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April 1 , 1981
L1L-100

Office of Nuclear Reactor Regulation
Attn: R. W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Open Item in NUREG 0680

The enclosed table and its attachments represent our response to your letter dated March 6, 1981 concerning the resolution of open items in the TMI-1 Restart Safety Evaluation Report NUREG 0680.

Functional/verification test procedures for the restart modifications will be described in the startup test specification to be submitted by May 1, 1981 (See L1L-054 dated February 28, 1981). The testing planned includes functional testing of the Reactor Protection System Diesel Loading Test and a 48-hour Emergency Feed Water Pump Endurance Test as indicated in NUREG 0680.

Technical Specifications for the restart modifications listed below will be forwarded by May 1, 1981.

Anticipatory Reactor Trips

Operability of PORV and Block Valve, Position Indications for PORV and Safety Valves, and Setpoints

Containment Isolation Modifications

Instrumentation to Detect Inadequate Core Cooling

Emergency Feedwater System Requirements

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TMI-1/TMI-2 Separation

Setpoints Associated with ECCS analyses

Sincerely,

H. D. Hukill
Director, TMI-1

HDH:CWS:vjf

Attachments

cc: B. H. Grier - w/o Attachments
B. J. Snyder - w/o Attachments
D. DiIanni - w/o Attachments
H. Silver - w/o Attachments
L. Barrett - w/o Attachments

SER (NUREG-0680) OPEN ITEMS

ORDER	OPEN ITEM DESCRIPTION	LICENSE RESPONSE
1a - Add'l 6	Commitment to complete EFW 2 hr. air supply modification and reset the EFW pump steam chest safety valves.	Licensee will complete these modifications prior to exceeding 5% reactor power. (See attached revised page to Licensee's response to NUREG-0737).
4	Provide additional design detail for the long-term Fuel Handling Building ESF ventilation system.	Additional design detail concerning major equipment location and capacity and indication of the system automatic actuation functions is attached. (See revised Pages 4 and 5 and a revised system flow drawing for Restart Report Supplemental 1, Part 2, Response to Question 52). We are also continuing to evaluate the criteria for this system in hope of achieving further simplification.
5	Implementation schedule and details for the low level solid radwaste storage facility.	A plot plan for this facility was provided on March 4, 1981 to Mr. R. Jacobs (NRC). Additional information is provided in the Restart Report Supplemental 1, Part 2, Response to Question 53 as amended by Amendment 22.
2.1.2	Justification for applicability of the EPRI Relief and Safety Valve Test Program to TMI-1.	NUREG-0737, Item II.D.1 specifies that the required justification be provided by October 1, 1981. Detailed justification for TMI-1 will be provided by this date. It should be noted that EPRI has obtained input from the various NSSS vendors concerning relief and safety valve flows under accident conditions. EPRI has enveloped these flow conditions with the flows used for the test program. In addition, the safety valves being tested by EPRI are the same make, model and oriface area as those installed at TMI-1. The relief valve (PORV) being tested is also the same make and model as the TMI-1 PORV, but the oriface area is larger, which is conservative. Also see Licensee's letter dated March 3, 1981 (L1L 036).

ORDER ITEM	OPEN ITEM DESCRIPTION	LICENSEE RESPONSE
2.1.3.a	1. Justification for the safety valve elbow tap pressure sensor.	1. The subject justification was provided with Restart Report Amendment 23 as Appendix 2A.
	2. Study of safety and relief valve discharge tailpipe thermocouple response.	2. See the attached revised response to Restart Report Supplement 1, Part 2, Response to Question 36.
2.1.4	Design details for containment isolation modifications.	The design details were provided to R. Jacobs (NRC) on March 9, 1981.
2.1.6.a	Leakage measurement program description and procedures.	The scope of the TMI-1 leakage reduction program is described in Restart Report Section 2.1.1.8. Procedures for leak testing of Decay Heat Removal System (SP1303-11.16) and the Reactor Building Spray System (SP 1303-11.50) were forwarded to Mr. R. Jacobs (NRC) on March 26, 1981. These procedures are representative of procedures under development for the other systems included in the leakage reduction program. In addition to the individual leakage procedures, a procedure is being prepared to evaluate the integrated leakages of all of the systems included in the leakage reduction program. The preventive maintenance program will consist of performing the above procedures on a routine basis (as specified in each procedure) and performing maintenance as necessary to maintain leakages within acceptance criteria. The results of leakage measurements inspections and maintenance will be reported to NRC annually in accordance with Technical Specification 6.9.1.B.3 (see attached Page 6-13 from Technical Specification Change Request 100).

ORDER	OPEN ITEM DESCRIPTION	LICENSEE RESPONSE
2.1.6.b	Plant shielding review using final source terms, and description of required modifications.	See the attached revised Section 2.1.2.3 of the Restart Report.
2.1.7.a	Design description of long-term EFW modifications.	See the attached discussion of progress toward completion of the long-term modifications.
2.1.8.a	Design and operational review of the Reactor Coolant Sampling System and Containment Air Sampling System.	The required review is described in Restart Report Section 2.1.2.4, Amendment 23. In addition, a discussion of the accuracy of the various analyses and installation of the equipment necessary to permit determination of hydrogen or desolved gases will be provided by January 1982.
2.1.8.b	Procedures and evaluations for interim high-range radio-effluent monitors and design details for the final monitors.	Design details of the final high-range radio-effluent monitors have been forwarded separately to Mr. R. Jacobs (NRC). It is expected that the final monitors will be in place before criticality; therefore, no interim methods are planned. If, by July 1, 1981, it becomes apparent that the final monitors will not be in place, the interim method procedures and description will be forwarded by August 1, 1981.
2.1.8.c	In-plant iodine monitoring procedures and training.	The procedures for in-plant iodine monitoring were forwarded March 26, 1981 to Mr. R. Jacobs. Training will be conducted on these procedures and will be completed by July 1, 1981. Emphasis will be placed on two aspects; insuring that exposures are maintained as low as reasonably achievable, and the requirements (frequency etc.) for sampling.

ORDER ITEM	OPEN ITEM DESCRIPTION	LICENSEE RESPONSE
2.1.9	Transient and accident procedures.	See Licensee's letter dated February 28, 1981 (L1L-054).
2.2.1.b	Long-term training program for the Shift Technical Advisor.	See Licensee's letter dated March 19, 1981 (L1L-068).
Additional Item No. 4	Design details and procedural guidelines for RCS high point vents.	The design details and procedural guidelines will be forwarded by July 1, 1981 as specified by NUREG-0737.
Long-Term Order Item No. 2	Program outline to meet NUREG-0737, Item II.K.3.30 (Small Break LOCA Methods).	See Licensee's letter dated March 30, 1981 (L1L-089).
Long-Term Order Item No. 3	Design detail for containment pressure, water level and hydrogen monitors.	Design details for the containment water level and pressure monitors was forwarded to R. Jacobs (NRC) on March 26, 1981. Design details for the hydrogen monitor will be forwarded by June 1981.

NRC NUREG 0680 (SER) ITEM NUMBER & SHORT TITLE	NRC SPECIFIC ITEM AND SCHEDULE PROPOSAL		LICENSEE SCHEDULE COMMITMENT (4)	LICENSEE TECHNICAL REFERENCES	COMMENTS
ADDITIONAL ITEMS (SER)	SER	0737			
1. CWST level alarms	Restart	Not Specified	Before > 5% power (See Comments)	R.R. S1, P2, Q10a	Alarms will not be from independent power supplies before restart. Inst. will alarm on loss of power and local tank readings will be taken.
2. EFW 48 hr. endurance test	Restart	Not Specified	Before > 5% power	R.R. S1, P2, Q7	Startup procedure to be submitted and conducted SER @ C1-8
3. EFW Water source transfer	Restart	Not Specified	Complete		Completed SER @ C1-9
4. MSL Rupt. detection system	Restart	Not Specified	Original design	R.R. S1, P2, Q9	Completed. SER @ C1-9
5. EFW water source protection	Restart	Not Specified	Original design	R.R. S1, P1, Q10	SER @ C1-9
6. EFW Indep of AC power: a. 2 hour air b. Reset EFW turbine safeties	Restart	Not Specified	Before > 5% power Before > 5% power	R.R. S1, P2, Q14 Supp. 1, P.2, Q31 & 33	SER @ C1-9
7. Upgrade EF-V30A/B for steam break environment	Restart	Not Specified	6/30/82 (1)	R.R., S1, P2 Q14	Will not be completed before restart due to equipment delivery. Vendor has not provided commitment date. SER @ C1-10
8. EFW Cross tie break	Restart	Not Specified	Before > 5% power (Targeted for June, '81)	R.R. S1, P2, Q12 ASLB testimony of Capodanno, et. al. and question 6.	NDE of the 10 highest stressed welds to be performed. SER @ C1-10. Engineering evaluation anticipated in 3/81 for basis of weld selection.

In addition to the modifications described above, a ventilation system to mitigate the consequences of a postulated fuel handling accident in the FHB will be installed. This new system will meet the requirements of Regulatory Guide 1.52, Revision 2. This system, and intermediate modifications to the Auxiliary and Fuel Handling Building Ventilation System, are described below.

The Auxiliary (Aux. Bldg) and Fuel Handling Building (FHB) Ventilation System will undergo extensive modifications which will be undertaken in two phases as described below.

Phase 1

Prior to restart, the TMI-1 ventilation system will have been modified as shown in Attachment 2. The following equipment will have been added:

1. Damper U with interconnecting ductwork.

2. Damper T

In the event that contamination (radioactivity) is sensed in the ductwork, Dampers U and T will close, Fan F will trip, and ventilation will be via filter trains M and N. Dampers U and T are seismic Category I, meet ANSI/ASME N509 Construction Classification B, Leakage Classification II, and are designed to fail closed.

Phase 2

Prior to the next refueling, the following additional modifications will be made, as shown in Attachment 3:

1. Filter trains Q and R will be added in the Aux. Bldg. elev. 305 ft. From the intake side of the filter trains, the composition of the train will consist of a prefilter, an electric heater, a HEPA filter, a charcoal filter, and a final HEPA filter. The design of filter trains Q and R will meet the requirements of NRC's Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety Feature Atmospheric Cleanup System Air Filtration and Absorption Units of Light Water-Cooled Nuclear Power Plants."

