

ENCLOSURE 1

TMI-1 DSAR, Revision 2
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THREE MILE ISLAND UNIT 1



DEFUELED SAFETY ANALYSIS REPORT (DSAR)

REVISION 2, MARCH 2024

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CHAPTER 1 – INTRODUCTION AND SUMMARY

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The Final Safety Analysis Report was submitted in support of the application of Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company for a license to operate a nuclear generating station designated as Three Mile Island Nuclear Station Unit 1 (TMI-1). After approval by the State and Federal regulatory agencies, Metropolitan Edison, Jersey Central, and Pennsylvania Electric owned the Three Mile Island Nuclear Station Unit 1 on a 50/25/25 percent basis, respectively. Metropolitan Edison Company was responsible for the safe operation of the station. After the Three Mile Island Unit 2 (TMI-2) accident, reorganization included the formation of the GPU Nuclear Corporation. With the post TMI-2 accident reorganization, Metropolitan Edison delegated to GPU Nuclear Corporation the responsibility for all aspects of operations, design, procurement and modifications of TMI-1. Aid in the design, construction, testing, and startup of the unit was supplied principally by Gilbert Associates, Inc.; United Engineers & Constructors, Inc.; and the Babcock & Wilcox Company. Assistance was rendered by other consultants and suppliers as required.

In 1999, as part of the restructuring of the electric utility industry in the United States, TMI-1 was sold to AmerGen Energy Company, LLC. AmerGen is a limited liability company formed to acquire and operate nuclear power plants in the United States. In 2008, the operating license for TMI-1 was transferred to Exelon Corporation. By letter dated February 1, 2022, the NRC issued Amendment No. 302 approving the transfer of TMI-1 Renewed Facility License from Exelon Generation Company, LLC to Constellation Energy Generation, LLC (the licensee).

Three Mile Island Unit 1 is permanently shut down. The Defueled Safety Analysis Report is submitted in support of the Renewed Facility License to possess special nuclear material at the Three Mile Island Nuclear Station.

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1.2 DESIGN SUMMARY

1.2.1 SITE CHARACTERISTICS

The site is located on the Susquehanna River about 10 miles southeast of Harrisburg, Pennsylvania. It is characterized by a 2,000 foot minimum exclusion area boundary distance; a two mile radius low population zone; sound bedrock as a structural foundation; an ample supply of emergency power and favorable conditions of hydrology, geology, seismology, and meteorology. The land within a 10 mile radius of the site is used primarily for farming.

There are two airports within ten miles of the site. Harrisburg International Airport (formerly Olmsted State Airport) is located approximately 2.5 miles northwest of the site, and Capital City Airport (formerly Harrisburg-York State Airport) is located approximately eight miles west-northwest of the site.

1.2.2 FACILITY DESCRIPTION

The Unit is located on the northern most section of Three Mile Island near the east shore of the Susquehanna River in Dauphin County, Pennsylvania. The Three Mile Island Nuclear Station Unit 1 was a pressurizer water reactor design which was licensed to operate at a rated power level of 2568 MWt. The original nuclear steam supply system and the fuel for the first core and reloads were supplied by the Babcock & Wilcox Company. The replacement Once Through Steam Generators were supplied by AREVA NP Inc.

The general arrangement of major equipment and structures of the operating facility are Drawings 1E-120-01-001 (Site Plan), 1E-151-02-series (Turbine Building), 1E-153-02-series (Reactor Building), 1E-154-02-series (Auxiliary and Fuel Handling Buildings), 1E-155-02-series (Control Building), 1E-156-02-series (Intermediate Building), 1E-157-02-series (Diesel Generator Building) and 1E-168-02-series (Intake Screen and Pump House).

1.2.3 SPENT FUEL STORAGE

There is no new fuel on site. The spent fuel generated by TMI-1 operation from 1974 to 2019 (1665 irradiated fuel assemblies) is stored in the TMI-1 Independent Spent Fuel Storage Installation (ISFSI).

The spent fuel at the ISFSI is stored in the NAC International MAGNASTOR dry cask storage (DCS) system. The ISFSI storage pad is located within the ISFSI Protected Area (PA) south of TMI2. The ISFSI storage pad has a capacity for 49 casks, which can include spent fuel or Greater Than Class C (GTCC) waste. Use of the ISFSI for spent nuclear fuel is authorized by compliance with the conditions of the General License issued under 10 CFR 72, Subpart K and the conditions contained in the MAGNASTOR Certificate of Compliance 1031 Amendment 9, **Revision 1** (CoC 1031-9 **R1**). The TMI-1 10 CFR 72.212 Evaluation Report (Reference 4) documents compliance with 10 CFR 72 and MAGNASTOR CoC 1031-9 **R1**, as well as the evaluations that demonstrate acceptable performance during design basis normal and accident conditions.

1.2.4 FUEL ASSEMBLIES AND CONTROL COMPONENTS

1.2.4.1 Fuel Assemblies

The fuel is sintered low-enriched UO_2 cylindrical pellets. The pellets are clad in either Zircaloy-4, ZIRLO, or M5 (an NRC-approved, zirconium-based alloy) tubing and sealed by Zircaloy-4, ZIRLO, or M5 end caps, welded at each end. The clad, fuel pellets, end caps, and fuel support components form a fuel rod. Two hundred and eight fuel rods, sixteen control rod guide tubes, one instrumentation tube assembly, eight spacer grids, and two end fittings make up the basic fuel assembly. The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array. The center position in the assembly was reserved for instrumentation. For details about the individual fuel assembly components, see Appendix 1A.

1.2.4.2 Control Components

Control components such as Control Rod Assemblies (CRA), Axial Power Shape Rod Assemblies (APSRA), Burnable Poison Rod Assemblies (BPRA), and Orifice Rod Assemblies (ORA) are stored within fuel assemblies. Each control component assembly contains sixteen rods attached to a stainless steel spider. The control components are inserted into the fuel assembly guide tubes through the upper end fitting. For details about each type of control component, see Appendix 1A.

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1.3 PRINCIPAL ARCHITECTURAL AND DESIGN CRITERIA

The Three Mile Island Nuclear Station Unit 1 was designed and constructed taking into consideration the general criteria for nuclear power plant construction permits as listed in the proposed AEC General Design Criteria, dated July 1967. Many of these criteria are not applicable to a permanently defueled facility and are not described. In the following section, the general basis for how the relevant criteria are satisfied is summarized:

1.3.1 CRITERION 1 - QUALITY STANDARDS

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality Assurance programs, test procedures, and inspection acceptance levels used shall be identified. A showing of sufficiency and applicability of codes, standards, Quality Assurance programs, test procedures, and inspection acceptance levels used is required.

DISCUSSION

The TMI Quality Assurance Program is described in NO-DC-10, the Decommissioning Quality Assurance Program (DQAP). The DQAP provides the top level requirements applied to quality related items and activities during the decommissioning phase of the plant life. The DQAP is based on the applicable portions of 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix B to Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; 10 CFR Part 71, "Packaging and transportation of radioactive material," Subpart H, "Quality Assurance"; and 10 CFR Part 72, "Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related greater than Class C waste," Subpart G, "Quality Assurance."

1.3.2 CRITERION 2 - PERFORMANCE STANDARDS (Historical)

Those systems and components of Reactor Building facilities which are essential to the prevention of accidents, which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornados, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (1) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area, and (2) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

DISCUSSION

The systems and components essential to public safety have been designed to performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces or effects which might be imposed by natural phenomena. The designs are

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based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data, or upon the most severe conditions established based on synthetic analyses. In the **ISFSI** condition, the Reactor Building **and SFP are** no longer a primary concern. The focus is shifted to the spent fuel contained **within the ISFSI Casks** and the external event hazards applicable are described **within the TMI Unit 1 10 CFR 72.212 Evaluation Report**.

1.3.3 DELETED

1.3.4 CRITERION 4 - SHARING OF SYSTEMS

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

DISCUSSION

Three Mile Island Nuclear Station Unit 1 makes use of some facilities on a shared basis with Unit 2 (DSAR 1.4). None of the shared components are connected with safety features or control systems of either nuclear steam supply systems.

1.3.5 CRITERION 5 - RECORDS REQUIREMENTS

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the licensee or under corporate control throughout the life of the reactor.

DISCUSSION

Records will continue to be retained at TMI as described in the Decommissioning Quality Assurance Program (10 CFR 50, Appendix B, Criterion XVII).

1.3.6 DELETED

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1.3.7 CRITERION 17 - MONITORING RADIOACTIVITY RELEASE

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

DISCUSSION

Monitors are provided for the facility effluent discharge paths, and the facility environs as required to monitor activity that is released as a result of normal operation. The monitors provided for the unit and their functions are described in Chapter 4. Containment monitors, and accident and transient monitors are no longer applicable in a plant without nuclear fuel. Radioactive release monitoring is required to ensure compliance with 10 CFR 20, and the ODCM.

The operating plant liquid release path has been positively isolated. Liquid releases required to manage ground or rain water intrusion into areas with radioactive material will be sampled and analyzed to verify compliance with ODCM requirements prior to batch release. The ODCM does not require liquid effluent process monitors for this method.

1.3.8 CRITERION 18 - MONITORING FUEL AND WASTE STORAGE

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

DISCUSSION

Radiation monitors and alarms are provided in the fuel handling and auxiliary buildings as required to warn personnel of impending excessive levels of radiation or airborne activity. The radiation monitors provided for the unit and their functions are described in Chapter 4. **With all fuel in dry-cask storage, no monitoring of decay heat removal is required.**

1.3.9 DELETED

1.3.10 DELETED

1.3.11 CRITERION 68 - FUEL AND WASTE STORAGE SHIELDING

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10CFR20.

DISCUSSION

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Components and piping located within the reactor, fuel handling, and auxiliary buildings containing primary coolant or other fluids requiring it, are shielded by reinforced concrete sufficient to meet or exceed the requirements of 10CFR20 for radiation protection. DSAR section 4.3 specifies the design criteria for shielding throughout the unit and the design dose rates at various locations. The criteria comply with 10CFR20 limits, and this design bounds the shielding requirements for the defueled condition.

1.3.12 CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

DISCUSSION

All spent fuel storage and radioactive liquid waste processing and storage facilities for this unit are housed in **designed for** Class I reinforced concrete structures constructed to withstand the hypothetical aircraft incident.

1.3.13 CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluent, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactivity effluents to the environment. In all cases, the design for radioactivity control shall be justified: (1) on the basis of 10CFR20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur, and (2) on the basis of 10CFR100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

DISCUSSION

The design of the unit incorporates the means necessary to maintain control of releases of radioactive liquid, gas, and solid waste effluents such that these will be within 10CFR20 limits, except tritium, for normal operation and reasonable transient situations. Ample holdup capacity is provided in the unit for the retention of liquid, gaseous, and solid wastes to ensure that they can always be released in a controlled manner. All normal releases of radioactive are continuously monitored and controlled by the radiation monitoring systems described in Chapter 4. The means provided to control releases of radioactive liquid, and solid waste effluents, under normal and reasonable transient situations, are discussed in Chapter 4.

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1.4 INTERACTIONS WITH UNIT 2

- 1.4.1 Following the TMI-2 accident, TMI-2 was placed in TMI-2 POST-DEFUELING MONITORED STORAGE (PDMS). TMI-2 has a possession only license (POL), and is maintained in accordance with the Nuclear Regulatory Commission **requirements for the Decontamination and Dismantlement phase of Decommissioning** (Reference 1 and 2).
- 1.4.2 Three Mile Island Nuclear Station Unit 1 shares some facilities with Unit 2. A partial list of the shared components includes:
- a. Fire Protection System
 - b. Fuel Handling Building Crane
 - c. Industrial Waste Treatment Facility
 - d. Sewage Treatment Plant

None of the shared components are connected with safety features or control systems of either nuclear steam supply system.

TMI-1 facilities may be used for TMI-2 functions. These activities include use of the TMI-1 laboratory to analyze TMI-2 samples, use of tools and instruments having residual contamination from TMI-2 at TMI-1, use of common shipping containers and the TMI-1 trash compactor for radioactive waste from both units and consolidation of decontamination facilities (Reference 3).

1.4.3 Fuel Handling Building Environmental Barrier

TMI-1 and TMI-2 Fuel Handling and Auxiliary Building Ventilation Systems have been isolated to prevent potential leakage paths between buildings or systems and to isolate the Unit 1 refueling floor from the Unit 1 Auxiliary Building and from the Control Building (Reference DSAR 3.6.2).

1.5 CONCLUSIONS

The health and safety of the public and station operating personnel and reliability of equipment and systems have been among primary considerations in the design and construction of the unit.

The radioactive waste systems are designed to **safely handle** all waste prior to disposal as permitted by Nuclear Regulatory Commission regulations.

In consideration of the above circumstances and the station design as presented herein, it is concluded that the Three Mile Island Nuclear Station Unit 1 can be **maintained** in a safe manner and that the design will provide adequate protection to the public from any natural or mechanical events resulting in disablement of equipment.

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1.6 REFERENCES

1. TMI Unit 2 **Decommissioning** Safety Analysis Report |
2. TMI Unit 2 Post Shutdown Decommissioning Activities Report (PSDAR)
3. NRC letter to GPUN, January 10, 1990, the NRC staff agreed that the scope of the Condition of Operation provided in the October 2, 1985 NRC letter did not preclude the types of activities discussed in GPUN letter of August 30, 1989.
4. TMI-1 10CFR72.212, "TMI-1 10 CFR 72.212 Evaluation Report." |

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APPENDIX 1A – FUEL ASSEMBLY AND CONTROL COMPONENT DESCRIPTION

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APPENDIX 1A

FUEL ASSEMBLY AND CONTROL COMPONENT DESCRIPTION

1.0 FUEL ASSEMBLY

Fuel assembly information is historical.

All spent fuel has been transferred to the ISFSI and is stored in the NAC International MAGNASTOR dry cask storage (DCS) system.

1.1 GENERAL

All fuel assemblies are similar in concept and are mechanically interchangeable. The Mark B2 fuel assembly shown on Figure 1A-1 is typical of the design used in the initial loading of the core. It is of the canless type where the eight spacer grids, end fittings, and the guide tubes form the basic structure. Fuel rods are supported at each spacer grid by contact points integral with the wall of the cell boundary. The guide tubes are permanently attached to the upper and lower end fittings. Use of similar material in the guide tubes and fuel rods results in minimum differential thermal expansion. Select fuel rods may have been replaced by stainless steel filler rods in fuel assemblies with failed/degraded fuel rods, or in locations with degraded spacer grids. Fuel assembly components, materials, and dimensions are given in Table 1A-1.

Mark B4 Fuel assemblies introduced in Cycle 2 incorporated minor modifications to the end fitting primarily to reduce fuel assembly pressure drop and to increase holddown margin.

In Cycle 7, the Mark B4Z assembly was implemented which replaced the intermediate Inconel spacer grids with zircaloy grids.

As of Cycle 8, the Mark B8 assembly was implemented with several design improvements including:

- a) reconstitutable upper end fittings
- b) elimination of BPRA retainers
- c) annealed guide tubes
- d) higher burnup capability
- e) debris-resistant fuel rod lower end plugs
- f) zircaloy intermediate spacer grids

The Mark B8V (Cycle 9) added a shot-peened coiled holddown spring, grippable upper fuel rod end plugs and bullet-nosed lower fuel rod end plugs.

In Cycle 10, the Mark B9 assembly was introduced with a removable lower end fitting, lower guide tube bypass flow and an optimized fuel rod with larger pellet diameter to improve thermal performance.

In Cycle 11, the Mark B10 assembly (Figure 1A-2) was introduced with a new cruciform leaf-type FA holddown spring in the upper end fitting to replace the helical coil spring design for better reliability and increased holddown force and with an improved reconstitutable guide tube upper attachment nut.

In Cycle 14, the Mark B12 assembly was introduced with new design features including:

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- a) low corrosion, low growth M5 cladding and guide tubes
- b) fine mesh debris filter
- c) redesign cruciform holddown spring to reduce holddown force
- d) heavier uranium loading

In Cycle 17, the Mark B-HTP assembly (Figure 1A-3) was introduced with new design features including:

- a) low corrosion M5 top and intermediate HTP spacer grids
- b) Inconel bottom HMP spacer grid
- c) a no-direct-line-of-sight FUELGUARD debris filter

1.2 FUEL RODS

All fuel rods are internally prepressurized with helium. The fuel is in the form of sintered and ground pellets of low-enriched UO_2 . As of Cycle 10, fuel in selected rods may contain uranium-gadolinia ($\text{UO}_2/\text{Gd}_2\text{O}_3$) integral burnable poison. As of Cycle 13, all fuel rods contain enriched axial blankets where the top and bottom sections of pellets are of a lower U235 enrichment than the nominal central section. Pellet ends are dished to minimize differential thermal expansion between the fuel and cladding. Radial growth of the fuel during burnup is accommodated by the pellet porosity, radial clearance between the pellets and the cladding, and by a small amount of permanent strain in the cladding.

Below each fuel column is a spring spacer which axially locates the bottom of the fuel column and separates the fuel from the lower fuel rod end cap. This spring is designed to deflect at high column loads to prevent excessive axial strain in the cladding.

Above the fuel column is a spring spacer that separates the fuel from the fuel rod upper end cap. This spacer maintains the fuel column in place during shipping and handling. This spacer also provides radial fuel rod cladding support.

Depending on the specific Mark B fuel rod design, metallic and ceramic spacers may be located between the fuel pellets and the spring spacers to thermally insulate and separate fuel pellets from the springs.

Fission gas release from the fuel is vented to voids within the pellets to the radial gap between the pellets and the cladding, and to the void spaces at top and bottom ends of the fuel rods.

1.3 SPACER GRIDS

Spacer grids are constructed from strips which are slotted and fitted together in egg-crate fashion. Each grid has 32 strips, 16 perpendicular to 16, which form the 15 x 15 lattice. The square walls formed by the interlaced strips provide support for the fuel rods in two perpendicular directions.

1.4 LOWER END FITTING

The fuel assembly weight is supported by the lower end fitting. The lower ends of the fuel rods rest on the grid of the lower end fitting. Penetrations in the lower end fitting are provided for attaching the control rod guide tubes and for access to the instrumentation tube. Depending on the specific Mk B fuel assembly design, lower end fittings may contain a fine mesh debris filter, as the Mark B12 fuel design, or a no-direct-line-of-sight debris filter, as the Mark B-HTP fuel design.

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1.5 UPPER END FITTING

The upper end fitting (UEF) provides means for coupling to the fuel handling equipment. An identifying number on each upper end fitting provides positive identification.

Integral with each upper end fitting are a holddown spring and retainer. These components have no function in the defueled mode.

Penetrations in the upper end fitting grid are provided for the guide tubes.

