d) Brief the NRTS fire chief when he arrives at the scene and transfer responsibility to him for fire fighting at a time agreeable to both parties.

NOTE: If the fire involves actual or potential release of radioactivity, the Emergency Director shall consult in detail on this subject with the NRTS fire chief but shall retain responsibility for the overall direction of the activities. If evacuation of personnel is indicated, the evacuation will be accomplished in accordance with the applicable portions of section B-2-b above.

(e) Organize and control personnel at the site to avoid confusion, protect property and avoid panic or injury.

(f) Consult with the NRTS fire chief when asked.

(g) Inform the AGN/NRTS Supervision of the fire.

5) All other personnel:

(a) Clear work areas and secure classified information.

(b) Stand by for further instructions.

2. Fire Occurring Outside the GCRE Area:

a. Receipt of Alarm. The Emergency Director will receive the information concerning the fire from the NRTS Control Center.

b. Emergency Procedure: The Emergency Director will take action in accordance with the directives received from the Control Center. If evacuation is ordered, this action will be accomplished by following the applicable portions of Section B-2-b, above.

D. CONELRAD (AIRCRAFT ATTACK)

1. Initiation of Alarm:

a. The person who receives a Warning Red message from the NRTS Control Center shall immediately sound this alarm:

1) Turn the "Evacuation Signal" switch, at console, to SHORT. This will actuate a short warbling tone on the public address system.

2) After ten seconds, discontinue the tone and make the following verbal announcement: "This is a CONELRAD Warning Red. All personnel prepare to evacuate."

3) Repeat the tone and the announcement, steps a-1) and a-2) above.

4) Inform the Emergency Director of the emergency when he reports to the Control Room.

5) The Emergency Director will then immediately order an evacuation as follows:

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(a) Turn the "Evacuation Alarm" switch to LONG.

(b) After ten seconds, discontinue the tone and make the following verbal announcement: "This is a CONELRAD evacuation. All personnel will proceed to the evacuation bus."

(c) Turn the "Evacuation Alarm" switch back to LONG.

b. The person who receives any other message from the NRTS Control Center shall immediately relay the information to the Emergency Director.

1) Upon being notified of a Warning Yellow, the Emergency Director will:

(a) Turn the "Evacuation Signal" switch to SHORT.

(b) After ten seconds, discontinue the tone and make the following announcement: "This is CONELRAD Warning Yellow. All personnel prepare to evacuate."

(c) Repeat the tone and the announcement above.

2) Upon receipt of the All Clear (Fade Out), the Emergency Director will:

(a) Confirm the all clear by telephone with the Control Center.

(b) Turn the "Evacuation Alarm" switch to STEADY. This will actuate a monotone on the plant public address system.

(c) After ten seconds, discontinue the tone and make the following verbal announcement: "This is the All Clear; CONELRAD Warning Yellow (or Red) is over: this is the All Clear."

(d) Repeat the tone and the announcement above.

2. Emergency Procedure

   a. In case of a <u>Warning Red</u>, all personnel will proceed to evacuate according to Section B-2-b, above.

   b. In case of a <u>Warning Yellow</u>, all personnel will prepare to evacuate in according with the applicable portions of Section B-2-b above, <u>except</u> that actual evacuation to the bus will not be made until ordered by the Emergency Director.

E. SECURITY ALERT

1. Initiation of Alarm:

   a. The person who receives the notification of the Security Alert from the NRTS Control Center shall immediately relay the information to the Emergency Director.

   b. The Emergency Director will then contact Day Supervision for guidance about the task of informing the whole crew of the security alert.

2. Emergency Procedures: The Emergency Director will initiate such action as may be directed by the Control Center.

F. FLOOD, EARTHQUAKE, RIOT, ETC.

1. In the event of flood, earthquake, riot, etc., the Emergency Director will take immediate steps to see that the reactor is shut down and the facility placed in safe condition.

2. The Emergency Director will then contact Day Supervision.

3. The Emergency Director will initiate such further procedures or actions as are directed by Day Supervision or by the NRTS Control Center.