2. Dampers S and V will be added.

3. Fans G and H will be installed in Aux. Bldg. on elev. 305 ft. together will all connecting ductwork and dampers.

SUPPLEMENT 1, PART 2

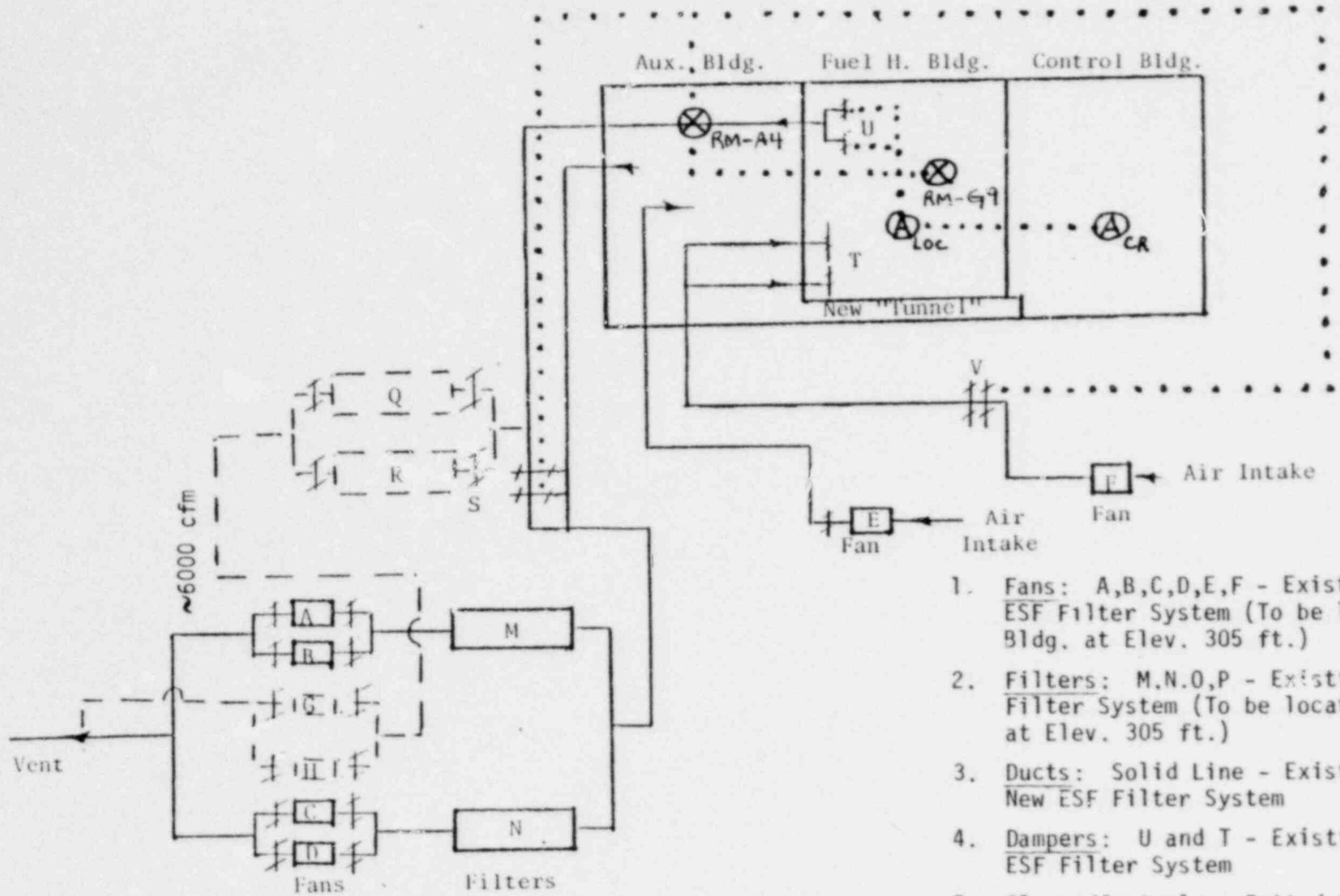
RESPONSE TO QUESTION 52, PAGE 5

4. Dampers U and T will be blocked open and will no longer respond to closure signals.

During normal operation, filter trains M and N (50% capacity) will be utilized together with fans A,C,E and F. During refueling operation, in addition to the equipment normally operating, filter trains R or Q and fan G or H would also operate.

If contamination (radioactivity) is sensed in the ductwork or Elev. 347' area of the FHB, dampers S and H will close, fan F will trip and an alarm will annunciate in the control room and FHB. This action will serve to separate the Bldg. and Ventilation systems and assure that all air leakage will be into the FHB.

ATTACHMENT 3 TO QUESTION 52



1. Fans: A,B,C,D,E,F - Existing G,H - New ESF Filter System (To be located in the Aux. Bldg. at Elev. 305 ft.)
2. Filters: M,N,O,P - Existing Q,R - New ESF Filter System (To be located in the Aux. Bldg. at Elev. 305 ft.)
3. Ducts: Solid Line - Existing; Broken Line - New ESF Filter System
4. Dampers: U and T - Existing; S and V - New ESF Filter System
5. Alarms/Controls: Dotted Lines

   $Ⓐ_{CR}$ - Alarm - Control Room

   $Ⓐ_{LOC}$ - Alarm - Local in FHB

   $Ⓧ_{RM}$ - Radiation Monitor

AUXILIARY AND FUEL HANDLING BUILDING
VENTILATION SYSTEM PRIOR TO NEXT REFUELING

SUPPLEMENT 1, PART 2

QUESTION:

36. Provide a detailed description of the backup capability provided for determining the position of the PORV and pressurizer safety valves beyond the differential pressure transmitters.

RESPONSE:

The differential pressure transmitters will be the primary indication that either the PORV or the safety valves are open. In addition, the PORV position will be monitored by accelerometers, which are described in Section 2.1.1.2. Another indication that either the PORV or the safety valves are open includes the temperature detectors on the discharge lines of these valves and Reactor Coolant Drain Tanks (RCDT) indications. The RCDT level is recorded in the control room, and there is a high level alarm. There are also control room indicators for RCDT temperature and pressure.

It has been determined that the ambient temperature in the vicinity of the tailpipe thermocouples influences them in such a way that it is difficult to determine PORV or safety valve position using them. In order to correct this situation, additional thermocouple junctions will be installed in the long-term to subtract out the influence of ambient temperature. Once modified, the tailpipe thermocouple system will respond in the following way:

Valve Opening:

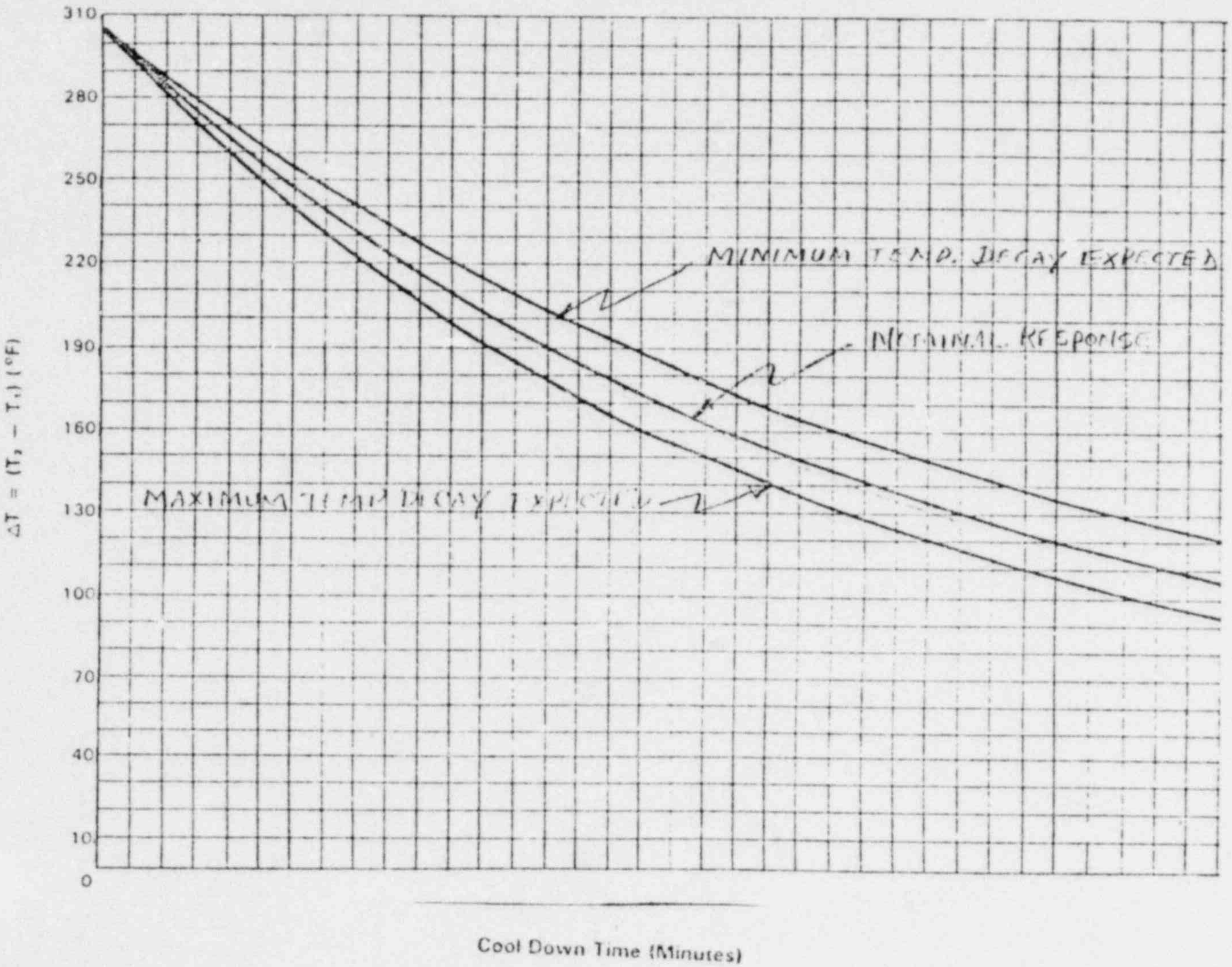
The thermocouple indications will rise quickly in response to steam condensation inside the tailpipes.