1.6 GUIDE TUBES

The zircaloy or M5 guide tubes provide continuous guidance to the control rods when inserted in the fuel assembly and provide the structural continuity for the fuel assembly. Welded to each end of a guide tube are flanged and threaded sleeves, which secure the guide tubes to each end fitting by lock-welded or crimped nuts depending on whether the design is reconstitutable. Transverse location of the guide tubes is provided by the spacer grids.

1.7 INSTRUMENTATION TUBE

This zircaloy or M5 tube served as a channel to guide, position, and contain the incore instrumentation within the fuel assembly. It is retained axially at the lower end fitting by a retainer sleeve.

1.8 SPACER SLEEVES

On the Mark-B design, the spacer tube segments fit around the instrument tube between spacer grids and prevent axial movement of the spacer grids during primary coolant flow through the fuel assembly. In Cycle 12, a design change was made to the Mark B10 assembly spacer grid retention system that allows the spacer sleeves to positively capture the grid insert tubes above and below the grid to improve prevention of axial grid movement. Spacer sleeves were eliminated on the Mark-B-HTP design, which prevents axial grid movement by welding the spacer grids to guides tubes.

2.0 RECAGING/RECONSTITUTION

To allow damaged fuel assemblies to be discharged from the Spent Fuel Pool into a cask, methods have been developed to remove fuel rods from Mark B fuel assemblies through either reconstitution or recaging.

Reconstitution is used to remove leaking or suspect fuel rods from Mark B fuel assemblies. It is accomplished using a specially designed assembly with a removable upper end fitting. When defective rods are detected and need to be removed, the UEF is unfastened, the rods removed and replaced with dummy fuel rods (stainless steel or zircaloy) and the UEF refastened.

Recaging is used when the original assembly structure is no longer usable. The UEF is removed, all reusable intact fuel rods are transferred to a new assembly structural cage and a new UEF is attached. The damaged rods remain in the original assembly structure and are replaced in the new assembly by dummy fuel rods. Recaging can be performed for both older assembly designs (non-reconstitutable) and the newer, reconstitutable designs.

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2.1. CYCLE 10 FUEL DEGRADATION/REPAIR

During the 11R refueling outage (1995), fuel inspections of all 177 Cycle 10 fuel assemblies indicated a total of nine (9) failed fuel rods in four (4) first-burn (Batch 12) assemblies. All failed rods were in the peripheral rows of the fuel. Visual inspections revealed that these rods, as well as adjacent rods that had no through-wall failure indications, had a distinctive crud pattern (DCP) exhibiting an intense dark and light mottled appearance. Eddy current (E/C) inspections of intact intense DCP rods showed that some had indications of up to 70% wall-thinning in the upper spans while others had no such indications. Rods with less intense DCP showed no wall-thinning.

All fuel assemblies scheduled for reinsertion in Cycle 11 were visually inspected. Those with rods with intense DCP were taken to the fuel repair station, the upper end fitting removed, and the DCP rods eddy current inspected for wall-thinning. No non-peripheral rods were found with the DCP. All peripheral rods with an E/C indication greater than 0% wall-thinning were removed and replaced with interior donor rods of similar burnups, from the same assembly or (for several rods) from another Batch 12 assembly, to avoid power peaking mismatches. The interior donor rods were replaced with stainless steel dummy rods to fill the water holes.

A total of 105 Batch 12 peripheral fuel rods in 23 fuel assemblies had some wall-thinning indication; a total of 290 peripheral rods were E/C inspected in 28 assemblies. A total of 35 non-peripheral rods were inspected and showed no DCP and no wall-thinning.

3.0 LEAD TEST ASSEMBLIES

3.1 WESTINGHOUSE LEAD TEST ASSEMBLIES

To qualify an alternative fuel supplier for TMI-1, four LTA's had been inserted with the Cycle 9 reload (1991) to be burned for three consecutive cycles.

The LTA incorporates features adapted from current Westinghouse advanced fuel assemblies which include:

- 15 x 15 array
- 0.422 inch OD fuel rod design
- Leaf-type holddown springs
- Removable improved-design top nozzle
- Inconel upper and lower spacer grids
- Rotated alternate mid-grids
- ZIRLO mid-grids with flow mixing vanes
- ZIRLO fuel rod cladding
- ZIRLO guide tubes/instrument tube
- Debris-resistant fuel rod lower end plug
- Debris-filter, removable bottom nozzle

LTA characteristics with respect to the BWFC Mark B fuel are:

- Smaller fuel rod diameter
- Slightly lower uranium loading
- Better uranium utilization

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- Slightly higher reactivity
- Lower U-235 enrichment
- Compatible hydraulics
- Matching spacer grid elevations
- Somewhat higher flow resistance
- Somewhat higher pressure drop
- Greater DNBR margin

During the 10R outage (1993) UT fuel inspections determined that a total of 24 LTA fuel rods had failed during Cycle 9. The LTA's were discharged and a root cause investigation was performed. Visual inspections and flow testing determined that the cause of failure was self-excited flow-induced vibration (FIV) similar to that which had been found at Salem-2 and Beaver Valley-1 early in 1993. The FIV led to grid-to-rod fretting failures. Westinghouse had developed a design change to mitigate the FIV for their 17 x 17 fuel. The change consisted of 90-degree rotation of alternate intermediate spacer grids.

A prototype LTA with rotated grids was tested at TMI operating flows by Westinghouse. Results showed that the higher-mode, larger-amplitude vibrations of concern exhibited by the original aligned-grid (non-rotated) LTA were eliminated. To improve corrosion performance at long residence times the advanced zirconium alloy ZIRLO was used in place of the original Zircaloy 4 material for assembly components and rod cladding. Four Westinghouse replacement Lead Test Assemblies (LTA) were inserted with the Cycle 11 reload fuel batch with the intent of three cycles of operation.

In Cycle 12, two of the replacement LTAs developed fuel defects in their second cycle of operation. All four of the LTAs were removed from the Cycle 13 reload design, thereby discontinuing the alternate supply program.

3.2 ADVANCED CLADDING ASSEMBLIES

As TMI cycle lengths, fuel burnups and fuel in-core residence time increased and the RCS chemical environment became more aggressive (e.g., higher lithium concentrations), enhanced corrosion of the Zr-4 fuel rod cladding and other assembly components became a concern. To help ensure that fuel would continue to meet all design and performance criteria for future cycles, a program was developed with BWFC to irradiate two advanced zirconium-based alloy claddings designed to have greater corrosion resistance than Zr-4 claddings. These non-Zircaloy alloys, designated M4 and M5, had been successfully irradiated in European reactors and in McGuire-1.

The TMI program consisted of a total of 16 demonstration rods, 8 each of M4 and M5. Four rods of each alloy were placed in the peripheral rows of two Cycle 11 Batch 13F MkB10-Gd fuel assemblies. They were scheduled to be burned for three 2-year cycles, Cycles 11, 12 and 13, with a projected residence time of 1950 ± 45 EFPD and rod burnups to 53 GWD/T.

The demonstration program was extended to Cycle 14 as four of the thrice-burned M5-clad rods were reconstituted into a host assembly to achieve burnups in excess of 62 GWd/mtU.

4.0 CONTROL COMPONENTS

4.1 CONTROL ROD ASSEMBLY (CRA)

TMI-1 DSAR

Each CRA (Figure 1A-4) has 16 control rods, a stainless steel spider, and a female coupling. The 16 control rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly, all nuts are lock welded. The CRAs and guide tubes are designed with adequate flexibility and clearances to permit freedom of motion within the fuel assembly guide tubes throughout the stroke.

Each control rod has a section of neutron absorber material. The absorber material is an alloy of silver-indium-cadmium (Ag-In-Cd). The initial-core CRAs were replaced in Cycles 9 and 10 with the new Extended-Life CRA. The ELCRA is clad in Inconel 625 tubing. Inconel end pieces are welded to the tubing to form a water and pressure-tight container for the absorber material. The tubing provides the structural strength of the control rods and prevents corrosion of the absorber material. A tube spacer similar to that in the fuel assembly is used to prevent absorber motion within the cladding during shipping and handling and to permit differential expansion in service.

Principal data pertaining to the CRA are given in Table 1A-2.

Because the Ag-In-Cd alloy does not yield gaseous products under irradiation, internal pressure is not generated within the clad.

Because of their length and the possible lack of straightness over the entire length of the rod, some interference between control rods and the fuel assembly guide tubes is expected. However, the parts involved, especially the control rods, are flexible and only small friction drag loads result. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. Consequently, CRAs will not encounter significant frictional resistance to their motion in the guide tubes.

4.2 AXIAL POWER SHAPING ROD ASSEMBLY (APSRA)

Each APSRA (Figure 1A-5) has 16 axial power shaping rods, a stainless steel spider, and a female coupling. The 16 rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly, all nuts are lock welded.

When the APSRA is inserted into the fuel assembly, it is guided by the guide tubes of the fuel assembly. The APSRAs are designed to permit maximum conformity with the fuel assembly guide tube throughout travel.

The APSRA loaded in the initial cycle have a 36 inch section of neutron absorber material that is an alloy of silver-indium-cadmium (i.e., black absorber) and is clad in cold-worked, type 304 stainless steel tubing. The APSRAs were replaced in Cycle 6 with an improved design where each axial power shaping rod has a 63 inch section of Inconel neutron absorber material (i.e., gray absorber) clad in cold-worked, type 304 stainless steel tubing. This tubing provides the structural strength for the axial power shaping rods and prevents corrosion of the absorber material. The absorber section is sealed by both an internal plug and an end plug shown on Figure 1A-5. The section of tubing above the absorber is vented so it is always filled with borated water and consequently the pressure differential across the tube wall is negligible.

Because the Inconel alloy does not yield gaseous products under irradiation, internal pressure is not generated within the clad.

TMI-1 DSAR

As of Cycle 12, the APSRA design was changed to ensure proper latchup with the Mark-B10 fuel assembly and to increase the rod lifetime. The latchup improvement consisted of lengthening the female coupling and related dimensional changes. The extended lifetime was achieved by adjusting the internal rod plenum volume.

Pertinent data on the APSRA are given in Table 1A-3.

Because of their great length and unavoidable lack of straightness, some slight mechanical interference between axial power shaping rods and the fuel assembly guide tubes must be expected. However, the parts involved are flexible and result in very small friction drag loads. Similarly, thermal distortions of the rods are small because of the low heat generation and adequate cooling. Consequently, the APSRAs will not encounter significant frictional resistance to their motion in the guide tubes.

4.3 BURNABLE POISON ROD ASSEMBLY (BPRA)

Each BPRA (Figure 1A-6) has up to 16 burnable poison rods, a stainless steel spider. The rods are attached to the spider. The BPRA is inserted into the fuel assembly guide tubes through the upper end fitting. For all Mark B4-type fuel assemblies, retainers were used to lock the BPRA into the fuel assembly while in the reactor. Starting with the Mark B8 fuel assembly, an improved BPRA spider design was used that eliminated the need for a separate retainer.

Each burnable poison rod has a section of sintered $\text{Al}_2\text{O}_3\text{-B}_4\text{C}$ pellets which serves as burnable poison. The length of this section is varied depending on fuel stack length and the presence of axial blankets to enhance neutron efficiency and to control axial power peaking. The burnable poison is clad in cold-worked Zircaloy-4 tubing and Zircaloy-4 upper and lower end pieces. The end pieces are welded to the tubing to form a water and pressure-tight container for the absorber material. The Zircaloy-4 tubing provides the structural strength of the burnable poison rods.

Pertinent data on the BPRA are given in Table 1A-4.

4.4 ORIFICE ROD ASSEMBLY (ORA)

Each ORA (Figure 1A-7) has 16 orifice rods, a stainless steel spider, and a coupling mechanism. The coupling mechanism provides a means for positive coupling between the ORA and the fuel assembly holddown latch when the orifice rods are inserted into the fuel assembly. The necked-down section of the rod permits lateral movement in order to facilitate the installation of the orifice assembly into the guide tubes in the fuel assembly. The ORA served to limit bypass flow through empty guide tubes.

Pertinent data on the ORA is given in Table 1A-5.

Two ORA have been modified to become Regenerative Neutron Sources (RNS). The RNS assembly consists of eight source rods, 140 inches long, and eight orifice rods, 10 inches long. The source assemblies are stored within selected fuel assemblies.

TMI-1 DSAR

TABLE 1A-1
(Sheet 1 of 1)

FUEL ASSEMBLY COMPONENT DIMENSIONS ^(a)

<u>Item</u>	<u>Material</u>	<u>Dimensions (inches)</u>
<u>Fuel Rod</u>		
Fuel	UO ₂ sintered pellets ^(b)	0.3735 (diameter)
Fuel Clad	M5	0.430 OD x 154.075 long
Cladding Thickness		0.025
Fuel Rod Pitch		0.568
Active Fuel Length		143.0
<u>Fuel Assembly</u>		
Overall Length		165.835
Control Rod Guide tube wall	M5	0.530 OD x 0.016
Instrumentation tube (Mark-B-HTP)	M5	0.493 OD x 0.400 ID for bottom 135" of tube)
End Fittings	Stainless Steel (castings)	
Spacer Grid Bottom: Inconel-718 Intermediate/Top: M5		<u>Interior Rib</u>
		<u>Exterior Rib</u>
		0.0125
		0.025
		0.014
		0.026

^(a) Data for Mark-B-HTP Fuel Assemblies.

^(b) 96% theoretical density fuel (undensified and cold).

TMI-1 DSAR

TABLE 1A-2
(Sheet 1 of 1)

CONTROL ROD ASSEMBLY DATA

<u>Item</u>	<u>Data</u>
Number of control rods per assembly	16
Outside diameter of control rod, inches	0.441
Cladding thickness, inches	0.0225
Cladding material	Inconel 625
End plug material	Inconel 625
Spider material	SS Grade CF3M
Poison material	80% Ag, 15% In, 5% Cd
Female coupling material	Type 304 SS, annealed
Length of poison section, inches	139
Stroke of control rod, inches	139

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TABLE 1A-3
(Sheet 1 of 1)

AXIAL POWER SHAPING ROD ASSEMBLY DATA ^(a)

<u>Item</u>	<u>Data</u>
Number of axial power shaping rods per assembly	16
Outside diameter of axial power shaping rod, inches	0.440
Cladding thickness, inches	0.027
Cladding material	Type 304 SS, cold-worked
Plug material	Type 304 SS, annealed
Poison material	100% Inconel
Spider material	SS, Grade CF3M
Female coupling material	Type 304 SS, annealed
Length of poison section, inches	63
Stroke of control rod, inches	139

^(a) Data for improved gray absorber APSRA design.

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TABLE 1A-4
(Sheet 1 of 1)

BURNABLE POISON ROD ASSEMBLY DATA

<u>Item</u>	<u>Data</u>
Number of burnable poison rods per assembly	16
Outside diameter of burnable poison rod, inches	0.430
Cladding thickness, inches	0.035
Cladding material	Zircaloy-4, cold-worked
End plug material	Zircaloy-4, annealed
Poison material	B ₄ C in Al ₂ O ₃ matrix
Length of poison section, inches	123.2 ^a
Spider material	SS, Grade CF3M
Coupling mechanism material	Type 304 SS, annealed

^a Increased from 121" in Cycle 14 with introduction of Mark B12 fuel design.

TMI-1 DSAR

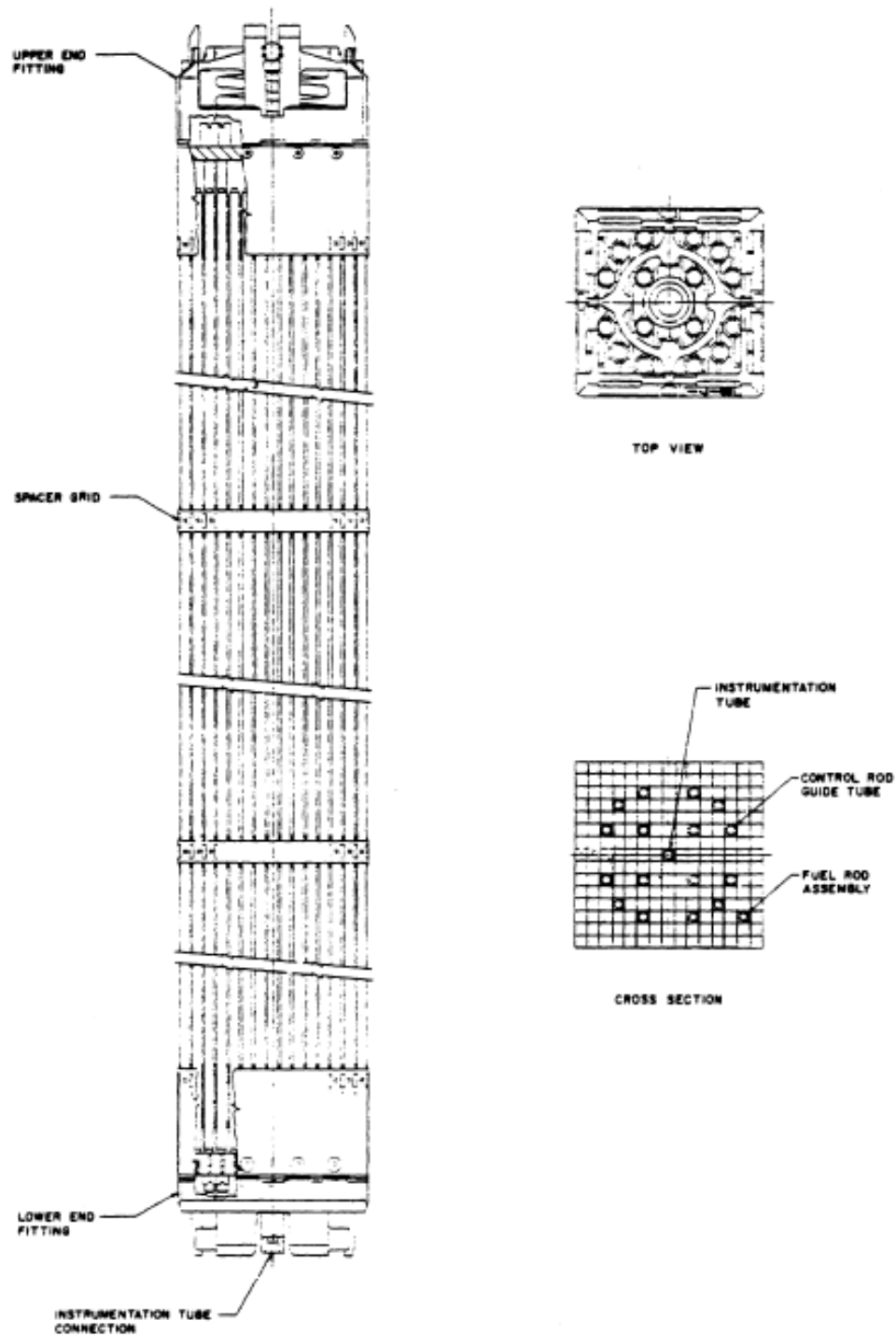
TABLE 1A-5
(Sheet 1 of 1)