Valve Closure:

The operator will plot tailpipe differential temperature (tailpipe surface temperature minus ambient temperature) at one minute intervals. The observed response will be compared to the predicted response range for a fully closed valve. If the temperature does not decay at a rate between the minimum predicted temperature decay rate and the maximum predicted temperature decay rate, it can be inferred that the valve has not fully closed. Expected response for a partially closed valve would be for the observed cooldown of the tailpipe to exceed the maximum expected cooldown rate initially. Eventually, the observed cooldown rate would level off and track above the minimum expected cooldown rate for a closed valve. If the valve is only leaking, the observed response will remain above the expected cooldown rate continuously.

*A typcial temperature decay is gradual as shown on the attached figure.

Cooldown of 4" Φ Tailpipe after PORV Completely Closed



1. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures and their associated man rem exposure according to work and job functions, (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. (This tabulation supplements the requirements of Section 20.407 of 10CFR Part 20).

2. The following information on aircraft movements at the Harrisburg International Airport:

   a. The total number of aircraft movements (takeoffs and landings) at the Harrisburg International Airport for the previous twelve-month period.
   b. The total number of movements of aircraft larger than 200,000 pounds, based on a current percentage estimate provided by the airport manager.

3. The following information from the periodic Leak Reduction Program tests shall be reported:

   a. Results of leakage measurements,
   b. Results of visual inspections, and
   c. Maintenance undertaken as a result of Leakage Reduction Program tests or inspections.

4. The following information regarding pressurizer power operated relief valve and pressurizer safety valve challenges shall be reported:

   a. Date and time of incident,
   b. Description of occurrence, and
   c. Corrective measures taken if incident resulted from an equipment failure.

C. Monthly Operating Reports. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. with a copy to the Region I Office no later than the fifteenth of each month following the calendar month covered by the report.

6.9.2 Reportable Occurrences

Reportable Occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of an occurrence. In case of corrected or supplemental reports, reference shall be made to the original report date. (These reporting requirements apply only to Appendix A Technical Specifications.)

2.1.2.3 Plant Shielding Review

2.1.2.3.1 Introduction

A shielding review has been conducted in response to USNRC Report NUREG-0578 and the September 13, 1979 letter from Darrel G. Eisenhut of the NRC to all operating nuclear power plants (Ref. 10). The requirements defined in the September 13 letter were subsequently clarified in letters to all operating nuclear power plants from Harold R. Denton dated October 30, 1979 (Ref. 11) and D. G.. Eisenhut dated September 5, 1980 (Ref. 12), and NUREG-0737 (Ref. 14).

Among the requirements defined in the three NRC letters and NUREG-0737 is a requirement to conduct a review to determine whether post-accident radiation fields unduly limit personnel access to areas which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident (i.e., vital areas) or unduly degrade the proper operation of safety equipment. An evaluation of plant areas and the methods used in conducting the review for the Three Mile Island Nuclear Station, Unit 1 (TMI-1) is discussed below.

The results of the review have determined which plant equipment and which plant areas requiring post-accident access need to be modified. A description of all the areas reviewed and their access requirements is presented in Section 2.1.2.3.4.

2.1.2.3.2 References

1. Midland Final Safety Analysis Report, Table 11.1-2, Total Core Fission Product Activity Versus Time in Equilibrium Cycle.

2. C. M. Lederer, J. M. Hollander and J. Perlman, "Table of Isotopes," Sixth Edition, John Wiley and Sons, Inc., 1967.

3. A. Tobias, "Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fission Products (Rev. 3)," RD/B/M2669, CNDC(73)P4, Central Electricity Generating Board, Research Dept., Berkely Nuclear Laboratories, United Kingdom, June (1973).

4. E. D. Arnold and B. F. Maskewitz, "SDC - A Shield Design Calculation Code for Fuel Handling Facilities," ORNL-3041, March 1966.

5. <u>Reactor Shielding Design Manual</u>, ed. by T. Rockwell, Technical Information Service, Dept. of Commerce, Washington, DC (1956).

6. H. Ono and A. Tsuruo, "An Approximate Calculation Method of Flux for Spherical and Cylindrical Sources with a Slab Shield," Journal of Nuclear Science and Technology, 2 (6), pp. 229-235, June 1965.

7. K. Shure and O. J. Wallace, "Compact Tables of Functions for Use in Shielding Calculations," Nuclear Science and Engineering, 56 pp. 89-94, January 1975.

8. J. L. Kamphouse, "Cylindrical Source Shielding Equations Utilizing Compact Functions and Including Buildup," Nuclear Science and Engineering, Technical Note, 68, pp. 212-217, November 1978.

9. U. S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Report and Short-Term Recommendations," USNRC Report NUREG-0578, July 1979, Recommendation 2.1.6.b.

10. Letter from D. G. Eisenhut (NRC) to All Operating Nuclear Power Plants, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated September 13, 1979.

11. Letter from H. R. Denton (NRC) to All Operating Nuclear Power Plants, Subject: Discussion of Lessons Short-Term Requirements, dated October 30, 1979.

12. Letter from D. G. Eisenhut (NRC) to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Subject: Preliminary Clarification of TMI Action Plan Requirements, dated September 5, 1980.

13. U. S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC Report NUREG-0660, Vols. 1 and 2, May 1980, Section II.B.2.

14. Clarification of TMI Action Plan Requirements Office of Nuclear Reactor Regulation, Division of Licensing, U.S. Nuclear Regulatory Commission, NUREG-0737, Section II.B.2, October 1980.

15. EDS Nuclear, Inc. Report No. 02-0370-1060, Revision 0, October 1980, Environmental Qualification of Class 1E Electrical Equipment, Three Mile Island Nuclear Station - Unit One Report, (See response to I&E Bulletin 79-01B of January 30, 1981, L1L 026).

2.1.2.3.3.0 Methods

2.1.2.3.3.1 Source Terms

An isotopic core inventory for 310 effective full power days in an equilibrium cycle with a power level of 2552 MWt was utilized for the development of the source terms (Ref. 1). This is presented in Table 2.1-7. This inventory is slightly conservative because the TMI-1 power level is 2535 MWt. It was utilized in the absence of comparable information in the TMI-1 FSAR.

The activity assumed for source term calculations was based on the following:

a. Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% of all others. In determining the source term for recirculated, depressurized cooling water, it was assumed that the water contains no noble gases.

b. Gas Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen inventory.

Two liquid source terms were used in the evaluation. For systems or portions of systems which will contain post accident fluid recirculated from the reactor building sump, the source term was based on diluting the isotopic inventory discussed in item a. above with the minimum expected volume of fluid in the bottom of the reactor building post accident. This volume includes that of the Borated Water Storage Tank (BWST) and the Reactor Pressure Vessel (PV). This is designated as the "Recirculation" source.

For systems which will contain post accident fluid from the reactor coolant system which will not be diluted as noted above, the source term was based on diluting the isotopic inventory in item a. above with the volume of fluid in the reactor coolant system. This is designated as the "Reactor Coolant" source.

The activity assumed for the containment gaseous source term calculations was based on 100% of the noble gas core inventory and 25% of the halogen core inventory. The containment airborne source term was based on diluting the isotopic inventory of item b. above with the air contained in the containment free volume. This is designated as the "Containment Gas" source.

The inventories and source terms discussed above were calculated for the time period immediately after the postulated accident. For other time periods, the decay parameters given in Refs. 2 and 3 were used to adjust the source terms for radioactive decay. The sources were converted to standard shielding source term format as a function of time after the postulated accident and were used as the basic input data to the shielding codes.

2.1.2.3.3.2 Calculation of Dose Rates

Both SDC code (Ref. 4) and the Gilbert/Commonwealth developed SPOT1 code were used in performing the dose rate calculations. The SDC Code uses the methodology of Ref. 5 which represents the cylindrical source by an equivalent line source. The SPOT1 Code uses the methodology originally presented in Ref. 6, which represents the cylindrical source by an equivalent cylindrical annular segment. This methodology was elaborated on by Shure and Wallace in Ref. 7 and is presented as utilized in the SPOT1 Code in Ref. 8*. These codes give comparable calculational results; therefore, the use of one code or the other was based on convenience.

These codes were used to calculate dose rates at the midplane laterally from cylindrical sources. Dose rates were calculated for explicit pipe segments containing various sources. Dose rates were calculated for shielded and unshielded conditions and as a function of time after the postulated accident. Dose rates rate from tankage was also calculated utilizing the shielding codes.

The dose rate was determined at a representative location within a given area and that dose rate was used as the general area dose rate. The criteria used for selection of a representative location was twofold: first, that the dose rate should be reasonably representative of the area and secondly that the dose rate should be conservative for the area. Usually, the main contribution of the dose rate comes from unshielded piping in the immediate area. Other sources farther away or behind shield walls usually contribute significantly less to the general area dose rate.

As indicated above, midplane dose rate data versus lateral distance was calculated for explicit pipe segments containing various sources. Then, the distance from a given pipe to the representative location within the area was determined from the physical piping layout drawings. This distance was then used to determine the dose rate from the dose rate versus distance

*See Appendix 2B for clarification of this methodology.

data. This being done for all pipes in the area, the dose rate contributions were added to obtain the total dose rate at the representative location within the area. Where significant, dose rate contributions from tankage or shielded pipes was also added to obtain the total dose rate for an area.

Denoting time at the inception of an accident as time T=0, and noting that some operations may be performed at times significantly after T=0, dose rates may generally be reduced by a factor of 5 for T=8 hours, and by a factor of about 30 for T=5 days.

2.1.2.3.3.3 Calculation of Doses to Personnel During Post Accident Access to Vital Areas

Personnel doses received in performing a specified operation in a given vital area were calculated as the sum of the doses received during travel to and from the vital area and the dose received while performing the specified operation in the vital area.

The doses received during travel were determined by calculating dose rates at selected locations along the travel route (or at a single location if the dose rate along the travel route is relatively uniform) using the methodology discussed in Section 2.1.2.3.3.2 and multiplying the dose rates by the appropriate travel time for each selected location along the travel route. It was assumed that the individual travels at a rate of 50 feet per minute along the travel paths as indicated in Section 2.1.2.3.4.

Doses received while performing a given operation were determined by multiplying the dose rate for the given area by the time assumed to perform the operation. Dose rates for the given vital area were determined using the methodology discussed in Section 2.1.2.3.3.2. Details with regard to explicit areas are given in Section 2.1.2.3.4.

2.1.2.3.3.4 Acceptance Criteria

The acceptance criteria for personnel access was based on the following guidelines:

a) The post accident dose rate in areas requiring continuous occupancy should not exceed 15 mr/hr (averaged over 30 days).

b) The post accident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 Rem whole body or its equivalent.

c) The minimum radioactive terms used in the evaluation are to be equivalent to the source terms recommended in Regulatory Guide 1.4 as clarified by the references mentioned in Section 2.1.2.3.1.

These guidelines were design objectives and were not a basis for limiting access in the event of an accident.

The acceptability of equipment was determined based on the results of the review of electrical safety equipment conducted in response to IE Bulletin 79-01B and has been reported separately (Ref. 15).

2.1.2.3.4 Results

2.1.2.3.4.1 Review of Plant Areas for Post Accident Access Requirements

Areas which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident have been designated as vital areas. The control room, Technical Support Center (TSC), Operations Support Center (OSC), sampling station and sample analysis area, post-LOCA hydrogen control system, motor control centers, instrument panels, emergency power supplies, security center and radwaste control panels, are included within this designation.

Specific designation and discussion of these and other areas where post accident access might prove useful follows. All areas evaluated are identified by Roman numerals and are shown in Figures 2.1-12, -13, -14, -15, -16, -17 and 18. A summary of the dose rate by area is given in Table 2.1-8.

This review is predicted on the fact that letdown of reactor coolant outside containment will not be employed when coolant activity is at unsatisfactory levels. Letdown will be automatically terminated via the containment isolation system and will not be re-established if activity levels are unacceptably high (refer to Section 2.1.1.5).

Power operated vent valves in the reactor coolant system will ensure that natural circulation and adequate core cooling can be maintained following a LOCA by venting the RCS to the containment atmosphere (refer to Section 2.1.2.2, RCS Venting). When letdown via the normal path to outside containment is not permitted, reactor coolant letdown can be accomplished through the RCS vents. Plant procedures will be modified to address post accident letdown as it relates to post accident shielding.

As a consequence of these actions high pressure injection and low pressure injection piping and components located outside the containment building will contain only the Recirculation source following an accident. Also, there will be no accumulation of radioactive gas resulting from the accident in the waste gas system and components located outside containment.

The radiation dose rates identified herein are based on the conservative assumption that any systems and components that may contain the sources of paragraph 2.1.2.3.3.1 will contain those sources at T=0. Actual recirculation of spilled reactor coolant may not occur until the clean water in the Borated Water Storage Tank is expended.

<u>Areas Requiring Access</u>

<u>Area I</u>

Area I is located in the Fuel Handling Building at the 306' elevation, as shown in Figure 2.1-12. This area is part of the travel route to other areas in the Auxiliary Building.

A radiation dose rate of 0.03 Rem/hour in this area results from the reactor coolant sample inlet line routed through Area V as shown in the figure.

The time to travel through the area will be less than a minute with resultant exposure of less than 0.5 mrem.

<u>Area II</u>

Area II is located on the 305' elevation of the Auxiliary Building as shown in Figure 2.1-12. It contains the liquid and waste gas control panels and the Decay Heat Removal Pumps remote oilers. Also, Area II is in the travel route to Areas III and IV on this elevation.

The source of radiation in this area is the Auxiliary Building HVAC exhaust duct which runs the length of the area about 15 feet above the floor. The dose rate from this source is less than 0.1 rem/hour at T=0 if all of the 0.1 percent per day containment leakage goes through this duct.

A dose rate of less than [illegible].0 rem/hour at T=0 results from an assumed 1 gpm makeup system liquid leakage inside the Auxiliary Building. This dose rate assumes a partition factor of 1 for the noble gases and 0.1 for the halogens. The 1 gpm leak rate greatly exceeds the anticipated leakage in the Auxiliary Building after implementation of the leakage reduction program (reference NUREG-0578 Item 2.1.6a and NUREG-0737 item III D.1.1).

Post-accident access to Area II, may be desired to perform liquid and gas waste transfers and read waste gas decay tank pressure. Occupancy time to perform these activities are 10 minutes and 1 minute respectively, during which radiation exposures of less than 183 and 18.3 mrem respectively, would be experienced by Area II occupants.

Access to Area II may also be required to add oil to the Decay Heat Removal Pumps bearings via the remote oilers. Occupancy time to perform this operation is 5 minutes with a resultant exposure of less than 91.5 mrem.

Areas III and IV

Areas III and IV are located on the 305' elevation of the Auxiliary Building as shown in Figure 2.1-12. Area III contains engineered safeguards motor control centers (MCC's) 1A and 1B, and Area IV contains the reactor coolant pump seal injection filter station.

Post accident occupancy of Area III may be required to reset circuit breakers in MCC 1A and 1B and occupancy of Area IV may be required to open valve MU-V198 to by-pass the seal injection filters.

A radiation dose rate of 6.3 + E3 Rem/hour at T=0 in Area III and 8.0 + E3 Rem/hour in Area IV results from the following Makeup and Purification system piping and components: reactor coolant pump seal injection piping, high pressure injection piping to cold legs 'C' and 'D', and the seal injection filters and valves.

Assuming access is not required to Area III until eight hours after an accident (i.e. T = 8 hrs) the dose rate would be 1.3 + E3 rem/hr resulting in a dose of 100 rem for a five minute stay time to reset breakers.

The dose to an operator staying in Area IV for five minutes at T = 0 to open the valve will be 666 rem.

The estimated radiation dose for travel to and from Areas III and IV is negligible in comparison to the dose received during area occupancy.

Areas XI and XII

Areas XI and XII are located on the 281' elevation of the Auxiliary Building, reference Figure 2.1-13. Within these areas are valves requiring post-accident access for boron precipitation control and continued operation of the decay heat system if the postulated accident was a break in one of the low pressure injection lines with the decay heat pump in the unfaulted line unavailable.

These operations currently require access to and manual operation of valves DH-V15A and B, DH-V19A and B and DH-V38A and B to achieve proper system alignment and flow control. Valves DH-V15A and B and DH-V19A and B are locked open. Valves DH-V38A and B are locked closed. These valves are all located in the decay heat pump pits and DH-V19A, B and DH-V38A, B have reach rod extensions for operation from elevation 281'.

Access may also be required post-accident to open and/or throttle air operated Decay Heat Closed Cooling System valves DC-V2A and B, and DC-V65A and B to achieve reactor coolant temperature control.

A dose rate in Area XI of 7.0 + E3 rem/hr at T=0 emanates from Decay Heat Removal System piping associated with the cross-over lines from the decay heat coolers to the makeup pumps via valves DH-V7A and B, the piping legs back to MU-V14A and B, and the sources located in Areas XIII and XIV.

A dose rate in Area XII of 2.0 + E4 mrem/hr. at T=0 results from decay heat piping associated with injection from the decay heat pumps via DH-V4A and B, Reactor Building sump crossover via DH-V12A and B, the piping leg back to DH-V14A and B, and the decay heat drop line to DH-V12A and B via DH-V3.

The occupancy time to unlock and realign the manual valves is estimated at 5 minutes, and to operate the powered valves is also 5 minutes. The resultant dose to an operator is 580 rem for Area XI and 1700 rem for Area XII. The doses for travel to and from Areas XI and XII are negligible in comparison to the dose received while performing the required operations in these areas.

Areas XIII and XIV

Areas XIII and XIV, as identified in Figure 2.1-13, are the locations of Decay Heat Removal System valves DH-V12A and B and DH-V64 on the 281' elevation of the Auxiliary Building.

These are manual valves which are locked closed. Post accident access is required for opening DH-V12A and B for boron precipitation control and to line up the Decay Heat System to take suction from the reactor coolant hot leg, if desired. Access to DH-V64 is required post-accident for boron precipitation control.

A radiation dose rate of 1.7 + E4 Rem/hour at T=0 in Area XIV emanates from decay heat removal piping associated with injection from the decay heat removal pumps via valves DH-V4A and B; reactor building sump crossover via DH-V12A and B; and the decay heat dropline to DH-V12A and B via DH-V3.

A radiation dose rate of 2.1 + E4 Rem/hour at T=0 in Area XIII emanates from the same piping sources as for Area XIV.

For an estimated occupancy time of 5 minutes to unlock and manually open either DH-V12A, DH-V12B, or DH-V64 the resultant radiation dose to the operator is 1.4 + E3 and 1.8 + E3 rem for areas XIV and XIII, respectively.

The extimated radiation dose for travel to and from Areas XIII and XIV via Areas VI, VII and VIII is 165 Rem. Access could also [illegible] made via area X.

Area XV

Area XV, located on the 281' elevation of the Auxiliary Building as shown in Figure 2.1-13, contains the engineered safeguards valve motor control center (MCC) 1C.

A radiation dose rate of 25 Rem/hour at T = 0 in this area comes from the unshielded reactor coolant sample recirculation line routed above MCC 1C. The rate conservatively assumes a pure undiluted reactor coolant sample is drawn at T = 0.

Access to Area XV may be required after an accident to reset thrown circuit breakers in MCC 1C. Occupancy time for resetting circuit breakers is estimated to be two minutes resulting in a radiation dose of 0.83 Rem to the operator.

Area XVI

Area XVI comprises two associated plant functions; the Nuclear Sampling Room (Area XVIA) and the Radiochemistry Laboratory (Area XVIB) on the 306' elevation of the Control Building as shown in Figure 2.1-15.

The radiation sources for these areas come primarily from the reactor coolant sampling and recirculation lines running into the Nuclear Sampling Room and direct radiation from the Containment Building. The reactor coolant sampling and recirculation source is conservatively assumed to contain pure, undiluted reactor coolant at T = 0.

The Containment Building direct radiation source is less than 0.5 mrem/hour. The radiation dose rate at T = 0 in the Nuclear Sampling Room is 276 rem/hour.