ORIFICE ROD ASSEMBLY DATA

<u>Item</u>	<u>Data</u>
Number of orifice rods per assembly	16
Outside diameter of orifice rod, inches	0.480
Orifice rod material	Type 304 SS, annealed
Spider material	SS, Grade CF3M
Coupling mechanism material	Type 304 SS, annealed, and 17-4PH, Condition H 1100

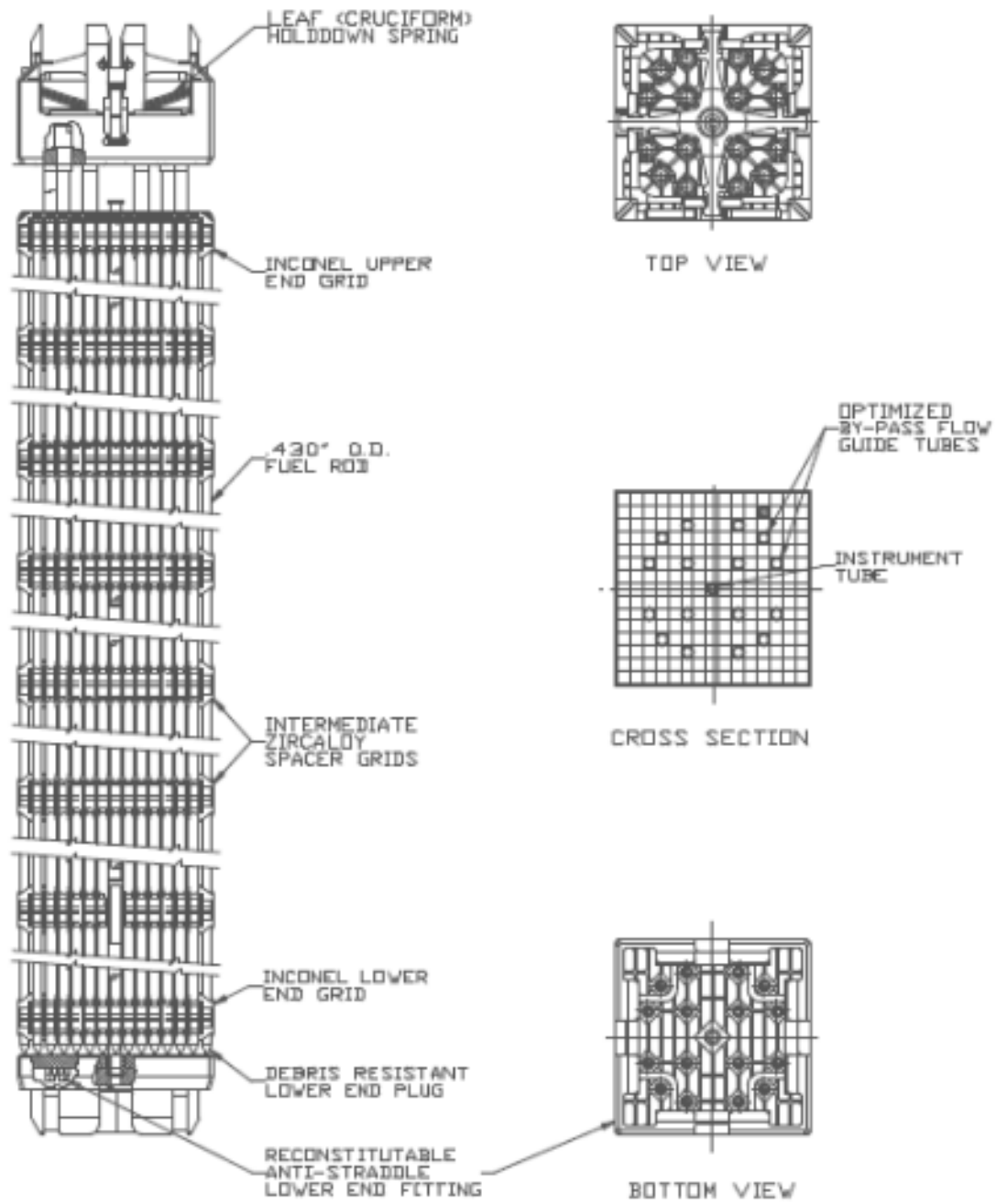
TMI-1 DSAR

FIGURE 1A-1, MARK B2 FUEL ASSEMBLY



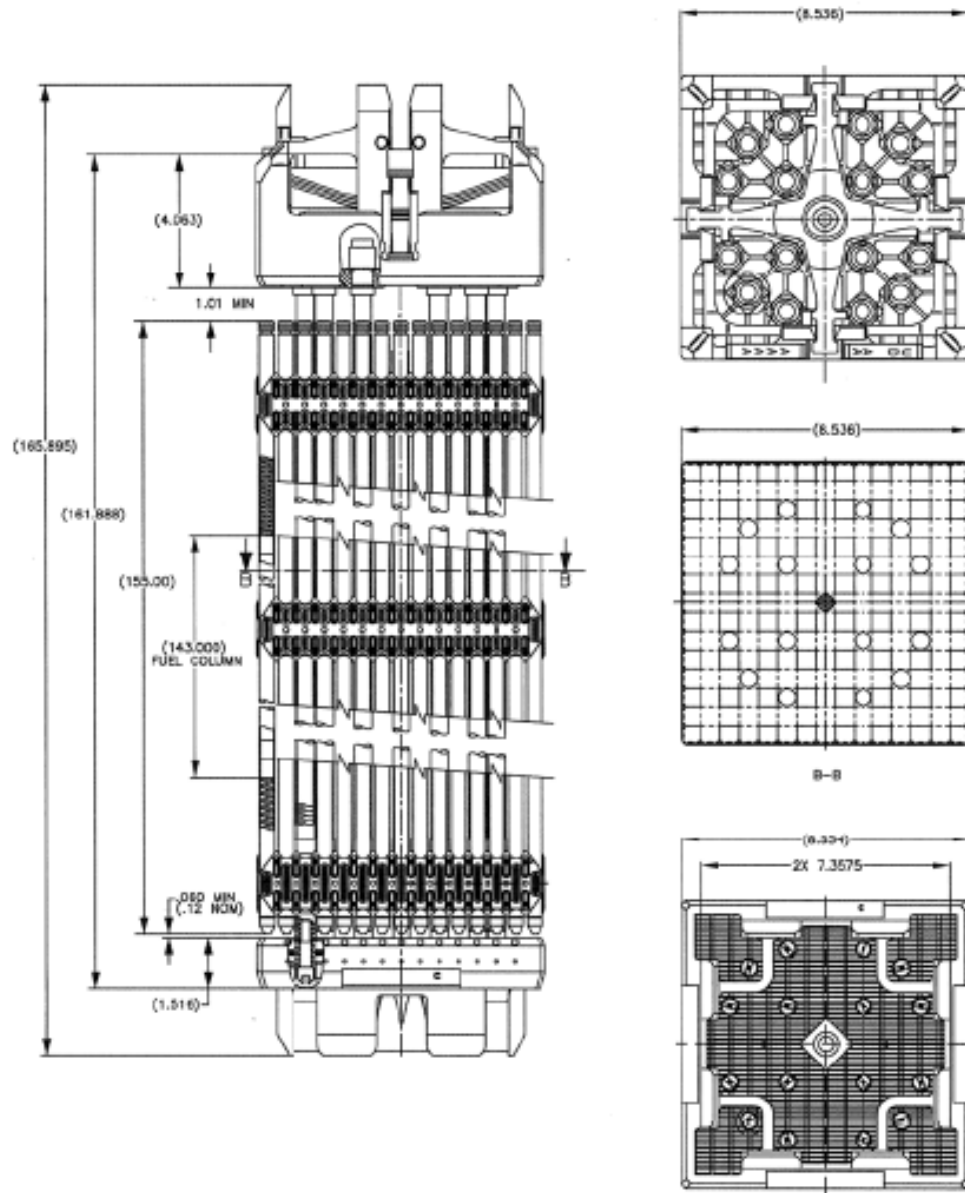
TMI-1 DSAR

FIGURE 1A-2, MARK B10 FUEL ASSEMBLY



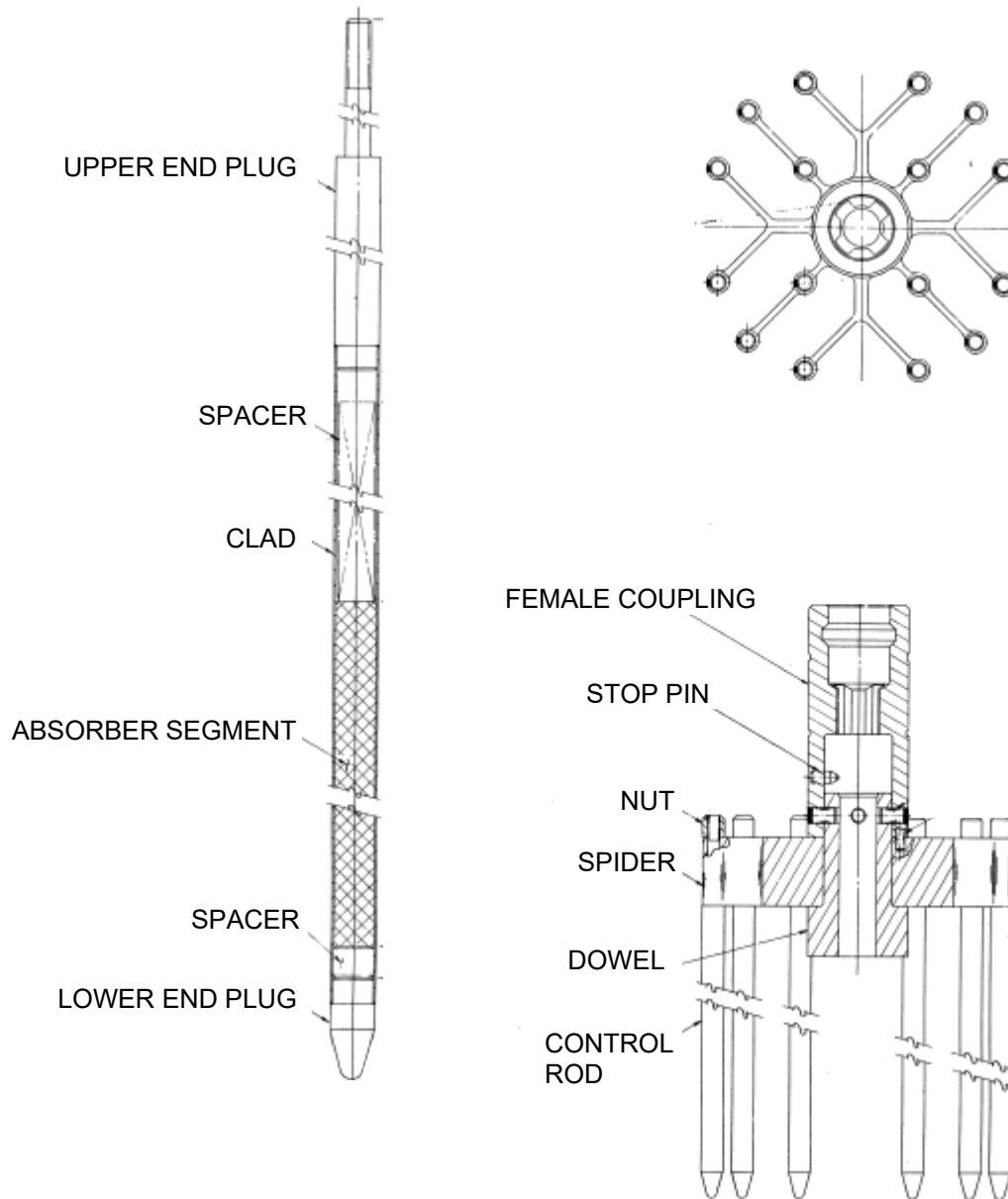
TMI-1 DSAR

FIGURE 1A-3, MARK B-HTP FUEL ASSEMBLY



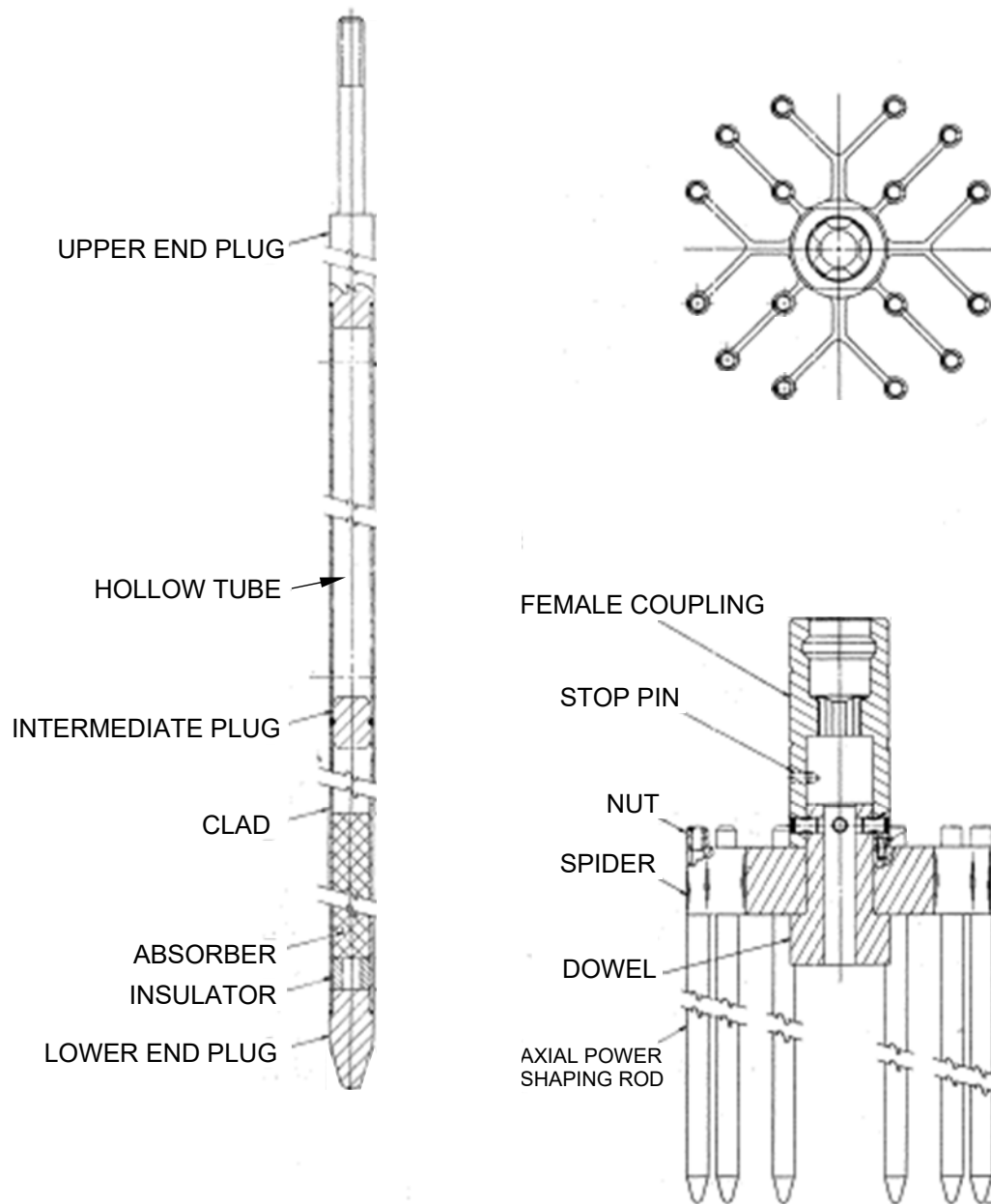
TMI-1 DSAR

FIGURE 1A-4, CONTROL ROD ASSEMBLY



TMI-1 DSAR

FIGURE 1A-5, AXIAL POWER SHAPING ROD ASSEMBLY



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FIGURE 1A-6, BURNABLE POISON ROD ASSEMBLY