Post-accident access to these areas may be required to draw and prepare a reactor coolant sample for analysis. Refer to section 2.1.2.4 for the discussion of doses received while obtaining a reactor coolant sample.

Area XVIII

Area XVIII is located in the Intermediate Building as shown in figure 2.1-16. This area is the location of the emergency feedwater pumps and associated piping and components. Post accident access to this area may be desirable for inspection and maintenance of equipment.

A dose rate at T=0 of 0.5 mrem/hr. in Area XVIII results from radiation from the containment building.

Area XX

Area XX, located on the 305' elevation of the Intermediate Building as shown on Figure 2.1-16, contains the hydrogen recombiners.

Access to Area XX would be required during the first six days following an accident for installation of the second recombiner. Start-up of a recombiner would be required approximately 12.6 days after the LOCA.

The dose rate in Area XX comes from the Containment Building and the operating recombiner. At T = 0 the dose rate (without a recombiner operating) is 30 mrem/hr. At T = 12.6 days the dose rate, with a recombiner in operation, is 1.56 rem/hr.

The radiation dose rate for travel to Area XX or the hydrogen recombiner control panels, located in a room directly below the recombiner on the 295' elevation, is negligible.

Areas XVII, XXI and XXII

Areas XVII, XXI and XXII are the Operations Support Center, Technical Support Center, and Control Room as shown in Figures 2.1-15, 2.1-17, and 2.1-18 respectively.

The Control Room is on the 355' elevation of the Control Building. The Technical Support Center is located on the 322' elevation of the Control Building. The Operations Support Center is located in the Health Physics Laboratory on the 306' elevation of the Control Building.

The radiation dose rates for these areas comes primarily from the reactor coolant sampling and recirculation lines running into the Nuclear Sampling Room (Area XVIA on Figure 2.1-15) and secondarily from direct radiation from the Containment Building. The reactor coolant sampling and recirculation source is conservatively estimated to contain the Reactor Coolant Source at T = 0. The Containment Building direct radiation source is less than 0.5 mrem/hour.

The radiation dose rates in the Technical Support Center, (Area XXI) the Control Room (Area XXII); and the Operations Support Center (Area XVIIB) and the monitor area (Area XVIIA) are 480, 1.7, 18 and 340 mrem/hour, respectively. The radiation dose rate in these areas would be negligible if reactor coolant sampling was not assumed at T = 0.

Area XXIV

Area XXIV is the diesel generator building. Access to Area XXIV is required post-accident for operation/maintenance of the diesel generators.

The radiation dose rate in area XXIV comes from the Containment Building. At T=0 it is less than 10 mrem/hour. The estimated dose accumulated during travel from the Control Building through the Turbine Building to area XXIV is negligible.

Security Access Center

The security access center is on a direct line of sight with the containment at a distance of about 285 feet. The post-accident dose rate at this location is 750 mr/hr at T=0. It will take approximately ten days for the dose rate to decay to 15 mr/hr. The dose rate average over 30 days is less than 15 mr/hr.

Areas for Which Post Accident Access is Not Required

The remaining areas of Figures 2.1-12 through 2.1-18 are those for which post accident access is not necessary. Access to these areas is not needed because;

1. These areas contain equipment and piping necessary for accident mitigation but do not require personnel access or,

2. These areas do not contain any components for which post accident operator access is necessary or desirable.

These areas are listed in Table 2.1-6.

2.1.2.3.4 Review of Radiation Affects on Electrical Equipment

The design review of electrical equipment has been performed in accordance with NRC IE Bulletin 79-01B and has been submitted seperately (ref. 15).

2.1.2.3.5 Conclusions

A. The TMI-1 plant shielding review has identified areas where post-accident radiation will preclude operator access.

Table 2.1-8 lists the plant areas evaluated and identifies the corresponding dose rates.

Areas where calculated doses preclude post-accident access without appropriate modifications are:

Areas III and IV-	Makeup and Purification System seal injection filter bypass valve MU-V198 and MCCs 1A and 1B
Areas XI and XII -	Decay Heat Removal System Valves DH-V15A & B, DH-V19A & B DH-V38A & B for Boron precipitation control and Decay Heat Closed Cooling System valves DC-V2A & B and DC-V65A & B for decay heat removal temperature control.
Areas XIII and XIV -	Decay Heat Removal System valves DH-V12A & B, DH-V64 for boron precipitation control and sytem operation..

B. Dose rates in the following areas exceed 15 mrem/hr but the dose rates, averaged over 30 days are less than 15 mrem/hr.

Area XVIIB	-	Operations Support Center
Area XXI	-	Technical Support Center

Security Access Center

C. The calculated dose rates for the remaining areas of the plant where post-accident access might be desirable are low enough to allow access for the expected period of occupancy.

2.1.2.3.6 Planned Modifications

A summary of planned modifications is given in Table 2.1-.9.

Table 2.1-6
Areas for Which Access is Not Required

Area Number	Description	Figure 2.1-XX
IX	Makeup Pump Compartments	13
XIX	Leak Rate Air Dryer Area & Piping Compartments	16
XXIII	Decay Heat Pits	14

TABLE 2.1-7
ISOTOPIC CORE INVENTORY

Isotope	Total Core Inventory in Curies	Isotope	Total Core Inventory in Curies
Br 84	1.57 +E7	Xe 133	1.27 +E8
Br 85*	2.19 +E7	Xe 135m	3.26 +E7
Kr 83m	9.25 +E6	Xe 135	2.09 +E7
Kr 85m	2.19 +E7	Xe 138	1.17 +E8
Kr 85	5.30 +E5	I 129*	1.80 +E0
Kr 87	4.00 +E7	I 131	7.35 +E7
Kr 88	5.60 +E7	I 132	8.62 +E7
Rb 88	5.64 +E7	I 133	1.28 +E8
Sr 89	7.42 +E7	I 134	1.60 +E8
Sr 90	3.99 +E6	I 135	1.27 +E8
Sr 91	9.72 +E7	Cs 134	1.27 +E6
Sr 92	9.50 +E7	Cs 136	8.02 +E5
Y 90	3.96 +E6	Cs 137	4.99 +E6
Y 91	9.85 +E7	Cs 138	1.23 +E8
Mo 99	1.28 +E8	Ba 137m	4.67 +E6
Ru 106	2.29 +E7	Ba 140	1.25 +E8
Xe 131m	4.38 +E5	La 140	1.27 +E8
Xe 133m	3.07 +E6	Ce 144	7.50 +E7

* Deleted as insignificant for subsequent calculations.

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TABLE 2.1-8
RADIATION DOSE RATE BY AREA

Area	Figure Reference (2.1-XX)	Dose Rate at T=0 MREM/HR
I	12	3.0 + E1
II	12	1.0 + E6
III	12	6.3 + E6
IV	12	8.0 + E6
V	12	1.3 + E4
VI	13	1.5 + E4
VII	13	8.8 + E4
VIII	13	2.1 + E6
IX	13	1.9 + E6
X	13	1.7 + E1
XI	13	7.0 + E6
XII	13	2.0 + E7
XIII	13	2.1 + E7
XIV	13	1.7 + E7
XV	13	2.5 + E4
XVI	15	2.8 + E5
XVIIA	15	3.4 + E2
XVIIB	15	1.8 + E1
XVIII	16	0.5 + E0
XIX	16	6.1 + E3
XX	16	1.4 + E5
XXI	17	4.8 + E2
XXII	18	1.7 + E0
XXIII	14	4.0 + E7
XXIV	16	1.0 + E1

TABLE 2.1-9
PLANNED MODIFICATIONS

REQUIRED ACTION	LOCATION	DOSE	MAJOR SOURCE OF RADIATION DOSE	MODIFICATION
Manually open valve MU-V198 to bypass seal injection filters	305' elevation of the Auxiliary Building. Area IV, Figure 2.1-12	666 Rem for valve operation, negligible for travel.	Reactor Coolant Pump seal injection piping, filters and valves; high pressure injection piping to cold legs "C" and "D"	Change operating procedures to require manual opening of MU-V198 before going to recirculation from reactor building sump (i.e. before BWST is depleted)
Reset any thrown circuit breakers in motor control centers 1A, 1B	305' elevation, Auxiliary Building. Area III Fig. 2.1-12	100 Rem for circuit breaker reset operation. Negligable for travel	Reactor Coolant Pump seal injection piping, filters and valves; high pressure injection piping to cold legs "C" and "D"	Install a shield wall in area IV, to isolate the MCC's from the piping
Manually operate valves DH-V15A & B, DH-V19A & B, DH-V38A & B, DC-V2A & B and DC-V65A and B for boron precipitation control and for continued decay heat removal.	281' elevation of the Aux. Building. Areas XI and XII Figure 2.1-13	580 Rem in Area XI, 1700 Rem in Area XII. Negligible for travel.	Makeup and purification system and decay heat removal system piping	Change valves DH-V19A & B, DH-V38A & B to remote air operated with air provided from bottled gas supply good for 2 hrs. operation. Provide DC power for valve actuation and manual loaders for positioning DH-V19A & B.* Revise procedure 1104-4 concerning post LOCA boron control so that valves DH-V15A & B remain locked open and valves DH-V5A & B and DH-V6A & B are closed.

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TABLE 2.1-9
PLANNED MODIFICATIONS

REQUIRED ACTION	LOCATION	DOSE	MAJOR SOURCE OF RADIATION DOSE	MODIFICATION
Unlock and open valves DH-V12A & B and DH-V64 for boron precipitation control	281' elevation of the Aux. Building. Areas XIII & XIV Figure 2.1-13	1800 rem for Area XIII, 1400 rem for area XIV. 165 rem for travel	Decay Heat System piping	Change valves DH-V12A & B to electric motor operated.* Operate DH-V64 via reach rod extension on Aux. Building 305' elevation. Extension stem is located so that operator is protected by above noted shield wall.
Sampling and analyzing reactor coolant in Nuclear Sampling Room and Radiochemistry Laboratory.	306' elevation of control Building. Area XVI Figure 2.1-15	Refer to section 2.1.2.4	Reactor coolant sample and sample lines.	Refer to Section 2.1.2.4

*Control panel for remote operation of these valves to be located in the Control Building.