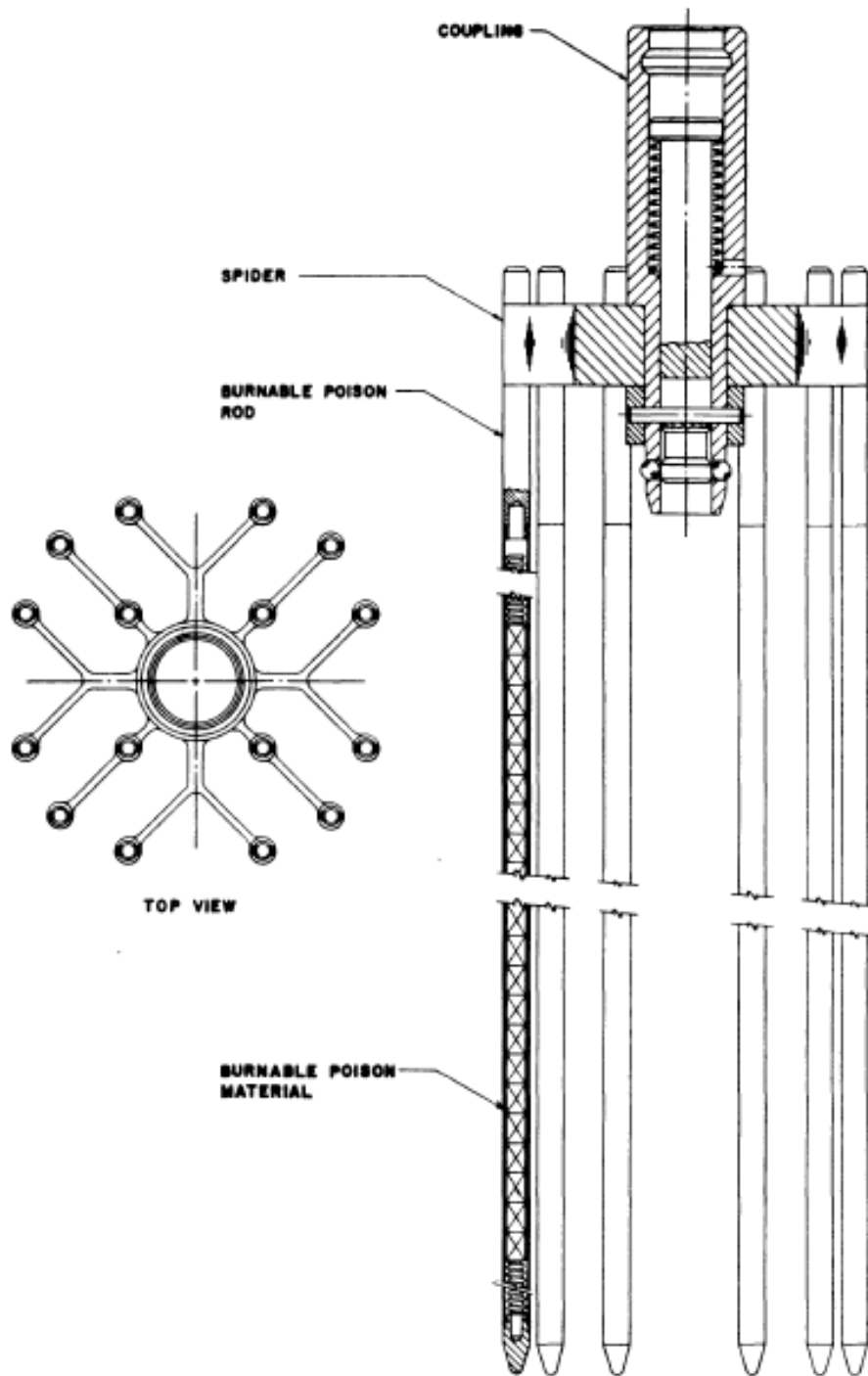
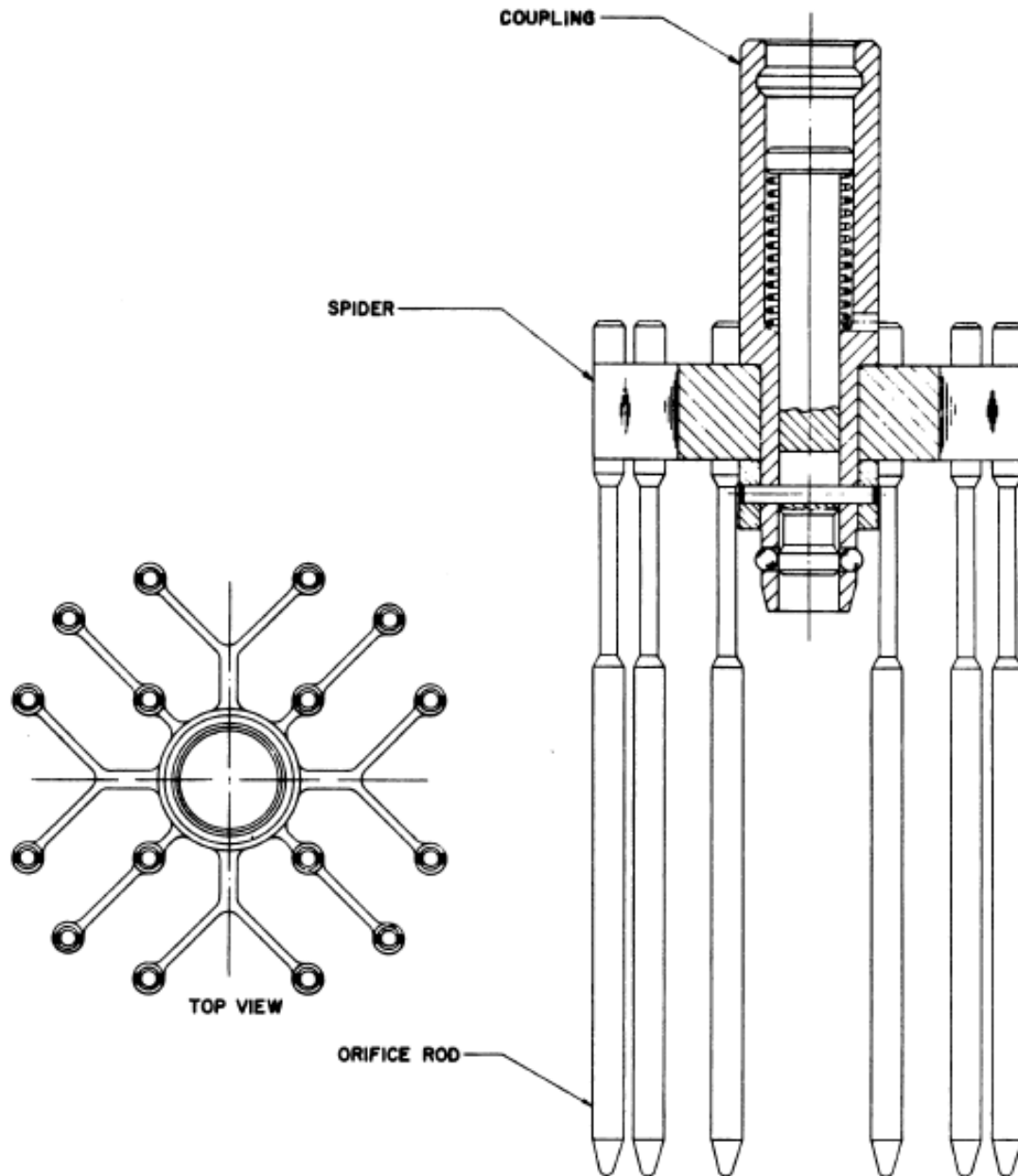


FIGURE 1A-7, ORIFICE ROD ASSEMBLY



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CHAPTER 02 – SITE AND ENVIRONMENT

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2.0 SITE AND ENVIRONMENT

2.1 GENERAL DESCRIPTION AND LOCATION

2.1.1 SITE LOCATION

Three Mile Island is located approximately 2.5 miles south of Middletown, Pennsylvania, at longitude 76°43'30" west and at latitude 40°09'15" north (approximate midpoint of the station). It is one of the largest of a group of several islands in the Susquehanna River and is situated about 900 ft from the east bank. It is elongated parallel to the flow of the river, with its longer axis oriented approximately due north and south. The island is about 11,000 ft in length and 1700 ft in width.

The southeasterly flowing Susquehanna River makes a sharp change in direction, to nearly due south, in the vicinity of Middletown. After this directional change just north of Three Mile Island, the river widens to approximately 1.5 miles.

The station is located on Three Mile Island, which is situated in the Susquehanna River upstream from York Haven Dam, in Londonderry Township of Dauphin County, Pennsylvania, about 2.5 miles north of the southern tip of Dauphin County, where Dauphin is coterminous with York and Lancaster Counties. Its location with respect to regional topographic and cultural features is shown on Figure 2.1-1 and with respect to local features on Figure 2.1-2.

2.1.2 SITE DESCRIPTION

Figure 2.1-3 shows Three Mile Island and the designated Exclusion Area. For accident evaluations, the distance to the Exclusion Area in each direction is used. Those distances may be determined from Figure 2.1-3.

2.1.2.1 Exclusion Area Control

The exclusion area is a 2,000 foot radius from a point equidistant between the center of the TMI-1 and TMI-2 Reactor Buildings. It includes portions of Three Mile Island, the river surface around it, and a licensee owned portion of Shelley Island, as shown on Figure 2.1-3. The site is located on the northern third of Three Mile Island. The licensee directly owns the TMI-1 site and owns or controls all land within the exclusion area. The Exclusion Area Agreement by and among the GPU companies and the licensee provides the licensee with authority, within those parts of the exclusion area for TMI-1 owned and controlled by the GPU companies, to determine and control all activities in the exclusion area, including exclusion of personnel and property from the area, to the extent necessary to comply with the requirements of 10 CFR 100. The licensee owns and controls the remaining portion of Three Mile Island.

York Haven Dam, immediately to the south of the site, has no locks. Thus, no commercial river traffic exists in the area. In the event of an emergency where control of the river in the exclusion area is recommended, the licensee will contact PEMA who will be responsible for controlling or restricting boat traffic in the area to prevent access within the exclusion area.

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2 1.2.2 Boundaries for Establishing Effluent Release Limits

The boundary line forming the base for the Technical Specification Limits on the release of gaseous effluents will be as shown on Figure 2.1-3. This boundary line includes portions of the Susquehanna River surface between the east bank of the river and Three Mile Island and between Three Mile Island and Shelley Island. Land surface included within this boundary line is owned or controlled by the licensee. Access to the exclusion area is readily controlled as stated in Subsection 2.1.2.1.

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2.2 POPULATION

2.2.1 POPULATION WITHIN 10 MILES

The population within a 10 mile radius of TMI-1 is shown on Figures 2.2-1 through 2.2-5. The current population figures within the 10-mile radius are based upon the 1990 census data, which are the most recent available. The population projections, by decade through 2010, were based on information obtained from State and county planning boards, where available, or on projections estimated from past growth. Note also that the projections were based on 1980 census data.

2.2.2 POPULATION WITHIN 50 MILES

The population within a 50 mile radius is shown on Figures 2.2-6 through 2.2-10. Population figures are based upon the 1980 census data, which are the most recent available (population within 10 miles based on 1990 census data). The population projections, by decade through 2010, were based on information obtained from State and county planning boards where available or on projections estimated from past growth. Counties and communities within a 50 mile radius of TMI are shown on Figure 2.2-11. The populations shown are from the 1990 census data.

2.2.3 LOW POPULATION ZONE

The low population zone for Three Mile Island Unit 1 was chosen as a 2 mile radius from the station. The zone was based on a low 1980 population density of approximately 1805 people in the 2 mile radius together with the highway facilities available for evacuation within a reasonable time, should it become necessary. The current population by sector is shown on the 0 to 10 mile population wheel, Figure 2.2-3. The population in the LPZ has decreased from 2539 (1980 census) to 1598 (1990 census).

2.2.4 TRANSIENT POPULATION

There are no schools or major industries within the low population zone; thus, no significant shift in population is expected to occur on a daily basis in the area.

There is some seasonal shift in population within approximately 5 miles of the station since there are over 100 summer cabins on the islands in the Susquehanna River within 5 miles of the site. The seasonal population has been included in the 0 to 10 mile population distribution described above.

Additional transient shifts in population may occur during the summer due to boating activity on the Susquehanna River in the vicinity of the plant site.

2.2.5 POPULATION CENTER

The nearest population center, as defined in 10CFR100, is Harrisburg, with a 1990 population of 52,376. The center of Harrisburg is located 12 miles northwest of the site.

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2.2.6 PUBLIC FACILITIES AND INSTITUTIONS

Public facilities and institutions, including county location and name within a 10 mile radius of Three Mile Island are given in Table 2.2-2.

The following is a summary of the facilities and institutions within a 10 mile radius: (Information based on the 1990 census data)

a. Schools

There are 65 public, private, and parochial schools within 10 miles: 13 in York County; 3 in Cumberland County; 39 in Dauphin County; and 10 in Lancaster County.

The nearest school to the site is the Northumberland Center (Capital Area Intermediate Unit), located in Dauphin County. The school is approximately 13,000 feet from the site.

b. Recreation

Seventeen recreational facilities have been identified within the 10 mile emergency planning zone; 7 in York County; and 10 in Dauphin County.

The nearest public recreational facility is the boat launching facility constructed by Metropolitan Edison Company approximately 8000 feet north from the site in Dauphin County. For more details see Appendix 3 of the Evacuation Travel Time Estimate for the TMI Nuclear Generating Station Emergency Planning Zone, (i.e., Transient and Special Facility Population Data).

c. Hospitals, Nursing Homes, and Similar Facilities

There are 13 facilities located within the 10 mile emergency planning zone, the largest being the Milton S. Hershey Medical Center of the Pennsylvania State University. This hospital and teaching institution is located approximately 10 miles from the site. For more details see Appendix 3 of the Evacuation Travel Time Estimate for the TMI Nuclear Generating Station Emergency Planning Zone, (i.e., Transient and Special Facility Population Data).

2.2.7 BUSINESS FACILITIES WITHIN 5 MILES

Table 2.2-1 lists all industrial and manufacturing facilities within 5 miles of the site. This information is based on the 1990 census data (Reference 29).

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TABLE 2.2-1
(Sheet 1 of 1)

INDUSTRIAL AND MANUFACTURING FACILITIES WITHIN FIVE MILES OF SITE

1. Chloe Textiles
2. Intercon Systems
3. Zeager Brothers, Inc.
4. Olivetti Supplies, Inc.
5. York Haven Power Co.
6. DIE-TECH
7. Brunner Island Station
8. LASCO Bathware
9. Fleetwood
10. Tyson Warehouse
11. American Air Filter

Reference: 1990 Evacuation Travel Time Estimates for Three Mile Island Nuclear Generating
Station Plume Exposure Pathway Emergency Planning Zone

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Table 2.2-2
(Sheet 1 of 3)

PUBLIC FACILITIES & INSTITUTIONS WITHIN A 10 MILE RADIUS

LANCASTER COUNTY

Donegal School District

Maytown Elementary School

Elizabethtown Area School District

Elizabethtown Area High School
Elizabethtown Area Middle School
Bainbridge Elementary School
East High Elementary School
Fairview Elementary School
Millroad Elementary School
Rheems Elementary School
St. Peter Parochial School
Mt. Calvary Christian Center

Elizabethtown College

YORK COUNTY

Northeastern School District

Northeastern Senior High School
Northeastern Junior High School
Conewago Elementary School
Mount Wolf Early Learning Center
Orendorf Elementary School
York Haven Elementary School

West Shore School District

Fairview Elementary
Fishing Creek Elementary
Mount Zion Elementary
Newberry Elementary
Red Land High School
Red Mill Elementary
The Circle School

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Table 2.2-2
(Sheet 2 of 3)

PUBLIC FACILITIES & INSTITUTIONS **WITHIN A 10 MILE RADIUS** **DAUPHIN COUNTY**

Middletown Area School District

Alice Demey Elementary School
George D. Mansberger Elementary School
John C. Kunkel Elementary School
Lyll J. Fink Elementary School
Middletown Senior High School
George W. Feaser Middle School
Middletown Christian School
Seven Sorrows BVM School

Lower Dauphin Area School District

Conewago Township Elementary School
Londonderry Township Elementary School
A.B. Nye Elementary School
E.Z. Price Elementary School
South Hanover Township Elementary School
Lower Dauphin Junior High School
Lower Dauphin Senior High School

Derry Township School District

M.S. Hershey Middle School
Hershey Elementary School
Hershey Senior High School
Mountain View Christian School
St. Joan of Arc School
Milton Hershey School

Steelton-Highspire School District

Steelton-Highspire Elementary School
Steelton-Highspire Junior/Senior High School
St. John Newmann Consolidated School

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Table 2.2-2
(Sheet 3 of 3)

PUBLIC FACILITIES & INSTITUTIONS **WITHIN A 10 MILE RADIUS**

DAUPHIN COUNTY (cont'd)

Central Dauphin School District

Paxtang Elementary School
Chamber Hill Elementary School
Rutherford Elementary School
Southside Elementary
Tri-Community Elementary School
Swatara Junior High School

Town and Country Day School
St. Catherine Laboure School
Lawnton

Capital Area Intermediate Units

Northumberland Center
Camp Harris
Oberlin School
Aspin Center

The Pennsylvania State University at Harrisburg **The Capital College**

Harrisburg School District **Foose Early Childhood Center**

CUMBERLAND COUNTY

West Shore School District **New Cumberland Middle School** **Hillside Elementary School** **St. Theresa School**

Reference: 1990 Evacuation Travel Time Estimates for Three Mile Island Nuclear Generating Station Plume Exposure Pathway Emergency Planning Zone.

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2.3 LAND USE

Land within a 10 mile radius of the site is used primarily for farming. Farm produce includes dairy, tobacco, poultry, vegetable, fruit, alfalfa, soybeans, corn, wheat, and other products. Agricultural characteristics for Dauphin, York, and Lancaster Counties are listed in Table 2.3-1.

The nearest dairy farm is located on Gingrich Road. It is approximately one mile from the center of Three Mile Island Nuclear Station and is known as the Alwine Farm.

A summary of land use for Dauphin County, in which the site lies, and for the two other adjacent counties (York and Lancaster) is as follows:

<u>Use</u>	Percent of Land Used		
	<u>Dauphin</u>	<u>York</u>	<u>Lancaster</u>
Forest	48.7	27.8	16.2
Crops	29.6	45.3	61.8
Pasture	3.6	7.6	3.4
Other	18.1	19.3	18.6

The location of these counties is shown on Figure 2.2-1. Typical industries in the site region are listed in Table 2.3-2. The predominant industrial areas are around Harrisburg and Middletown, up river from the site.

Transportation routes in the site vicinity on the east bank of the river include State Highway 441, a two-lane blacktop road passing north and south, and a Norfolk Southern Railroad one-track line adjacent and parallel to this highway; each is outside the exclusion boundary. On the west bank, there is a multitrack Norfolk Southern Railroad line at the river's edge about 1.25 miles west of the site and a blacktop, two-lane road parallel to it. There is no commercial water transportation on the river adjacent to the site. The river is transected south of the site by the York Haven Dam, which does not have locks.

There is no commercial fishing in the Susquehanna River in the vicinity of TMI. Recreational fishermen can expect to catch about 42 species of fish in the Susquehanna River. Species other than those listed in Table 2.3-3 may be taken, but not in significant numbers.