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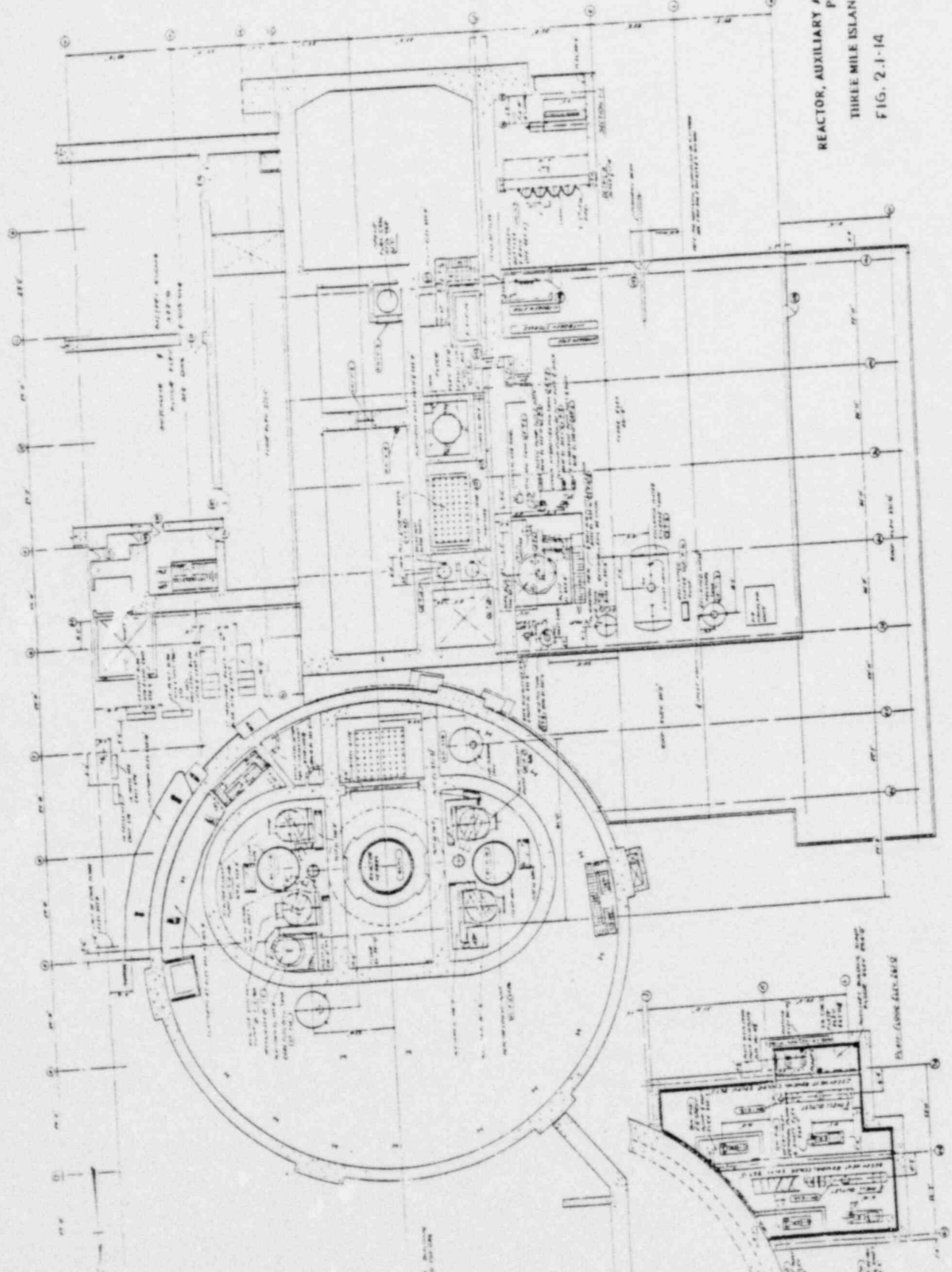
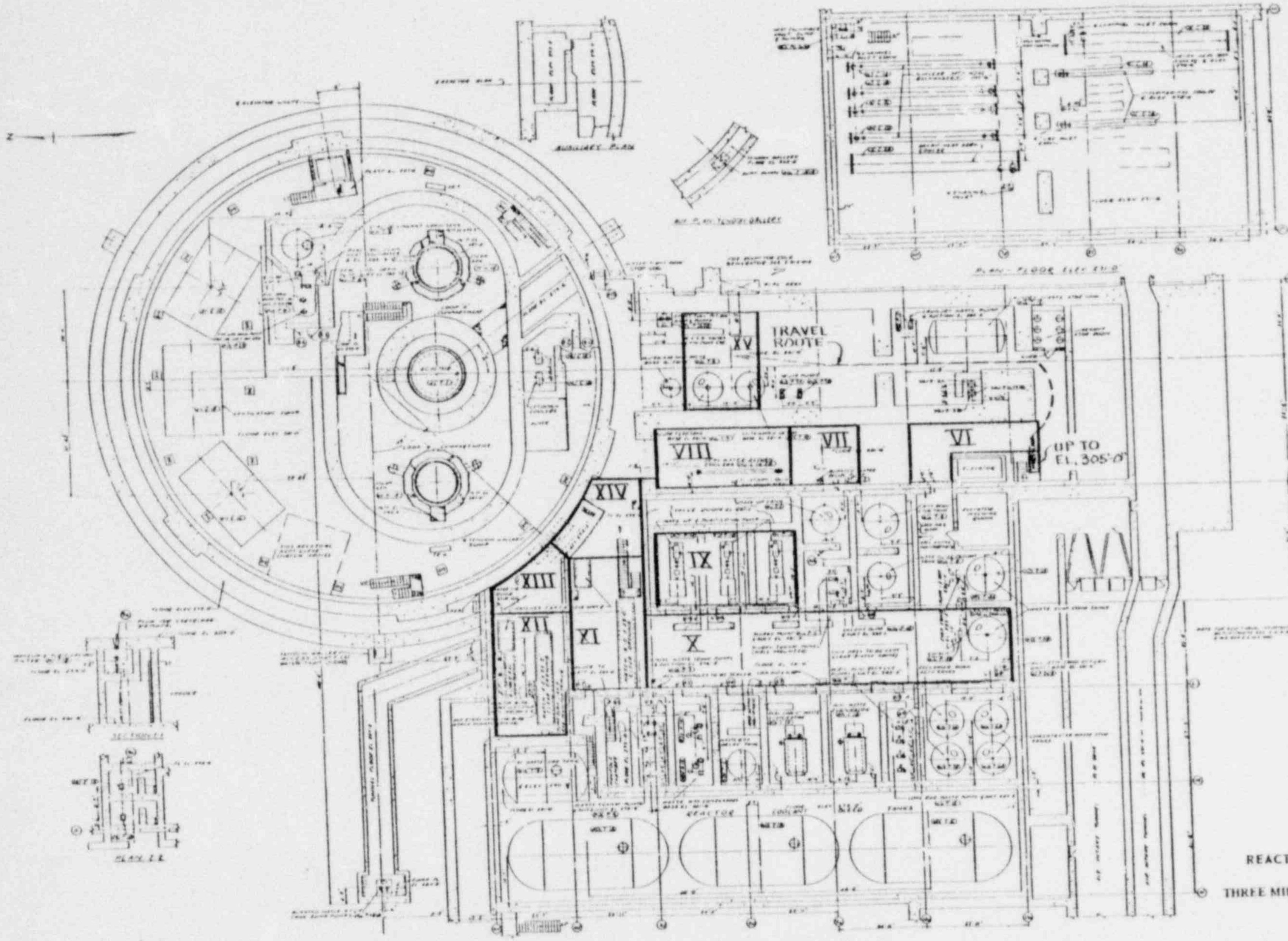
POOR ORIGINAL



REACTOR, AUXILIARY AND FUEL HANDLING BUILDING
PLAN VIEW EL. 261'0" & 331'0"

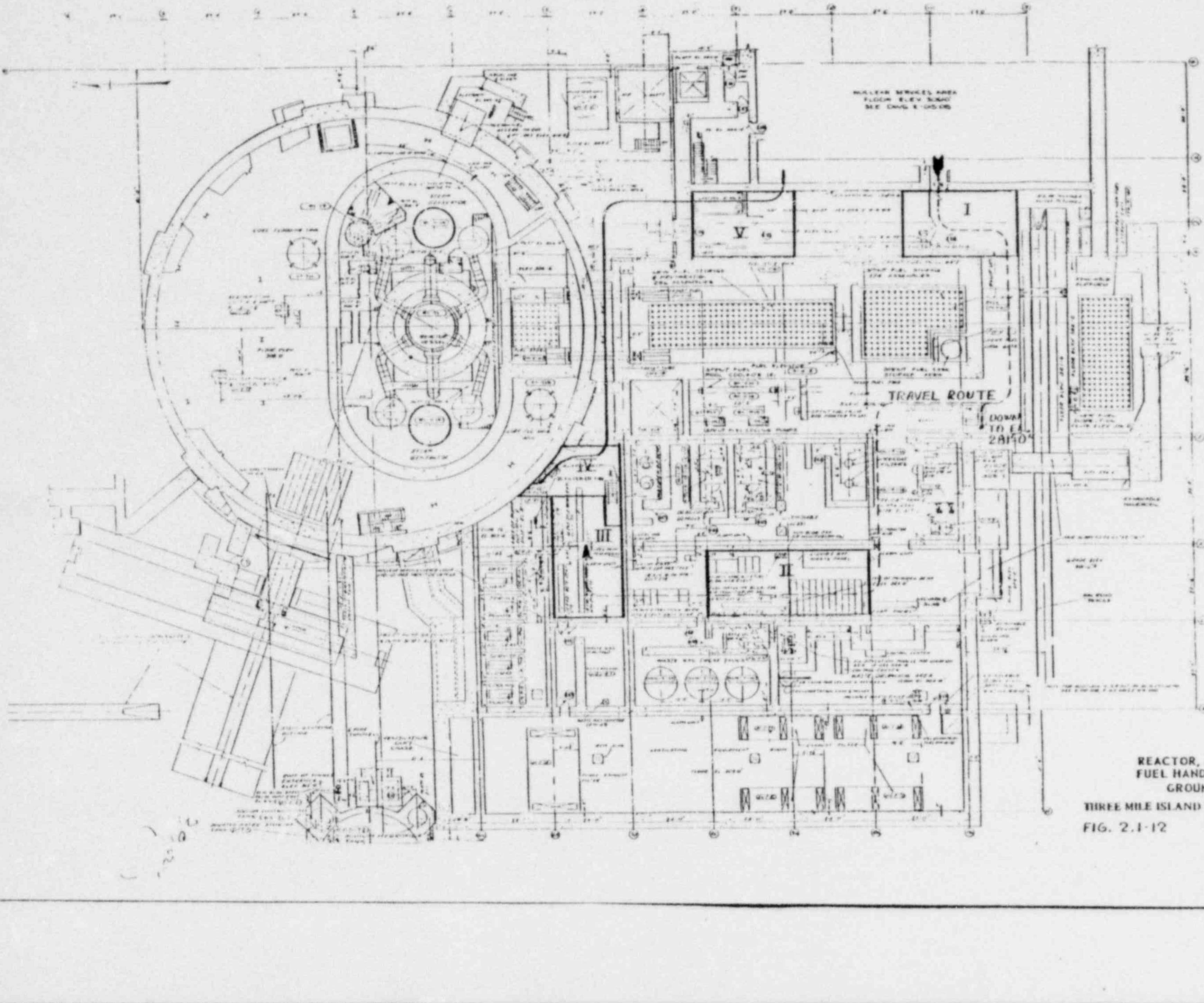
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIG. 2.1-14