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TABLE 2.3-1
(Sheet 1 of 1)

AGRICULTURAL CHARACTERISTICS (AS OF 1987)*

	<u>Lancaster</u>	<u>Dauphin</u>	<u>York</u>
Number of farms (1987)	5,080	790	2,340
Land in farms (acres)	417,296	112,821	299,879
Percent of total land area	68.5	33.4	51.7
Average size of farm (acres)	84	145	130
Value of Products Sold (\$)	109,295,000	13,086,000	41,942,000
Field crops (\$)	101,635,000	11,761,000	33,161,000
Vegetables (\$)	6,019,000	774,000	4,285,000
Fruits & nuts (\$)	1,641,000	551,000	4,496,000
Forest products & horticultural products (\$)	10,824,000	1,267,000	5,388,000
Poultry & poultry products (\$)			
Chickens	21,488,000	1,190,000	1,434,800
Eggs	85,943,000	5,381,000	7,840,000
Milk/Dairy Products(\$)	203,741,000	14,582,000	30,106,000
Livestock & Livestock Products (\$)	201,551,500	17,272,500	38,692,350
Crops Harvested			
Corn (acres) Grain	105,500	26,000	70,000
Silage	75,100	6,700	16,500
Wheat (acres)	17,500	4,000	23,500
Potatoes (acres)	1780	80	2,040
Tobacco (acres)	10,010	20	110

* Source: Statistical Summary 1987-88 and PA Dept. of Ag. Annual Reports; Natl. Ag. Statistics Service U.S. Dept. of Agriculture/PA Dept. of Agriculture Publ. PASS-99

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TABLE 2.3-2
(Sheet 1 of 3)

TYPICAL MAJOR* INDUSTRIES WITHIN 10 MILES OF THREE MILE ISLAND

<u>COMPANY</u>	<u>LOCATION/DIRECTION</u>
1. Covenco	Lower Swatara 6-7 miles NW
2. Bethlehem Steel	Steeltown 7-8 miles NW
3. SAIC	Londonderry 5-6 miles N
4. AMP Inc.	Lower Swatara 5-6 miles NNW
5. AMP	Swatara 8-9 miles NW
6. Mack Trucks	Lower Swatara 5-6 miles NW
7. Turnpike Ind. Park	Lower Swatara 5-6 miles NW
8. Advanced Conversion	Lower Swatara 5-6 miles NNW
9. Chloe Textiles	Lower Swatara 3-4 miles NNW
10. UPS	Swatara 9-10 miles NW
11. UPS	Lower Paxton 9-10 miles NNW
12. Pa Workers Comp	Harrisburg 8-9 miles NW

* Business with greater than 100 employees

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TABLE 2.3-2
(Sheet 2 of 3)

TYPICAL MAJOR* INDUSTRIES WITHIN 10 MILES OF THREE MILE ISLAND

<u>COMPANY</u>	<u>LOCATION/DIRECTION</u>
13. Jackson Manufacturing	Harrisburg 9-10 miles NW
14. Hershey Foods 1	Derry 9-10 miles NNE
15. Hershey Foods 2	Derry 9-10 miles NNE
16. Hershey Foods 3	Derry 9-10 miles NNE
17. Hershey Foods 4	Derry 9-10 miles NNE
18. Hershey Foods 5	Derry 9-10 miles NNE
19. Hershey Foods 7	Derry 9-10 miles NNE
20. Maple Press	Manchester 7-8 miles S
21. Offsite Advantage	Fairview 6-7 miles WNW
22. Admin Center for Education	Fairview 6-7 miles WNW
23. DuPont Electronics	Manchester 9-10 miles S
24. DuPont Electronics	Fairview 5-6 miles W
25. FES, Inc.	Manchester 9-10 miles S

* Business with greater than 100 employees

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TABLE 2.3-2
(Sheet 3 of 3)

TYPICAL MAJOR* INDUSTRIES WITHIN 10 MILES OF THREE MILE ISLAND

<u>COMPANY</u>	<u>LOCATION/DIRECTION</u>
26. Trenwyth	Manchester 9-10 miles S
27. Preston Trucking	Manchester 9-10 miles S
28. Progressive Typeset	Manchester 9-10 miles S
29. Pa Auto Dealers Assoc.	Conawago 5-6 miles SSW
30. New Cumberland Depot	Fairview 7-8 miles WNW
31. USA Direct	Manchester 7-8 miles S
32. M&M Mars	Elizabethtown 6-7 miles E
33. Armstrong Industries	E. Donegal 9-10 miles SE
34. Wengers Feed Mill	Mount Joy 7-8 miles E
35. Verdelli Farms	Derry 8-9 miles N
36. Turnpike Commission	Lower Swatara 5-6 miles NW

Reference: 1990 Evacuation Travel Time Estimates for Three Mile Island Nuclear Generating Station Plume Exposure Pathway Emergency Planning Zone.

* Business with greater than 100 employees

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TABLE 2.3-3
(Sheet 1 of 2)

FISH SPECIES OF THE SUSQUEHANNA RIVER

<u>Common Name</u>	<u>Scientific Name</u>
Trout	
Brown Trout	<u>Salmo trutta</u>
Pikes	
Northern Pike	<u>Esox lucius</u>
Muskellunge	<u>Esox masquinongy</u>
Minnows	
Goldfish	<u>Carassius auratus</u>
Carp	<u>Cyprinus carpio</u>
Cutlips minnow (rare)	<u>Exoglossum maxillina</u>
Golden shiner	<u>Notemigonus crysoleucas</u>
River chub (rare)	<u>Nocomis micropogon</u>
Common shiner	<u>Notropis cornutus</u>
Spottail shiner	<u>Notropis hudsonius</u>
Swallowtail shiner	<u>Notropis procne</u>
Spotfin shiner	<u>Notropis spilopterus</u>
Bluntnose minnow	<u>Pimephales notatus</u>
Blacknose dace (rare)	<u>Rhinichthys atratulus</u>
Longnose dace	<u>Rhinichthys cataractae</u>
Creek chub	<u>Semotilus atromaculatus</u>
Fallfish (rare)	<u>Semotilus corporalis</u>
Comely shiner	<u>Notropis amoenus</u>
Mimic shiner	<u>Notropis volucellus</u>
Suckers	
Quillback	<u>Carpiodes cyprinus</u>
White sucker	<u>Catostomus commersoni</u>
Northern hog sucker	<u>Hypentelium nigricans</u>
Shorthead redhorse	<u>Moxostoma macrolepidotum</u>
Catfish	
White catfish	<u>Ictalurus catus</u>
Black bullhead	<u>Ictalurus melas</u>
Yellow bullhead	<u>Ictalurus natalis</u>
Brown bullhead	<u>Ictalurus nebulosus</u>
Channel catfish	<u>Ictalurus punctatus</u>
Freshwater eels	
American eel	<u>Anquilla rostrata</u>
Herrings	
Gizzard shad	<u>Dorosoma cepedianum</u>

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TABLE 2.3-3
(Sheet 2 of 2)

FISH SPECIES OF THE SUSQUEHANNA RIVER

<u>Common Name</u>	<u>Scientific Name</u>
Sunfish and Bass	
Rock bass	<u>Ambloplites rupestris</u>
Redbreast sunfish	<u>Lepomis auritus</u>
Green sunfish	<u>Lepomis cyanellus</u>
Pumpkinseed	<u>Lepomis gibbosus</u>
White crappie	<u>Pomoxis annularis</u>
Black crappie	<u>Pomoxis nigromaculatus</u>
Bluegill	<u>Lepomis macrochirus</u>
Smallmouth bass	<u>Micropterus dolomieu</u>
Largemouth bass	<u>Micropterus salmoides</u>
Perches and Darters	
Tessellated darter	<u>Etheostoma olmsted</u>
Yellow perch	<u>Perca flavescens</u>
Walleye	<u>Stizostedion vitreum</u>

Other species may be found in the Susquehanna but not in significant numbers.

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2.4 AIRPORT TO SITE RELATIONSHIP

There are two airports in the site region, Harrisburg International Airport, formerly called Olmsted State Airport, 2.5 miles northwest, and Capital City Airport, formerly called Harrisburg-York Airport, 8 miles west northwest. The former handles primarily commercial and the latter primarily private aircraft. Runway location and use is discussed in Section 2.4.1, and the probability of airplane strikes on the plant is discussed in Section 2.4.2.

2.4.1 RUNWAY LOCATION AND USE

Harrisburg International Airport is located on the east bank of the river and has only one runway (130°/310°). Instrument landing approaches to 310° would align with the runway direction and the aircraft would pass approximately 7500 feet NNE of the site. Aircraft intending to land on 310° could pass near or over the site prior to turning on final approach; however, this would not be a standard VFR approach. The normal takeoff pattern on 130° is away from the site, i.e., the aircraft turns to the left after takeoff. Aircraft takeoff and landing patterns in the other respective direction are out of the site area. The missed approach holding pattern for Harrisburg International Airport is also not in the site area.

The Capital City Airport has two runways (120°/300° and 80°/260°. Instrument landing approaches to 300° would align with the runway direction and the aircraft would pass approximately 1/2 mile to the NNE of the site at an elevation of about 2300 feet. Aircraft on VFR intending to land on 300° could pass near, or over, the site prior to turning on final approach. However, 300° is seldom used due to high terrain considerations and short length (4000 feet). Occasionally, strong crosswind effects on other landing approaches require the use of this runway. Aircraft departing on 120° would normally start a right turn approximately 1 to 3 miles from the end of the runway, depending upon type of aircraft. Aircraft takeoff and landing patterns in the other respective directions are out of the site area. The Capital City Airport has one missed approach holding pattern. It is located such that aircraft would pass near the site at an altitude of roughly 3000 feet. However, aircraft in the holding pattern would comprise considerably less than 1 percent of all aircraft instrument approaches. Most aircraft having missed the landing would immediately be vectored by radar to make another approach.

2.4.2 PROBABILITY OF AIRPLANE STRIKES*

2.4.2.1 General

The Three Mile Island Station is 2.5 miles (straight-line distance) from the eastern end of the single runway of Harrisburg International Airport. The station is about 1.5 miles to the southwest of the extended runway center line. The respective locations of the station and the airport and its runway are shown on Figure 2.4-1 of the FSAR. Air traffic patterns in the site area are based on estimates made in 1991.

Estimates of the probability of various types of airplane crashes into the plant and of related fires are given in Table 2.4-1. The development of each estimate is summarized below.

* Note: The original analyses and licensing bases can be found in the PSAR and Update-1 to the UFSAR and is based on the period of record 1956 to 1965.

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2.4.2.2 Probability of Strike by Large Aircraft

Accident records in the annual statistical summaries of U.S. air carrier accidents (Reference 1) and in individual aircraft accident reports available from the National Transportation Safety Board covering about 118 million aircraft movements (landings plus takeoffs) during the 10 year period 1978 through 1987 were examined and are summarized in Tables 2.4-2, 2.4-3, and 2.4-4. This information along with data concerning the number and types of aircraft movements at Harrisburg International Airport were used to estimate the probability of a hypothetical aircraft incident as shown in Items 1 and 2 of Table 2.4-1.

The types of air carrier aircraft used in the aircraft incident study for Harrisburg International Airport in terms of approximate percent of total air carrier movements are:

DC-9	31%
Boeing 727	26%
Boeing 737	17%
Shorts 360	16%
Other	<u>10%</u>
	100%

In addition to air carrier movements, there are some Air National Guard flights from a unit stationed at Harrisburg International Airport (using C-130 and C-5A type aircraft) and some small percentage of transient military flights including helicopters. Also, United Parcel Service (UPS) began daily flights to Harrisburg International Airport using Boeing 757 in October 1989. Even though the accident records available for this study update do not go beyond 1987, this additional information has been incorporated in Section 2.4.2.3; "Probability of a strike by a very large aircraft".

Of all the aircraft using the airport, at the time of evaluation the C5A was the largest type. Table 2.4-5 gives pertinent characteristics of typical air carrier aircraft. Since most of the aircraft movements at Harrisburg International Airport are of the air carrier type, the use of air carrier accident statistics were considered appropriate.

In estimating the probability of an air carrier aircraft strike, it was assumed that there are 80,000 air carrier movements per year at Harrisburg International Airport, which was about five times the 1989 annual rate. During the 10 year period 1979 to 1988, total air carrier movements in the U.S. increased by a factor of about 37 percent. If the national increase in the next ten years is like that in the past ten and Harrisburg International Airport increases at twice that national rate, it would have about 62,000 movements per year by 1999 but would not reach 177,000 movements during the plant lifetime if movements continued to increase by the same increment each year. Even if a doubling of the movements in five to ten years is assumed and if this very fast increase were achieved and sustained, 177,000 movements per year could be realized by sometime between 1990 and 2000. Since the midpoint of assumed plant life will be about 1994, the assumption of 177,000 air carrier movements a year as a basis for statistical analysis is believed to be reasonable.

Using the data of the National Transportation Safety Board summarized in Table 2.4-2, individual accident briefs were examined to determine the portion of the total fatal accidents which occurred in the proximity of airports (i.e., within a 5 mile radius of the end of the runway being used). The results are summarized in Table 2.4-3. The types of aircraft involved in these accidents are listed in Table 2.4-4.

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The accidents which involved one or more fatalities were chosen as the basis for estimating the probability of the types of crashes which could have a significant effect on the plant because the occurrence of fatalities is usually due to high deceleration rates and/or large fires. Nonfatal accidents were not included because examination of the records indicates that those occurring away from the airport runways usually have some direction and altitude control before impact and are of the type in which there is a good chance large structures could have been avoided.

Fatal landing accidents inside the area ± 0.5 mile from the runway extended centerline were excluded.

Fatal takeoff accidents within a radius of 1 mile were excluded. Fatal accidents outside a 5 mile radius were excluded on the grounds that accidents further out were not representative of the type which would affect the plant due to its proximity to the airport.

Random geographic distribution within a 4 mile radius was assumed for the fatal accidents selected as a data base. Random distribution was assumed because the actual distribution with respect to a runway in use appeared to be random.

The estimates of strike probability are based on statistics for the probability of a fatal accident per landing or takeoff for the 10 year period 1978 through 1987, inclusive. Accident statistics for the future will probably be different. However, fatal accident probability per landing and takeoff is expected to decrease in the future because of expected improvements in aircraft and engine reliability, new aircraft testing, navigation equipment and methods, pilot training, and fire control after impact. Further, Harrisburg International Airport has a long runway (8000 ft) which should contribute to the safety of landing and takeoff operations (it is about 1500 ft longer than the main runway at Washington National Airport and has a 1000 ft overrun on each end).

For the reason discussed above, it is probable that the fatal accident probability chosen for the statistical analysis is reasonable and may be conservative.

During the 10 year period 1978 - 1987, there were approximately 59 million aircraft arrivals and 59 million departures. Therefore, the applicable accident frequency (f) is about $4/(59 \times 10^6)$, or 6.8×10^{-8} per departure and $7/(59 \times 10^6)$, or 1.2×10^{-7} per landing considering the selection criteria aforementioned.

The probability of a crash on the station for any one landing or takeoff was taken to be the applicable accident frequency times the ratio of the "target area" of the plant to the "total area" in which the applicable accidents are assumed to happen with random distribution. These areas were estimated as follows:

- a. The "target area" for arrival (landing) accidents was assumed to be approximately the horizontal area (on the ground) which would be covered by the plant plus the shadow cast by the largest vertical cross section of the plant (excluding cooling towers) assuming light rays emanate from the plane as it approaches the plant along a line inclined 10 degrees above the horizontal. This angle was chosen as being a typical descent line for airplanes crashing on landing. (If the angle were greater, the area would be less and the probability of a strike would be less.) The area of the shadow so obtained was increased by 50 percent to account for airplanes which might crash in front of the plant and slide

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into it. The resulting target area for arrival accidents (here called A_a) is about 0.0225 square miles.

- b. The "target area" for departure (takeoff) accidents was similarly estimated using a 45-degree approach angle believed typical of departure crashes. This area (here called A_d) was estimated to be 0.0066 square miles.
- c. The "total area" for random distribution of departure accidents (here called A_{td}) is $\pi (4)^2 - \pi (1)^2 = 47.1$ square miles. Similarly, the "total area" for arrival accidents (A_{ta}) is approximately $\pi (4)^2 - \pi (0.5)^2 = 49.5$ square miles.

For any one arrival, the probability (P_a) of hitting the plant is:

$$P_a = f_a A_a / A_{ta} = 1.2 \times 10^{-7} \times \frac{0.0225}{49.5} = 5.4 \times 10^{-11}$$

Similarly, for any one departure, the probability of hitting the plant is:

$$P_d = f_d A_d / A_{td} = 6.8 \times 10^{-8} \times 0.0066 / 47.1 = 9.5 \times 10^{-12}$$

and for both departures and arrivals the average probability is:

$$\frac{P_a + P_d}{2} = 3.2 \times 10^{-11}$$

This is equivalent to a recurrence interval of one strike every 3.1×10^{10} years per aircraft movement per year.

If it is assumed that there are about 177,000 aircraft movements a year at Harrisburg International Airport and that half of the takeoffs (44,250) and half the landings (44,250) are from the end of the runway nearest the plant and therefore could affect it, the chance for the plant being hit is:

$$p = 44,250 (P_a + P_d) = 44,250 (5.4 \times 10^{-11} + 9.5 \times 10^{-12}) = 2.8 \times 10^{-6}$$

This is equivalent to a recurrence interval for a crash on the plant of about once in 0.36 million years.

In estimating the effect of impact on the plant, it has been assumed that the impact speed is up to 200 knots. The fact that the speed limit in the geographical area of interest is 180 knots would indicate that the assumption of a 200 knot impact speed is reasonable.

2.4.2.3 Probability of A Strike by a Very Large Aircraft ($\geq 200,000$ lb) At a High Angle ($>60^\circ$) On a Critical Structure

The critical structures referred to in the title of this Subsection are those which are protected for direct strikes of large aircraft as described in Chapter 5.

Very large aircraft are taken as those having gross weights in excess of 200,000 lb during landing or takeoff operations at Harrisburg International Airport. Maximum takeoff and landing weights for typical aircraft involved are shown in Table 2.4-5. It should be noted in Table 2.4-5

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that while the maximum takeoff weights of the B-720 and the B-707-12-B are more than 200,000 lb, the maximum allowable landing weight is less. Also, if aircraft are departing for a nearby destination, takeoff weights may be substantially less than the maximum because a full fuel load would not usually be carried.

It is assumed that very large aircraft comprise 3 percent of the total assumed air carrier movements at Harrisburg International Airport or $0.03 \times 177,000 = 5310$ movements/year.

For purposes of approximation, a strike angle of 60° on critical building surfaces was chosen as a basis for investigation. Strike angles less than this (associated with any given weight, speed, and deceleration pattern at impact) would impose loads less than those derived from assuming a 90° impact such as has been done in checking structures for the effect of large aircraft strikes.

Flight path angle (relative to the horizontal) was assumed to be randomly distributed from 0° to 20° for landing accidents and 0° to 90° for takeoff accidents. The probability of a strike from selected directional quadrants was assumed to be 40 percent from a quadrant from 300° through north to 30° ; 40 percent from a quadrant from 30° to 120° ; 10 percent from 120° to 210° ; and 10 percent from 210° through 300° .

These percentages were selected by considering the plant location with respect to the airport and surrounding terrain. The horizontal angle of approach in any quadrant was assumed to be random.

The strike probability for large aircraft was taken to be $2.8 \times 10^{-6}/\text{yr}$ based on an assumed virtual target area of 630,000 ft² for landing and 185,000 ft² for takeoff accidents. About 64 percent of the strike probability was due to landing and 36 percent due to takeoff accidents.

Given the information and assumptions described above, the probability of an aircraft larger than 200,000 lb striking the plant on a critical structure at an angle of greater than 60° relative to the structure surface can be estimated as follows.

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For takeoff accidents, the probability of a very large airplane strike from a given quadrant is:

$$P_T = 2.8 \times 10^{-6} \times 0.03 \times 0.36 \times Q = 3.0 \times 10^{-8} Q$$

and for landing accidents it is:

$$P_L = 2.8 \times 10^{-6} \times 0.03 \times 0.64 \times Q = 5.4 \times 10^{-8} Q$$

Where Q represents the fraction of total strikes arriving from a given directional quadrant.

For takeoff and landing accidents, the probability of such a strike hitting a critical building at 60° for takeoff accidents is:

$$P_{CT} = \frac{f'A'P_T}{1.85 \times 10^5}$$

where f= the fraction of strikes which would impact at greater than 60° and A is the "virtual target area" of critical surfaces which could be hit at >60°. Similarly, for landing accidents the probability is:

$$P_{CL} = \frac{f'A'P_L}{6.3 \times 10^5}$$

Using this method, values of P_{CT} and P_{CL} were estimated for strikes on critical vertical surfaces from each quadrant and on critical horizontal surfaces from all quadrants.

The result indicates that the sum of probabilities from all quadrants is about $1.7 \times 10^{-8}/\text{yr}$.

2.4.2.4 Probability of Small Aircraft Strike

The amount of general aviation movements in the Harrisburg area has been estimated by reviewing information received directly from airport records. In 1989, there were a total of 124,700 landings plus takeoffs, or 62,350 landings and 62,350 takeoffs at Harrisburg area airports. Of these, about 46 percent occurred at Capital City and 54 percent at Harrisburg International Airport. Typical types of aircraft involved are Beechcraft, Piper, and Cessna. Characteristics of the largest and smallest of each aircraft of these types are given in Table 2.4-6.

Accident data for general aviation operations were obtained from the National Transportation Safety Board Annual Review Reports (Reference 22) and are given in Table 2.4-7. In 1986 and 1987, there were a total of 58,525,000 hours flown under the category of general aviation. In order to relate these data to numbers of landings and takeoffs, it is necessary to make a judgment of the average flight duration. This is assumed to be 1 hour. Thus, the assumed total number of landings plus takeoffs is 11.7×10^7 for the years 1986 and 1987.

Of the general aviation accidents, only the fatal accidents are considered because in nonfatal accidents the pilot is assumed to have enough control to be able to avoid the plant. Some fatal accidents may also be of this type.

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The ranges of interest are 2 to 3 miles for Harrisburg International Airport traffic and 7 to 8 miles for Capital City traffic. From a smooth curve fitted to the data in Table 2.4-7, the respective numbers of accidents are 60 and 15 over the 2 year period for these two ranges.

For Harrisburg International Airport operations, the probability (P_H) of there being a fatal crash within 2 to 3 miles is as follows for any landing or takeoff operation:

$$P_H = \frac{60}{(11.7 \times 10^7)} = 0.51 \times 10^{-6}/\text{operation}$$

The projected number of landing plus takeoff operations at Harrisburg International Airport is 0.54 (124,700) = 67,300 per year. Thus, the probability of there being a fatal airplane crash within the 2 to 3 miles is:

$$0.51 \times 10^{-6} \times 67,300 = 0.34 \times 10^{-1}/\text{year}$$

The average "virtual target" area assumed for the plant for landing and takeoff accidents is approximately 0.015 square miles. This is 0.95×10^{-3} times the area within the 2 and 3 mile circles.

Thus, assuming random geographical distribution of the crashes within the 2 to 3 mile radius, the probability that a fatal crash resulting from Harrisburg International Airport operations would strike the Three Mile Island plant in any one year is:

$$P_H = 0.34 \times 10^{-1} \times 0.95 \times 10^{-3} = 0.32 \times 10^{-4}/\text{year}$$

Similarly, the probability ($P_{C/C}$) of a strike by a fatal crash resulting from Capital City operations would strike the Three Mile Island plant in any one year is:

$$P_{C/C} = \frac{15}{(11.7 \times 10^7)} = 0.13 \times 10^{-6}/\text{operation}$$

The projected number of landings plus takeoffs at Harrisburg/York is 0.46 (124,700) = 57,400 per year. Thus, the probability of there being a fatal crash between 7 and 8 miles of the airport is:

$$0.13 \times 10^{-6} \times 57,400 = 0.75 \times 10^{-2}/\text{year}$$

The area between the 7 and 8 mile circles is 47.1 square miles so that the Three Mile Island plant occupies only 0.32×10^{-3} times this area. Thus, the probability that a fatal crash resulting from Harrisburg/York operations would strike the Three Mile Island plant in any one year is:

$$P_{C/C} = 0.75 \times 10^{-2} \times 0.32 \times 10^{-3} = 0.24 \times 10^{-5}/\text{year}$$

The combined probability (P) of the Three Mile Island plant being hit by a fatal crash in any one year is then:

$$P = P_H + P_{C/C} = 0.32 \times 10^{-4} + 0.24 \times 10^{-5} = 3.4 \times 10^{-5}/\text{year}$$

The probabilities estimated above are based on the approximate number of general aviation operations in 1989. If general aviation operations in the Harrisburg area increase by a factor of

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5 on the average, during life of the plant, and if the accident rates remain the same as assumed, the probability (P'_s) would increase by a similar factor and would be about:

$$P'_s = 2 \times 10^{-4}/\text{year}$$

2.4.2.5 Probability of Fire from an Aircraft Strike

a. Small Fires

As indicated in Table 2.4-7 for general aviation aircraft, about 32 percent of fatal crashes have post-accident fires. If this ratio is assumed valid for crashes on the plant, then the probability (P) of crash fires would be about:

$$P = 2 \times 10^{-4} \times 0.32 = 6.4 \times 10^{-5}/\text{year}$$

This assumes that the general aviation movement rate is five times the 1989 rate. Examination of Table 2.4-6 indicates these crashes will probably involve less than 400 gallons of fuel and average less than 100 gallons.

b. Medium Fires

Medium fires are taken to be those wherein more than 400 but less than 3000 gallons of fuel are involved. At the present time, about 57 percent of movements at Harrisburg International Airport involve airplanes with a maximum fuel capacity of 3000 gallons, or less. If, however, it is assumed that at the time air carrier movements reach 177,000 per year, 50 percent of the airplanes involved carry less than 3000 gallons when landing or taking off, then the probability (P) of a medium fire is:

$$P = 2.8 \times 10^{-6} \times 0.5 = 1.4 \times 10^{-6}/\text{year}$$

assuming all air carrier crashes on the plant result in fires.

c. Large Fires

Similarly, the probability (P) of large fires (where more than 3000 gallons of fuel are involved) can be estimated assuming 50 percent of air carrier operations have more than 3000 gallons aboard when landing or departing. Thus:

$$P = 2.8 \times 10^{-6} \times 0.5 = 1.4 \times 10^{-6}/\text{year}$$

Improvements in aircraft design, fire prevention systems, and fuel technology, especially for large aircraft, are expected to reduce the probability of postcrash fires in the future. By the time air traffic movement rates reach those assumed in making the probability estimates above, significant improvements should be realized. Thus, from this viewpoint, the probability of postcrash fires, especially for large aircraft, should be less than assumed.

d. Fuel or Fires Affecting Critical Ventilation Openings

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The probability of fire or fuel from a small airplane crash affecting the ventilation intake or outlet for the Control Room and other protected areas can be approximated by assuming that the "virtual target" is the area of the opening plus the area around it which could be hit and cause the opening to be subjected to fire or to liquid fuel or vapors at flammable concentrations. The openings are less than 400 ft². For a small plane crash carrying an average of about 100 gallons of fuel, it is assumed the fuel-affected area could be about 10 ft x 50 ft or 500 ft². This is believed to be a larger area than would be affected on the average.

The "virtual target" area assumed in deriving the probability for a small plane crash in Subsection 2.4.2.4 above is 0.015 miles² or about 4×10^5 ft², and the probability of a crash (at five times present traffic density) is 2×10^{-4} /year. The probability (P) of a strike on the ventilation openings can be estimated by multiplying this probability by the ratio of "virtual target" areas or:

$$P = 2 \times 10^{-4} \times \frac{500}{4 \times 10^5 \text{ Ft}^2} = 2.5 \times 10^{-7}/\text{Year}$$

This neglects the effect of protection afforded to the openings by structures which could intercept an approaching aircraft.

Consequently, the probability has been taken as being one half that estimated above or 1.3×10^{-7} /year.

To estimate the probability contribution from large aircraft (air carrier planes), the average amount of fuel carried has been assumed to be 5000 gallons, assuming that only a very few, if any, very large planes (i.e., B747's) will use Harrisburg International Airport.

The area affected by spread of fuel from the crash of an aircraft carrying 5000 gallons is assumed to be about 25 ft x 1000 ft or 25,000 ft².

The probability (P) for arriving and departing accidents is given as 2.8×10^{-6} . If these are multiplied by the ratio of the "virtual target" area estimated above to the average virtual target area assumed in deriving the large plane strike probability, the result is an approximate estimate of the probability of fuel or fire from a large aircraft crash affecting critical ventilation openings. Thus,

$$P = 2.8 \times 10^{-6} \times \frac{2.5 \times 10^4}{4 \times 10^5} = 1.8 \times 10^{-7}/\text{year}$$

This also neglects the effect of protection afforded the ventilation openings by structures which could intercept the approaching aircraft. Consequently, the probability is taken to be about one half that estimated above, or 9×10^{-8} /year.

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The combined probability (P) for large and small aircraft crashes affecting the ventilation opening is:

$$P = 2.2 \times 10^{-7}/\text{year}.$$

2.4.3 AIRCRAFT IMPACT ON SITE

Sections 2.4.1 and 2.4.2 provided an analysis of the airport-to-site relationship as well as a probability of an aircraft striking the station. The following presents a summary description of the capability of vital areas of the station to safely withstand a hypothetical aircraft incident.

Although the probability of an aircraft incident at the Three Mile Nuclear Station is extremely remote, the vital structures of the station were designed to have the capability of withstanding a hypothetical aircraft incident. Post permanent defueling, the principal structure is the Fuel Handling Building. The impact loadings on this structure are assumed to come from: (1) an aircraft of 200,000 lbs. travelling at 200 knots impacting an effective area of 19 feet diameter (based on the load-time description in Appendix 3A, (2) an aircraft of 300,000 lb. traveling at 200 knots impacting an effective area of 16 feet diameter (based on load-time description in Appendix 3A, (3) an object of 6000 lb. traveling at 200 knots impacting an effective area of 5 feet diameter, and (4) an object of 4000 lbs. traveling at 200 knots impacting an effective area of 3 feet diameter.

Items (1) and (2) above were analyzed utilizing a dynamic elastic analysis developed by Franklin Institute Research Laboratory. This analysis is based upon time-dependent forces acting on the structure. These time-dependent forces are described in Appendix 3A. Items (3) and (4) above were analyzed by use of the displacement bound theorem for a rigid-plastic continuum. This analysis is based upon an inelastic collision. These analyses produce conservative results which support the conclusion that the stability of the structures is not jeopardized. The load-time curve for an aircraft strike is derived from the angle of impact, velocity, mass distribution, contact area, and structural characteristics of the aircraft. Therefore, different aircraft will have a different time curve depending upon the above variables.

The structural stability of the Fuel Handling Building and Spent Fuel Pool following the hypothetical aircraft incident ensures public safety. The following design provisions provide additional margin of safety by reducing the probability of damage to systems which protect personnel and plant equipment. These provisions include the following:

- a. An air intake tunnel, which brings air into the Auxiliary, Control, and Fuel Handling Buildings, with a remote intake located 125 feet from the plant.
- b. Exhaust openings from ventilation systems, such as the control tower complex, are provided with protective shields designed to withstand hypothetical aircraft incident.
- c. Piping which passes through the top protected level of the Auxiliary Building will be protected with a loop seal approximately six feet deep to prevent the passage of fuel in the event of a ruptured aircraft fuel line.

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TABLE 2.4-1
(Sheet 1 of 1)
APPROXIMATE PROBABILITIES FOR AIRCRAFT CRASH EFFECTS
ON THE THREE MILE ISLAND PLANT

	<u>Approximate Mean Strike Probability/yr¹</u>	<u>Approximate Recurrence Interval/yr</u>
1. Large aircraft on plant (see Section 2.4.2.2)	2.8×10^{-6}	3.6×10^5
2. Large aircraft (>200,000 lb.) at high angle (>60°) on surface of critical structures (see Section 2.4.2.3)	1.7×10^{-8}	5.9×10^7
3. Small aircraft on plant (see Section 2.4.2.4)	2×10^{-4}	5×10^3
4. Fire from an aircraft strike on the plant (see Section 2.4.2.5)		
Small fires (<400 gal. of fuel)	6.4×10^{-5}	1.6×10^4
Medium fires (400-3000 gal of fuel)	1.4×10^{-6}	7.1×10^5
Large fires (>3000 gal of fuel)	1.4×10^{-6}	7.1×10^5
Fuel or fire affecting critical ventilation openings	2.2×10^{-7}	4.5×10^6

¹ In making these approximations of strike probability, the effect of overflights has been ignored. In a region of medium air traffic overflight density this probability may be in the range of 10^{-7} /yr for light aircraft and 10^{-8} /yr for large aircraft if the same type of assumptions are used as in devising the probabilities in this table.

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TABLE 2.4-2
(Sheet 1 of 1)

SUMMARY OF U.S. AIR CARRIER ACCIDENTS ^(1,2)

<u>YEAR</u>	<u>TOTAL ACCIDENTS</u>	<u>FATAL ACCIDENTS</u> ³
1978	22	5
1979	29	5
1980	19	1
1981	26	4
1982	20	5
1983	24	4
1984	17	1
1985	22	7
1986	24	3
1987	36	5

¹ From "Annual Review Aircraft Accident Data, U.S. Air Carrier Operations, Calendar year 1987", National Transportation Safety Board, PB91-119693, NTSB/ARC-90/01

² All 14CFR121, 125 and 127 operations.

³ Fatal accidents are those in which one or more human fatalities occurred.

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TABLE 2.4-3
(Sheet 1 of 1)
FATAL ACCIDENTS IN THE PROXIMITY OF AIRPORTS^{1,2}
(14CFR121, 125, 127 OPERATIONS)

<u>YEAR</u>	<u>TOTAL</u>	<u>NUMBER ARRIVING</u>	<u>DEPARTURE</u>
1978	2	2	0
1979	3	1	2
1980	0	0	0
1981	0	0	0
1982	3	1	2
1983	1	1	0
1984	0	0	0
1985	4	2	2
1986	0	0	0
1987	2	1	1
	<u>15</u>	<u>8</u>	<u>7</u>

¹ Source: Aircraft Accident Briefs, National Transportation Safety Board, Department of Transportation.

² Within a 5 mile radius of the end of the runway being estimated. National Transportation Safety Board started in 1982 compiling data for airport proximity based on the following three categories: on airport, on airstrip, and off airport/airstrip.

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TABLE 2.4-4
(Sheet 1 of 1)

TYPES OF AIRCRAFT INVOLVED IN THE FATAL
ACCIDENTS LISTED IN TABLE 2.4-3^{1,2}

<u>YEAR</u>	<u>AIRCRAFT</u>
1978	B727, B727
1979	L188C, DHC6, DC10
1980	None
1981	None
1982	B737, DC10, B727
1983	Hawker HS748
1984	None
1985	L188A, L188C, L1011, DC9
1986	None
1987	B707, DC9

¹ Source: Aircraft Accident Briefs, National Transportation Safety Board, Department of Transportation.

² Within a 5 mile radius of the end of the runway being estimated. National Transportation Safety Board started in 1982 compiling data for airport proximity based on the following three categories: on airport, on airstrip, and off airport/airstrip.

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TABLE 2.4-5
(Sheet 1 of 2)

CHARACTERISTICS OF TYPICAL AIR CARRIER AIRCRAFT¹

TYPE	MODEL	WEIGHT ²		DIMENSIONS	
		Maximum Takeoff (lb)	Maximum Landing (lb)	Wing Span	Length (Overall)
C-121		135,400			
Conv.	580	58,140	52,000	105'4"	81'6"
DC-9	30	98,000	93,400	93'5"	119'3-1/2"
F-27	J	42,000	40,000	95'2"	77'2"
B-707	331-C)	331,000	247,000	145'9"	162'11"
	320-C)	331,000			
Shorts	360	27,100	26,500	74'9-1/2"	70'9-5/8"

¹ Others-Not Used at Harrisburg International But in Operation in U.S.

B-737	200	107,000	97,000	93'	100'
B-727	200	169,000	148,000	108'	153'2"
B-720	B	234,000	175,000	130'10"	136'9"
B-707	120B	257,000	190,000	130'10"	145'1"
C-130	E	155,000	130,000	132' 7"	97'9"
Electra (185 Orion)		128,000	91,300	99'8"	116'10"

¹ Other Planned for Operation in U.S.

DC-10		386,500	?	155'4"	179'8"
B-747		680,000	564,000	195'8"	231'4"
C5A (107-C)		764,500	635,850	222'8-1/2"	245'11"

¹ Unless otherwise noted, from Jane's All the World's Aircraft (1967-68 and 1990-91 editions).

² Weight for heaviest model.

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TABLE 2.4-5
(Sheet 2 of 2)
CHARACTERISTICS OF TYPICAL AIR CARRIER AIRCRAFT¹

TYPE	MODEL	ENGINES Number	FUEL Gallons	SPEED		PASSENGERS Max. No.
				Cruise	Stall (Max. No. Flaps Down)	
C-121						
Conv.	580	2	2,800	--		50
DC-9	30	2	2,786	561		90
F-27	J	2	2,786	324		48
B-707	331-C) 320-C)	4	23,855	600	100	202
Shorts	360	2	576	216	--	36

¹ Others-Not Used at Harrisburg International But In Operation in U.S.

B-737	200	2	4,670	600	100	113
B-727	200	3	7,680	592	115	189
B-720	B	4	14,880	557	100	116
B-707	120B	4	17,334	627	110	185
C-130	E	4	9,680	368	--	--
Electra (185 Orion)		4	9,200	476	--	--

¹ Others Planned for Operation in U.S.

DC-10		4		600	100	250
B-747		4	50,320	615	100	490
C5A (107-C)		4	49,000	577	--	--

¹ Unless otherwise noted, from Jane's All the World's Aircraft, (1967-68 and 1990-91 editions).

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TABLE 2.4-6
(Sheet 1 of 1)

TYPICAL GENERAL AVIATION AIRCRAFT CHARACTERISTICS^{1,2}

Type	<u>Piper</u>		<u>Beechcraft</u>		<u>Cessna</u>	
Model	<u>Cherokee</u> <u>(PA 28-180)</u>	<u>Navajo</u>	<u>Bonanza</u> <u>(V-35)</u>	<u>99</u>	<u>150</u>	<u>421</u>
Weight, lb	2400	6200	3400	10,200	1600	6800
Engines- number	1	2	1	2	1	2
Fuel	50	190	80	374	38	202
Gallons (max.)						
Type	gas	jet	gas	jet	gas	jet
Cruise Speed	152	224	210	250	123	238
Flaps down stall speed	57	77	63	-	48	87

¹ From Jane's All the World's Aircraft, (1967-1968 edition).