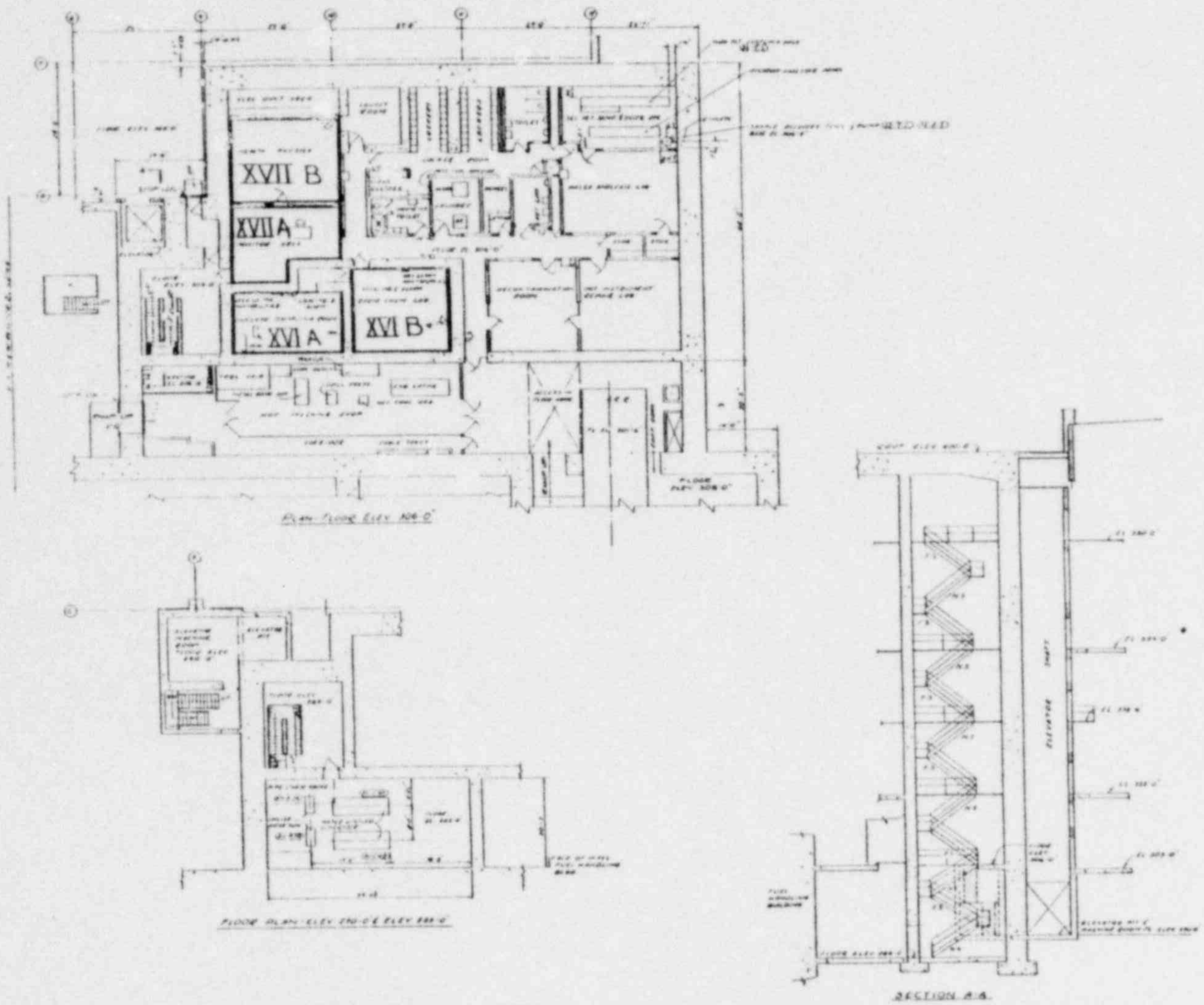


REACTOR AND AUXILIARY BUILDINGS
BASEMENT FLOOR EL. 281'-0"
THREE MILE ISLAND NUCLEAR STATION UNIT

FIG. 2.1-13



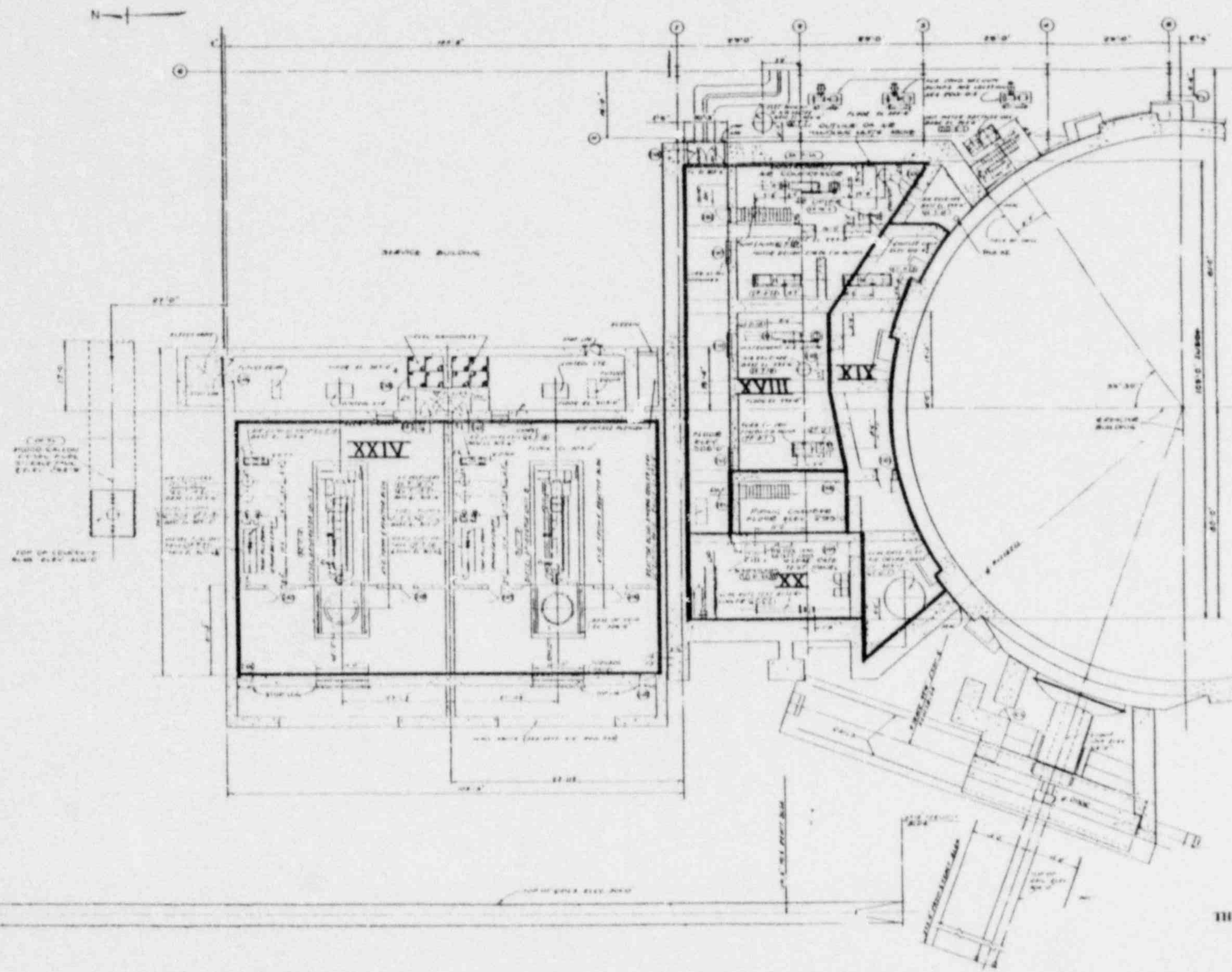
REACTOR, AUXILIARY AND
FUEL HANDLING BUILDINGS
GROUND FLOOR EL. 305'-0"
THREE MILE ISLAND NUCLEAR STATION UNIT 1
FIG. 2.1-12



POOR ORIGINAL

CONTROL ROOM TOWER
THREE MILE ISLAND NUCLEAR STATION UNIT 1
FIG. 2.1-15

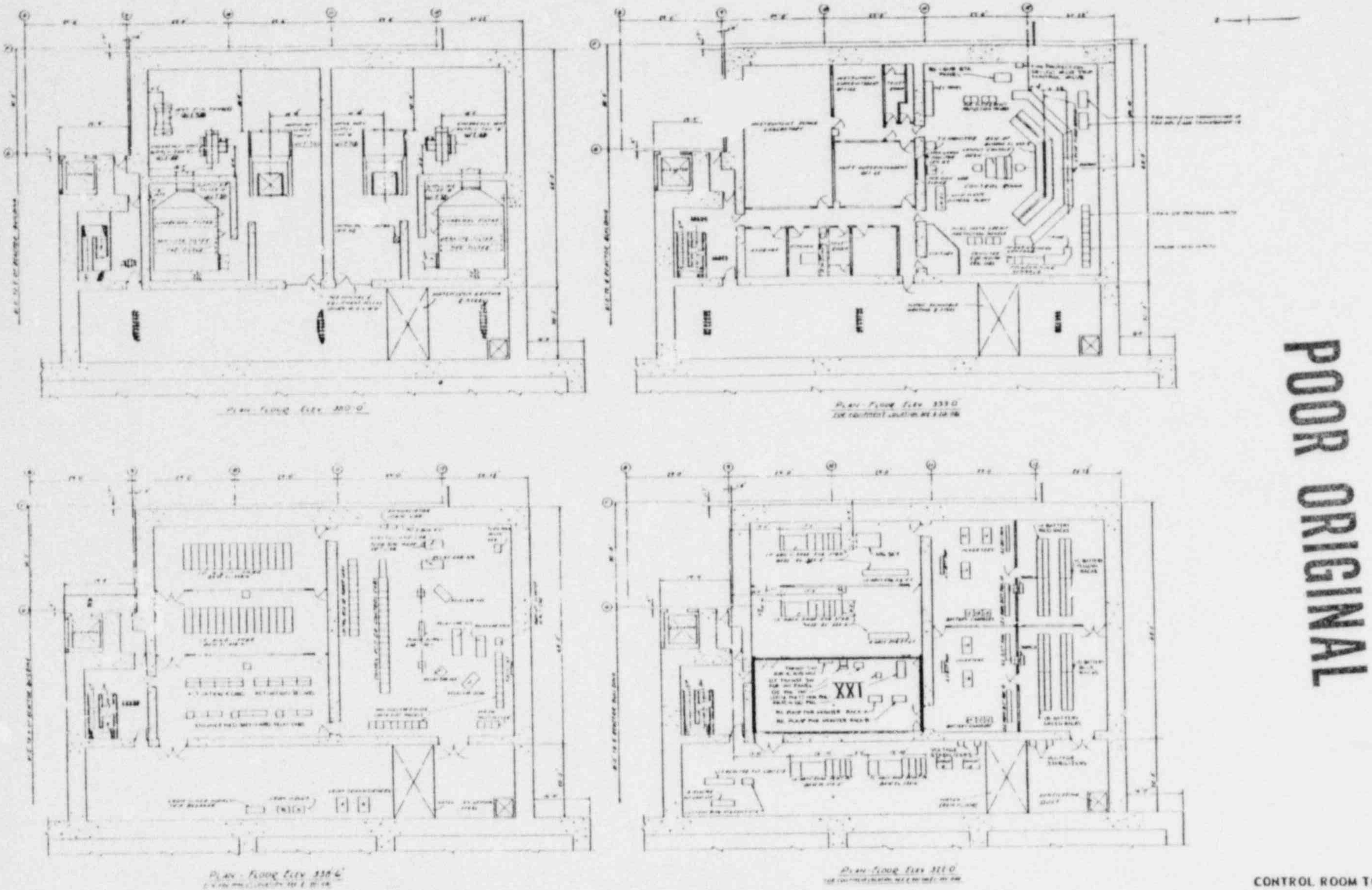
POOR ORIGINAL



INTERMEDIATE BUILDING
BASEMENT FLOOR ℄ EL. 305'-0

THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIG. 2.1-16



CONTROL ROOM TOWER
THREE MILE ISLAND NUCLEAR STATION UNIT 1
FIG 2.1-17

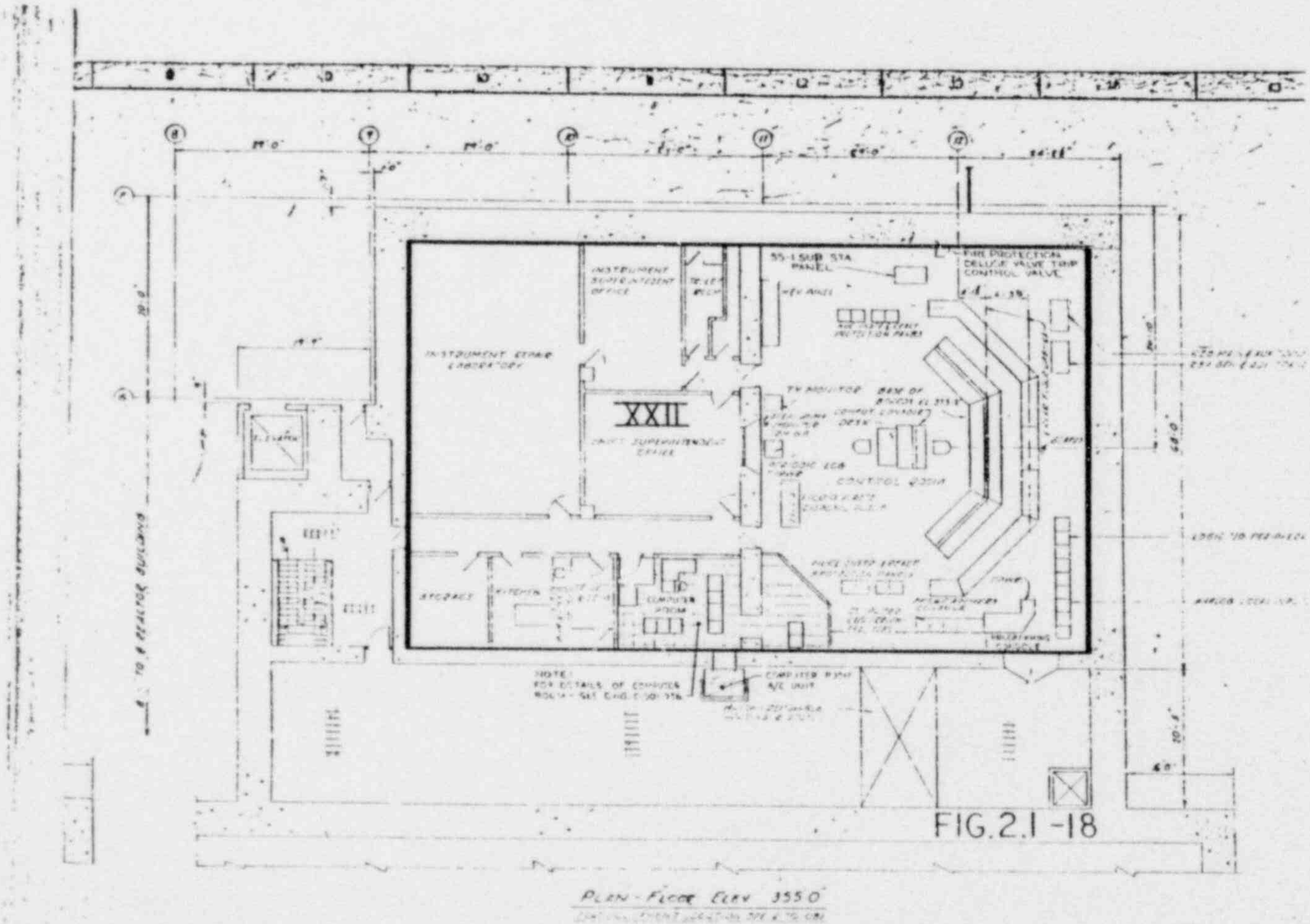


FIG. 2.1 -18

CONTROL ROOM TOWER
THREE MILE ISLAND NUCLEAR STATION UNIT 1
FIGURE 2.1 -18

STATUS OF LONG TERM MODIFICATIONS
FOR UPGRADING OF THE TMI-1
EMERGENCY FEEDWATER (EFW) SYSTEM

1. Cavitating Venturi

Engineering and calculations necessary to size and purchase this equipment are complete. A purchase order is expected to be issued in May 1981 and no delivery difficulties are anticipated.

2. Redundant Control and Block Valves

The block valves have been selected and orders are expected to be placed by June 1981. The control valves were selected to be compatible with the existing air operated control valves since upgrading of the existing valve operators was planned. Recent information from the vendor indicates that upgrading of the existing valves is more difficult than previously indicated and may require the involvement of test facilities not under the control of the vendor and may cause schedule problems. We are evaluating the use of control valves supplied by other vendors; however, the delivery schedules for the alternatives may be as long as two years. In addition, the compatibility of these alternative designs would also require a redesign of the control system and the air supply. Selection of the valve design is expected to be made by May 1981 and delivery schedules will be known at that time.

3. Addition of OTSG level and Feedwater/Steam d/p To The Safety Grade EFW Pump Auto-Start System

The Feedwater/Steam d/p initiation signal is being reviewed for reliability since inadvertent actuation may contribute to overcooling events. It is possible that when the review is complete, this signal will be determined to be unnecessary due to the installation of automatic start on low OTSG level.

The automatic start on low OTSG level is delayed due to the need to obtain qualified level transmitters. The transmitters have been ordered and are currently undergoing qualification testing. (See also item 4 below)

The design details are complete but are on hold pending further review of the Feedwater/Steam d/p signal noted above.

4. Safety Grade OTSG Level Instrumentation and Control

Safety grade le l indicators independent of ICS/NNI qualified to IEEE-323 (1971), will be installed for restart. This system will be replaced with a safety grade system qualified to IEEE-323 (1974), in the long term. The long term level indication system is being used to generate the signals necessary for the safety grade automatic OTSG level control system. The control system is currently being designed; however, the level transmitters have been ordered and are currently undergoing qualification.

We are participating with several other utilities in an effort to obtain qualified differential pressure transmitters to meet the requirements of IEEE-323 (1974). Quarterly each utility receives an allotment of transmitters. The number of transmitters needed by each utility depends on the number of modifications required to be implemented and may exceed the quarterly allotment. This modification is competing with item 3 above for level transmitters. We are currently reviewing the designs for this modification and item 3 in the hope of achieving further simplification. The design is expected to be completed during the first quarter of 1982.

5. Condensate Storage Tank Level Indication

Low Low level alarms are planned to replace the existing control grade system in the long term. The level transmitters have been ordered and the design is under development with with engineering expected to be complete by November 1981.