² Includes Largest and Smallest of each Type.

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TABLE 2.4-7
(Sheet 1 of 1)

GENERAL AVIATION
TOTAL FATAL ACCIDENTS IN CONTINENTAL U.S.
(1956-1966 Inclusive)

<u>Phase of Operation</u> <u>(Estimated Distance from Airport)*</u> <u>(miles)</u>	<u>Fatal Accidents</u>
Standing, Taxi, Takeoff, Approach or Landing (0-2)	311
Climb or Descent (2-5)	102
Maneuvering or Cruise (Beyond 5)	470
Other or not reported	<u>44</u>
Total	927
Fire after impact	296

- * Due to change in how General Aviation Accident Data is compiled, assumptions regarding estimated accident distance from airport were made to keep the methodology for this update consistent with the original FSAR.

2.5 METEOROLOGY

2.5.1 GENERAL

General climatic conditions in the site region are characterized by a continental type climate, modified and protected somewhat from more severe weather by the Appalachian Mountain Ridge to the north. Summers tend to be warm and humid, and winters are cool, with frequent periods of precipitation. Severe weather conditions are discussed in Subsection 2.5.2.

An on site meteorological data collection program has been in operation since May 1967. Wind speed and direction have been continuously recorded 100 ft above grade elevation on Three Mile Island; and, 2.5 miles north, at the 25 ft above grade elevation at Crawford Station for a 2 year period of record ending May 1969. Identical data were analyzed to provide a basis for evaluation of routine gas release limits, as described in Subsection 2.5.3. Accident meteorology is described in Subsection 2.5.4. Tables 2.5-1A through 2.5-1D show histograms of the valid wind speed and direction data (an X represents a good data hour) for the Three Mile Island weather tower from May 1967 through May 1969.

Wind roses based on measured data (1967 through 1969) at the site and at Crawford Station are shown on Figures 2.5-1 and 2.5-2, respectively.

2.5.2 SEVERE WEATHER

In more than 75 years of record at the Harrisburg-York Municipal Airport and 56 years of record at the Harrisburg International Airport, the highest and lowest temperatures recorded were 107 and -22F, respectively. Maximum monthly rainfall was 18.55 inches, maximum 24 hour rainfall 12.55 inches, and maximum 24 hour snowfall 24.0 inches. Maximum annual snow accumulation was 81.3 inches.

During the 92 year period 1871 through 1963, 33 hurricane or tropical storm center paths passed within about 100 miles of the site. Most of these were in dissipation stages. The most severe was "Hazel," the center of which passed just west of Harrisburg on October 15, 1954. A peak gust of 80 mph was recorded at the Harrisburg-York Municipal Airport during the passage of Hazel.

In June 1972, Hurricane Agnes passed through causing severe flooding in Harrisburg, Middletown, and surrounding areas.

During the 52 year period from 1917 through June 1969, 22 tornados were reported within a 25 mile radius of the plant site (Reference 2). They are listed in Table 2.5-1. During this same period, four were reported within 10 miles and one within 5 miles of the site. Based on these observations, the probability of a tornado occurring within a 25 mile radius would be about 0.4 per year. However, during the 10 year period 1958 through 1968, the frequency was about 0.7 per year. The higher frequency in the later period probably reflects an improvement in observations rather than a real increase. A small tornado occurred on October 18, 1967, which caused minor damage in Londonderry Township, approximately 3 miles from the plant.

Based on studies by the Weather Bureau (Reference 3) wind speeds 30 ft above the site grade are expected to exceed 78 mph once in 100 years. As used here, wind speed is taken as the average speed during the passage of one linear mile of air past the wind instrument. Taking account of the relationship between maximum wind speeds and short term gusts (Reference 4) it is not expected that gusts will exceed 103 mph at the site more than once in 100 years. The

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fastest mile wind during 27 years of records at Harrisburg was 68 mph from the west in March 1955.

2.5.3 ANNUAL AVERAGE ATMOSPHERIC DIFFUSION CONDITIONS

Annual average atmospheric diffusion conditions at the site have been developed for use in determining routine gas release limits discussed in Chapter 11. Using 2 years of TMI site data (1967 through 1969), averaged values of X/Q for each direction sector were determined according to the following relationship:

$$X/Q \text{ Sector Average} = 2.03 (1/x) (1/\bar{u}) \frac{1}{\sigma_z\{x+x'\}}$$

This is an integrated form of the Pasquill diffusion relationship (References 5 and 6) and assumes a ground level source at a virtual source distance x' upwind from the actual source to account for dilution vertically from the building wake.

The symbols have the following meaning:

x = Distance from source (m)

x' = Virtual source distance (m)

X = Average concentration at ground level in a given 22-1/2 degree sector (curies/m³)

Q = Average release rate (curies/sec)

$\frac{1}{\bar{u}}$ = Harmonic mean of the reciprocal of wind speed (sec/m)

$\sigma_z(x+x')$ = Vertical diffusion coefficient (m),
evaluated at $(x+x')$

Using the site data to determine wind speed, atmospheric diffusion, and wind direction, values of X/Q were computed for each of 48 combinations of wind speed and diffusion categories. These values were then time weighted and added in each of the 16 direction sectors to determine the average values. The wind speed (mph) categories were: 1 to 3, 4 to 7, 8 to 12, 13 to 18, 19 to 25, 26 to 32, 33 to 40, and 40+. The stability categories were A through F as discussed below. The type of diffusion condition was assumed to be a function of wind direction range using relationships suggested in Reference 7. It was assumed that the standard deviation of horizontal wind direction is related to Pasquill diffusion conditions as in the following table:

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<u>Measured Values of Wind Direction Range Divided by 6</u>	<u>Type of Diffusion Condition</u>
0 to 3.67	F
3.67 to 7.5	E
7.5 to 12.5	D
12.5 to 17.5	C
17.5 to 22.5	B
Greater than 22.5	A

Values for the standard deviation of wind direction were determined by dividing the maximum wind direction range measured over a 1 hr period by a factor of 6, as suggested in Reference 7. The values used for the vertical diffusion coefficient σ_z were taken as a function of distance and diffusion condition as shown in Reference 6. The virtual source distances assumed for diffusion conditions A through F are 100, 100, 100, 300, 400, and 700 meters, respectively.

In evaluating the data, it was found that during periods of low wind speed (usually during nighttime), the wind was unsteady, resulting in large values of wind direction range. These highly variable wind direction measurements probably represent good lateral diffusion conditions (higher values of σ_y); however, the vertical diffusion coefficient may not be conservatively represented. To account for this, any range values exceeding 120° during the daytime with wind speeds less than 3 mph were given a value of 120° corresponding to B type diffusion. For nighttime hours, any range values exceeding 75° with wind speeds less than 3 mph were given a range value of 30° corresponding to E type diffusion. Thus, the valley inversion condition which might be expected to occur under these conditions is conservatively taken into account.

Calm conditions were assumed to have the same diffusion (X/Q) and direction value as the next noncalm hour of data.

Using methods described above, isopleths of average annual (X/Q) were derived. They are shown with respect to the local area around the site on Figure 2.5-3. The nearest uncontrolled land area would have an estimated ground level average annual dispersion factor (X/Q) of about 4.5×10^{-6} .

Figure 2.5-7 shows an isopleth of average annual X/Q based on the methods described above except that ΔT from the north tower and the AEC Safety Guide 23 diffusion categories were used to determine σ_z . The data were from the 1 yr period April 1971 through March 1972. Based on this figure, the nearest uncontrolled land area would have an estimated dispersion factor (X/Q) of about $7.5 \times 10^{-6} \text{ sec/m}^3$.

Summary tables of the reduced data, corrected as discussed above for unsteady conditions, used in this analysis are shown in Tables 2.5-8 through 2.5-13. The measured data are given in Tables 2.5-2 through 2.5-7. The number of calms and time of day during which they occurred are given in Table 2.5-14.

2.5.4 ACCIDENT METEOROLOGY

Accident atmospheric dispersion coefficients (X/Q) have been determined at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and various locations of potential contaminant entrance into the Control Room to provide a basis for assessing radiation exposure during accidents described in Chapter 14. Hourly site meteorological data for the years 1992, 1993, 1995, and 1996 were used to determine the dispersion coefficients. The data included wind speed and wind direction collected at the 100-Ft. instrument level and vertical temperature difference determined from the 150-Ft. and 33-Ft. instrument levels. The combined data recovery of speed, direction and delta temperature is indicated below:

<u>Year</u>	<u>Combined Percent Recovery</u>
1992	99.3
1993	99.3
1995	99.2
1996	98.9

2.5.4.1 EAB and LPZ Dispersion Coefficients

Atmospheric dispersion coefficients for the EAB and LPZ were determined utilizing the methodology described in NRC Regulatory Guide 1.145 Revision 1. The analysis (Ref. 32) was performed using the proprietary WINDOW computer code of PLG Inc. and the meteorological data described above. The analysis assumed the total of all releases as ground level releases from point sources at the containment building surface. The effects of building wake and plume meander were evaluated for these releases per R.G. 1.145. Distances to the EAB were determined in accordance with the methods of R.G. 1.145 for each of the 16 direction sectors. The distance measurements excluded the containment radius in each of the sixteen directions. The LPZ distance consisted of a 2-mile radius circle (3218 meters) centered on the containment building.

Wind speed measurements taken at the 100-Ft. instrument level were corrected to 33-Ft. level (10 meters) for use with the R.G. 1.145 equations. The correction was made using the following relation:

$$U_{33\text{Ft}} = U_{100\text{Ft}} (h_{33\text{Ft}} / h_{100\text{Ft}})^n$$

U - wind speed

h - height

n - exponent as function of stability

n = 0.25 for Pasquill Stability Classes A, B and C

n = 0.33 for Pasquill Stability Class D

n = 0.50 for Pasquill Stability Classes E, F and G

The Pasquill diffusion class was determined using vertical temperature difference and the categories given in NRC Regulatory Guide 1.23. Values of σ_y and σ_z were determined as a function of distance and stability class using the standard Pasquill – Gifford curves contained in Figures 1 and 2 of R.G. 1.145.

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Cumulative probability distributions were determined in each of the 16 directions using the direction dependent method for the EAB and LPZ distances and the 4-year hourly data. An envelope was constructed around the 16 direction dependent curves to determine the sector with the maximum 0.5% probable value (i.e. the value exceeded no more than 0.5% of the time). An additional cumulative probability distribution was determined for both the EAB and LPZ using the direction independent method for all the data. The two methods were compared as required by R.G. 1.145 to determine the higher X/Q value. The direction dependent method yielded higher X/Q values for the EAB and LPZ.

The results of the analysis yield the following X/Q values for the time intervals of interest.

<u>EAB</u>	
<u>Interval (Hours)</u>	<u>X/Q (sec/m³)</u>
0-2	8.0E-04

<u>LPZ</u>	
<u>Interval</u>	<u>X/Q (sec/m³)</u>
0-2 Hours	1.4E-04
2-8 Hours	6.0E-05
8-24 Hours	3.9E-05
1-4 Days	1.6E-05
4-30 Days	4.0E-06

2.5.4.2 Control Room Dispersion Coefficients

The site geometry of structures and pathways into the control room were considered to determine the atmospheric dispersion coefficients. Historically, many different paths have been evaluated. The path from the southwest corner of the Auxiliary Building to the Air Intake Structure is the only path considered in the accident analysis in the DSAR. This is a bounding conservative path for the control room air intake.

This path was originally evaluated in support of MHA dose consequence, and in that context was used to evaluate system leakage in the Auxiliary Building. The point of release at ground level at the southwest corner of the Auxiliary building is conservative with respect to any Auxiliary or Fuel Handling Building release. The dispersion coefficients were determined using the ARCON96 computer code and a point source release approach (Reference 33). The model neglected the enhanced dispersion effects produced by adjacent structures. The resulting X/Q values were determined:

<u>Interval</u>	<u>X/Q (sec/m³)</u>
0-2 Hours	3.02E-03
2-8 Hours	2.08E-03
8-24 Hours	1.02E-03
1-4 Days	6.63E-04
4-30 Days	4.37E-04

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TABLE 2.5-1
(Sheet 1 of 1)

TORNADOS WITHIN 25 MILES OF THREE MILE ISLAND¹

Date	Time	Location (Closest approach to site,miles)	Approx. Path Length	Direction of Movement
11-17-1918	11:30 PM	N of Harrisburg(10+)	?	?
6-15-1926	3:00 PM	Paradise, Lancaster Co. (25)	2.5 Miles	S
4-25-1929	8:00 PM	Center of York Co. (12)	?	?
6-12-1938	2:30 PM	W. Central Dauphin Co. (15)	10 Miles	SE
7-15-1938	3:45 AM	Southern York Co.(15)	12-13 Miles	NNE
8-13-1944	2:15 PM	Central York Co. (14)	8 Miles	NE
5-22-1949	?	N. Harrisburg (9)	8 Miles	ENE
4-5 -1952	1:15 PM	Steelton (14)	3 Miles	NNE
4-5 -1952	1:15 PM	Wrightsville York Co.(10)	4 Miles	NNE
4-5-1952	1:45 PM	C. of Lancaster (19)	3 Miles	NNE
11-23-1953	2:00 AM	Landisville, Lancaster Co. (15)	3 Miles	NE
6-13-1956	7:45 PM	Central York Co. (14)	4 Miles	E
4-28-1957	8:00 PM	Central Lancaster Co. (25)	?	?
8-21-1957	2:30 PM	Harrisburg (1)	?	?
11-19-1957	2:30 PM	West-Central Lebanon Co(15)	2 Miles	NE
5-26-1961	?	Central Lancaster Co. (25)	?	?
7-29-1961	3:00 AM	Central York Co. (16)	19 Miles	ENE
?-1963	?	North-Central York Co.(7)	13 Miles	ENE
?-1964	?	Central Lebanon Co. (21)	?	?
8-26-1965	?	Northern York Co. (12)	?	?
9-21-1967	6:00 PM	Perry County (25)	Short	?
10-18-1967	Morning	Dauphin County (3)	15 Miles	

¹ For Period 1917 - June 1969

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TABLE 2.5-2
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group A
(Measured Data)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	31	4	6	17	25	33	67	65
4 - 7	9	2	7	0	5	10	13	19
8 -12	10	2	0	0	2	3	1	8
13-18	6	4	0	0	0	0	1	2
19-25	1	1	1	0	0	0	0	0
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	1	1
Totals	57	12	14	17	32	46	83	95

TMI-1 DSAR

TABLE 2.5-2
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group A
(Measured Data)

Wind Speed mph)	Wind Direction Sector								
	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTALS
1 - 3	60	69	68	48	71	51	65	37	717
4 - 7	24	17	20	15	34	35	55	40	305
8 -12	7	9	10	4	7	13	13	13	102
13-18	2	1	1	0	4	5	10	2	38
19-25	0	0	1	1	0	1	3	0	8
26-32	0	1	0	0	0	0	2	0	3
33-40	0	0	0	0	0	0	0	0	0
40+	1	0	0	0	0	0	0	0	3
Totals	94	97	100	68	116	105	148	92	1176

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TABLE 2.5-3
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group B
(Measured Data)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	14	6	2	11	23	30	37	41
4 - 7	4	0	0	2	11	20	20	16
8 -12	2	0	0	1	0	3	3	7
13-18	1	0	0	0	0	0	3	1
19-25	0	0	0	0	0	0	0	1
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	1	0	0	0	0	0
Totals	21	6	3	14	34	53	63	66

TMI-1 DSAR

TABLE 2.5-3
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group B
(Measured Data)

Wind Speed mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NE	NNW	
1 - 3	36	25	25	36	29	31	25	25	396
4 - 7	28	23	19	14	35	31	32	10	265
8 -12	9	5	12	18	16	23	15	8	122
13-18	2	5	2	4	4	8	2	2	34
19-25	0	1	0	0	1	2	1	1	7
26-32	0	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0	0
Totals	75	59	58	72	85	95	75	46	825

TMI-1 DSAR

TABLE 2.5-4
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group C
(Measured Data)

Wind Speed mph)	Wind Direction Sector							
	N NNE	NE	ENE E	ESE SE	SSE			
1 – 3	36	8	6	41	63	37	50	44
4 – 7	17	1	2	15	33	47	42	46
8 –12	13	1	0	3	10	19	10	16
13-18	0	0	0	2	2	7	3	4
19-25	0	0	0	0	0	0	1	1
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Totals	66	10	8	61	108	110	106	111

TMI-1 DSAR

TABLE 2.5-4
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group C
(Measured Data)

Wind Speed mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	55	48	64	36	34	34	37	48	641
4 – 7	47	49	56	52	62	71	44	55	639
8 – 12	20	18	32	39	58	78	47	36	400
13-18	9	5	10	20	49	40	8	5	164
19-25	1	0	3	4	9	6	5	2	32
26-32	0	0	0	0	0	1	1	0	2
33-40	0	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	1	0	0	1
Totals	132	120	165	151	212	231	142	146	1879

TMI-1 DSAR

TABLE 2.5-5
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group D
(Measured Data)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 – 3	70	34	32	93	52	44	62	63
4 – 7	76	19	6	65	79	90	70	82
8 –12	55	4	3	16	47	42	51	41
13-18	17	4	0	3	9	13	17	15
19-25	9	1	0	0	5	3	0	2
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Total	227	62	41	177	192	192	200	203

TMI-1 DSAR

TABLE 2.5-5
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group D
(Measured Data)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	63	65	57	46	43	36	34	44	838
4 - 7	78	73	73	52	130	82	78	70	1123
8 - 12	30	38	42	53	115	121	106	86	850
13-18	10	14	10	32	110	126	95	34	509
19-25	0	0	1	7	24	54	35	13	154
26-32	0	0	0	1	2	4	11	0	18
33-40	0	0	0	0	0	1	1	0	2
40+	0	0	0	0	0	0	0	0	0
Totals	181	190	183	191	424	424	360	247	3494

TMI-1 DSAR

TABLE 2.5-6
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group E
(Measured Data)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 – 3	29	82	86	59	31	22	29	32
4 – 7	78	99	53	106	58	39	55	70
8 –12	64	28	14	24	22	35	33	36
13-18	32	7	4	4	8	6	15	18
19-25	19	1	0	0	3	0	0	3
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Totals	222	217	157	193	122	102	132	159

TMI-1 DSAR

TABLE 2.5-6
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group E
(Measured Data)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	73	30	17	16	9	27	25	18	585
4 - 7	57	66	29	28	65	72	72	57	1004
8 - 12	43	35	11	9	77	156	102	59	748
13-18	12	6	5	9	48	175	116	39	504
19-25	0	1	3	2	19	64	54	11	180
26-32	0	0	0	0	0	9	9	0	18
33-40	0	0	0	0	0	0	1	0	1
40+	0	0	0	0	0	0	0	0	0
Totals	185	38	65	64	218	503	379	184	3040

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TABLE 2.5-7
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group F
(Measured Data)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	7	8	39	4	2	4	10	7
4 - 7	22	32	42	11	8	7	19	12
8 -12	10	16	19	2	1	2	3	8
13-18	6	4	4	0	0	5	3	3
19-25	0	1	0	0	0	0	0	1
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Totals	45	61	104	17	11	18	35	31

TMI-1 DSAR

TABLE 2.5-7
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group F
(Measured Data)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	25	3	4	2	1	3	1	4	124
4 - 7	20	5	3	3	11	17	11	12	235
8 - 12	5	4	3	2	14	34	5	7	135
13-18	0	0	0	0	6	27	4	7	69
19-25	0	0	0	0	2	4	3	1	12
26-32	0	1	0	0	1	0	0	0	2
33-40	0	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0	0
Totals	50	13	10	7	35	85	24	31	577

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TABLE 2.5-8
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group A
(Corrected for Unsteady Conditions)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	16	1	1	5	5	7	18	28
4 - 7	9	2	7	0	5	10	13	19
8 -12	10	2	0	0	2	3	1	8
13-18	6	4	0	0	0	0	1	2
19-25	1	0	1	0	0	0	0	0
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	1	1
Totals	42	9	9	5	12	20	34	58

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TABLE 2.5-8
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group A
(Corrected for Unsteady Conditions)

Wind Speed (mph)	Wind Direction Sector								
	S	SSW	SW	WSW	WWNW	NW	NNW	Totals	
1 - 3	21	14	18	12	31	21	31	27	256
4 - 7	24	17	20	15	34	35	55	40	305
8 -12	7	9	10	4	7	13	13	13	102
13-18	2	1	1	0	4	5	10	2	38
19-25	0	0	1	1	0	1	3	0	8
26-32	0	1	0	0	0	0	2	0	3
33-40	0	0	0	0	0	0	0	0	0
40+	1	0	0	0	0	0	0	0	3
Totals	55	42	50	32	76	75	114	82	715

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TABLE 2.5-9
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group B
(Corrected for Unsteady Conditions)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	14	5	4	13	23	35	37	35
4 - 7	4	0	0	2	11	20	20	16
8 -12	2	0	0	1	0	3	3	7
13-18	1	0	0	0	0	0	3	1
19-25	0	0	0	0	0	0	0	1
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	1	0	0	0	0	0
Totals	21	5	5	16	34	58	63	60

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TABLE 2.5-9
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group B
(Corrected for Unsteady Conditions)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	33	37	36	37	39	19	30	18	415
4 - 7	28	23	19	14	35	31	32	10	265
8 -12	9	5	12	18	16	23	15	8	122
13-18	2	5	2	4	4	8	2	2	34
19-25	0	1	0	0	1	2	1	1	7
26-32	0	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0	1
Totals	72	71	69	73	95	83	80	39	844

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TABLE 2.5-10
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group C
(Corrected for Unsteady Conditions)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	23	4	2	29	41	21	31	30
4 - 7	17	1	2	15	33	47	42	46
8 -12	13	1	0	3	10	19	10	16
13-18	0	0	0	2	2	7	3	4
19-25	0	0	0	0	0	0	1	1
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Totals	53	6	4	49	86	94	87	97

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TABLE 2.5-10
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group C
(Corrected for Unsteady Conditions)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	26	32	46	24	21	19	22	28	399
4 - 7	47	49	56	52	62	71	44	55	639
8 -12	20	18	32	39	58	78	47	34	398
13-18	9	5	10	20	49	40	8	5	164
19-25	1	0	3	4	9	6	5	2	32
26-32	0	0	0	0	0	1	1	0	2
33-40	0	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	1	0	0	1
Totals	103	104	147	139	199	216	127	124	1635

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TABLE 2.5-11
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group D
(Corrected for Unsteady Conditions)

Wind Speed mph)	<u>Wind Direction Sector</u>							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	70	34	32	93	52	44	62	63
4 - 7	76	19	6	65	79	90	70	82
8 -12	55	4	3	16	48	43	51	41
13-18	17	4	0	3	9	13	17	15
19-25	9	1	0	0	5	3	0	2
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Totals	227	62	41	177	193	193	200	203

TMI-1 DSAR

TABLE 2.5-11
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group D
(Corrected for Unsteady Conditions)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	63	65	57	46	43	36	34	44	838
4 - 7	78	73	73	52	130	82	78	70	1123
8 -12	30	38	42	53	115	121	106	86	852
13-18	10	14	10	32	110	126	95	34	509
19-25	0	0	1	7	24	54	35	13	154
26-32	0	0	0	1	2	5	11	0	19
33-40	0	0	0	0	0	1	1	0	2
40+	0	0	0	0	0	0	0	0	0
Totals	181	190	183	191	424	425	360	247	3497

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TABLE 2.5-12
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group E
(Corrected for Unsteady Conditions)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 - 3	57	90	93	81	73	59	97	91
4 - 7	78	99	53	106	58	39	54	70
8 -12	64	28	14	24	22	35	33	36
13-18	32	7	4	4	8	6	15	18
19-25	19	1	0	0	3	0	0	3
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Totals	250	225	164	215	164	139	199	218

TMI-1 DSAR

TABLE 2.5-12
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group E
(Corrected for Unsteady Conditions)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	144	89	74	63	52	84	69	55	1271
4 - 7	57	66	29	28	65	72	72	57	1003
8 -12	43	35	11	9	77	156	102	59	748
13-18	12	6	5	9	48	175	116	39	504
19-25	0	1	3	2	19	64	54	11	80
26-32	0	0	0	0	0	9	9	0	18
33-40	0	0	0	0	0	0	1	0	1
40+	0	0	0	0	0	0	0	0	0
Totals	256	197	122	111	261	560	423	221	3725

TMI-1 DSAR

TABLE 2.5-13
(Sheet 1 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group F
(Corrected for Unsteady Conditions)

Wind Speed mph)	Wind Direction Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
1 – 3	7	8	39	4	2	4	10	7
4 – 7	22	32	42	11	8	7	19	12
8 –12	10	16	19	2	1	2	3	8
13-18	6	4	4	0	0	5	3	3
19-25	0	1	0	0	0	0	0	1
26-32	0	0	0	0	0	0	0	0
33-40	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0
Totals	45	61	104	17	11	18	35	31

TMI-1 DSAR

TABLE 2.5-13
(Sheet 2 of 2)

WIND VELOCITY DISTRIBUTION VS PASQUILL DIFFUSION GROUP

Three Mile Island Tower
(5/15/67 - 5/14/69)
Diffusion Group F
(Corrected for Unsteady Conditions)

Wind Speed (mph)	Wind Direction Sector								Totals
	S	SSW	SW	WSW	W	WNW	NW	NNW	
1 - 3	25	3	4	2	1	3	1	4	124
4 - 7	20	5	3	3	11	17	11	12	235
8 -12	5	4	3	2	14	34	5	7	135
13-18	0	0	0	0	6	27	4	7	69
19-25	0	0	0	0	2	4	3	1	12
26-32	0	0	1	0	0	1	0	0	2
33-40	0	0	0	0	0	0	0	0	0
40+	0	0	0	0	0	0	0	0	0
Totals	50	12	11	7	34	86	24	31	577

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TABLE 2.5-14
(Sheet 1 of 1)

NUMBER OF CALMS VS TIME OF DAY¹

<u>Hour of the Day</u>	<u>TOTAL</u>
1 AM	30
2	27
3	28
4	42
5	40
6	42
7	47
8	25
9	12
10	5
11	2
12	0
1 PM	2
2	6
3	2
4	3
5	3
6	6
7	9
8	18
9	25
10	25
11	29
12	<u>21</u>
	449

NOTES:

¹ Three Mile Island Met Tower-Period of Record 5/67-4/69

TMI-1 DSAR

2.6 HYDROLOGY

2.6.1 CHARACTERISTICS OF STREAMS IN VICINITY

The major stream affecting the site is the Susquehanna River, having a drainage area at the site of approximately 25,000 square miles. The Susquehanna has a total drainage area of 27,400 square miles, of which 21,000 lie within the State of Pennsylvania. This constitutes approximately 46 percent of the total area of the state, embracing all of 21 and a portion of 22 other counties. Approximately 6200 square miles of the drainage area is in New York State and 200 square miles in western Maryland.

The drainage basin lies in the three main topographic divisions of Pennsylvania: The northern portion of the Allegheny Plateau, the central portion of the Allegheny Mountains, and the lower portion of the rolling Piedmont Plateau.

The main tributaries in the vicinity of the site are the following:

<u>Stream</u>	<u>Drainage Area</u>	<u>Average Flow</u>
Conodoguinet Creek	506 sq.mi	1.20 cfs
Yellow Breeches Creek	219 sq.mi	1.26 cfs
Swatara Creek	571 sq.mi	1.66 cfs
Conewago Creek (East)	52 sq.mi	--
West Conewago Creek	515 sq.mi	1.10 cfs

The Juniata River enters the Susquehanna River about 25 miles upstream from the site. Its drainage area is about 3404 square miles and its average flow is 1.26 cfs.

The plant site on Three Mile Island is located approximately 11 river miles downstream from Harrisburg gauging station, which has a continuous period of record since 1890. The drainage area of the Susquehanna River at the Harrisburg gauge is 24,100 square miles. The average river flow per square mile at Harrisburg is 1.42 cfs. The data for the Harrisburg gauge are assumed to be applicable to the site. Table 2.6-6 is the stage discharge table for the Susquehanna River at the TMINS intake structures.

The Susquehanna River is rather extreme in its flow characteristics, as is evident in the following summary of recorded data at Harrisburg:

Minimum daily flow	1,700 cfs
Average annual discharge	34,000 cfs
Average runoff per square mile	1.41 cfs
Mean annual flood	300,000 cfs
Maximum flood of record (1972)	1,020,000 cfs

The present uses of the streams in the vicinity are for water supply, both public and industrial, power generation, boating, fishing, and recreation. Sport fishing is done in all streams in the general area of the site. Commercial fishing is not practiced in the area.

Figure 2.6-1 shows the location of lakes and reservoirs within a 50 mile radius of the site. Table 2.6-1 lists all known data for these lakes and reservoirs.

Figure 2.6-2 shows the location of major water supplies within a 20 mile radius of the site which take water from streams in the vicinity or from wells. Table 2.6-2 lists all known data for these water supply systems.

2.6.2 STREAM USERS

2.6.2.1 Upstream Users

No large dams or reservoirs exist immediately upstream from the site. A hydroelectric dam and reservoir on the Susquehanna River in Harrisburg are proposed as discussed in Section 2.6.2.2. The Army Corps of Engineers (ACOE) has constructed the Raystown Lake dam and reservoir project for flood control, low flow augmentation, and recreation, on the Raystown Branch of the Juniata River. Hydraulic studies performed by the Corps indicate that Three Mile Island is too far from the Raystown project for a failure of the dam to pose any threat to the site, even under worst case scenarios. Raystown Dam is an earth and rockfill dam with a maximum height of 230 ft. Raystown Lake has a volume of 762,000 acre-feet at full flood control pool.

In 1986, the ACOE performed hydraulic studies to evaluate the impact of a hypothetical dam failure as part of the emergency plan for Raystown Lake (Reference 26) under several scenarios. Based on the results of the studies, the ACOE concluded that Hazardous Conditions as a result of a Raystown Dam failure would not extend downstream of Harrisburg International Airport on the Susquehanna River, which is 115 miles downstream of Raystown dam and 2 miles upstream of TMINS. Hazardous Conditions are defined as flood depths exceeding 2 ft and velocities exceeding 4 ft per second. It is therefore evident that a failure of Raystown Dam during normal Susquehanna River flow conditions or relatively frequent floods would not top the TMINS dike. If failure of the Raystown Dam (at normal high pool or during the Spillway Design Flood) were to coincide with a major flood in the Susquehanna River river flow at TMINS could be increased by approximately 300,000 cubic feet per second, or less, depending on the magnitude of the flood. For example, during the 200 year flood of 800,000 cubic feet per second, an incremental flow increase from the failure of Raystown dam (300,000 cubic feet per second) would result in the design flood of 1,100,000 cubic feet per second at TMINS. At higher Susquehanna River flows, the incremental flow contribution from failure of the Raystown Dam would be somewhat less, due to increased valley storage. In any case, the site would have 20 to 28 hours warning before the incremental flows resulting from a failure of the Raystown Dam peaked at TMINS.

Tables 2.6-3 and 2.6-4 demonstrate data on water quality and temperature for the Susquehanna River gathered by the Pennsylvania Department of Health for Harrisburg, Marietta, and Columbia from 1962 to 1966.

2.6.2.2 Other Power Projects in Vicinity

Immediately downstream from the site is the York Haven hydroelectric project, consisting of a main dam averaging 10 ft in height, extending about 5000 ft across the main river channel from Three Mile Island; a secondary dam, about 8 ft high, extending 950 ft across the east channel of the river; the water impounded by these dams form a lake (Lake Frederick) which extends approximately 3.5 miles upstream from the dams, containing about 10,000 acre-feet of volume. A head race wall about 20 ft in height extends approximately 3000 ft from the west end of the main dam to the powerhouse, which contains 20 units which can generate a total capacity of about 20,000 kW.

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The lake formed by the York Haven Dam is a principal source of recreational use of the river in the vicinity of the site. Sport fishing is done along the river, but the primary use of the lake is for pleasure boating.

The York Haven Station is operated on a run-of-the-river basis, and its power output is dependent upon river flow. The reservoir is used for limited peaking operation during periods of low river flow. Under the peaking conditions, the drawdown of the lake is 1.1 ft maximum.

Brunner Island Station, a large generating plant owned by the Pennsylvania Power & Light Company, is located on the Susquehanna River approximately 1 mile downstream from York Haven powerhouse. This station uses water from the river on a run-through basis for cooling water. York Haven Station maintains a minimum flow in the river of 1000 cfs for Brunner Island.

Three other hydroelectric generating stations are located downstream from the project, with each project having a dam and reservoir on the Susquehanna River. The three stations are Safe Harbor, Holtwood, and Conowingo hydroelectric projects, located approximately 25, 31, and 47 miles south of Three Mile Island, respectively. The Muddy Run pumped storage project is associated with Conowingo Station. The Peach Bottom nuclear generating station is located along the Susquehanna River, just north of the Maryland-Pennsylvania border.

2.6.2.3 Downstream Users

The following summarizes the available information on water supplies downstream from Three Mile Island Nuclear Station for a distance to 50 miles. The consumers include public water supplies, industries, and utilities. There are no points downstream from Three Mile Island where significant concentration of river flow occurs. Storage capacity for each supply is indicated; however, it was not possible to determine, in most cases, the amount of water in storage reserved for fire protection.

- a. Metropolitan Edison Company owns and operates a hydroelectric generating station at York Haven, total installed capacity 20,000 kW. A dam across the Susquehanna River impounds 10,000 acre-feet of water for electric generation, recreation, and fire protection. Potable water is obtained from two wells.
- b. The Pennsylvania Power & Light Company owns and operates a generating station (Brunner Island) on the west shore of the Susquehanna River 5 miles downstream from the Nuclear Station. The plant has a generating capacity of 1415 MWe and utilizes 1155 cfs of river water for the circulating water system and ash removal, which is returned to the river without reconcentration. River water for cycle makeup demineralizer requirements is treated prior to use.
- c. The Wrightsville Water Supply Company has a public water supply intake on the Susquehanna River 16.25 miles downstream from the Nuclear Station. The intake is utilized as a summer reserve supply with complete treatment facilities provided in the system.
- d. The Borough of Columbia takes an average supply of two million gallons per day from the Susquehanna River from an intake located 16.75 miles downstream from the Nuclear Station. The potable water receives complete treatment.

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- e. The Susquehanna Water Treatment Plant (City of Lancaster) has its intake on the Susquehanna River 17 miles downstream from the TMINs and is a partial water source for the city of Lancaster. The plant processes between 12,000,000 and 16,000,000 gallons per day with occasional summer highs of 20,000,000 gallons per day. The water receives complete treatment with a flow through time of two hours.
- f. The Red Lion Municipal Authority has an intake on the Susquehanna River approximately 25 miles downstream from TMINs and is a backup source of water for the city of Red Lion. Operation is limited to typically once per week. Typically, 900,000 to 1.5 million gallons per day would be withdrawn. Operating permit will allow daily withdraw up to 3 million gallons. Water is treated through a filter treatment plant prior to distribution.
- g. The Safe Harbor Water and Power Corporation owns and operates a hydroelectric generating station in the Susquehanna River about 27.25 miles downstream from the Nuclear Station. The dam across the river impounds 92,000 acre-feet for power generating purposes. Water for a public water supply system for the village of Safe Harbor is withdrawn from the reservoir at a rate of 25,000 gallons per day and receives complete treatment.
- h. A public water supply system serves the village of Holtwood on the east side of the Susquehanna River 34.75 miles downstream from the Nuclear Station, at Holtwood Hydroelectric Station. Approximately 22,000 gallons per day are withdrawn from the 19,300 acre-foot Holtwood Reservoir and treated prior to distribution.
- i. The Muddy Run Pumped Storage generating station is located 38 miles downstream of the Nuclear Station. It is a remotely controlled plant, 800 MW capacity, operating between an upper reservoir and Conowingo Reservoir using reversible pump-turbines. River water is used for fire protection. Potable water is supplied by deep wells.
- j. Peach Bottom Nuclear Generating Station is located along the west bank of the Susquehanna River about 41 miles downstream from Three Mile Island. Two nuclear units, each rated at 1065 MWe, withdraw approximately 3450 cfs from the river for cooling purposes.
- k. The City of Baltimore has an intake on the river which draws water from Conowingo Reservoir. The intake is located about 49 miles downstream from Three Mile Island. Baltimore is permitted to withdraw up to 250 MGD when the river flow exceeds 5000 cfs but is limited to 65 MGD at lower river flow. Two water treatment plants provide complete treatment.
- l. Philadelphia Electric Company owns and operates a hydroelectric generating station at Conowingo Dam which impounds 321,500 acre-feet in Conowingo Reservoir. The total installed capacity of the facility is 513,000 kW, which is located 50 miles downstream from Three Mile Island. Potable water supply is obtained from the reservoir to supply the station and Conowingo

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Village. Complete treatment is provided for an average supply of 12,000 gallons per day.

2.6.3 LOW FLOW STUDIES

Low flow studies of the Susquehanna River were conducted using data from the Harrisburg and Marietta gauging station, as recorded by the U.S. Geological Survey (Reference 12). The Harrisburg gauge has a continuous period of record since 1890, and Marietta since 1931.

Figures 2.6-3 and 2.6-4 show annual low flow frequency and flow duration curves for the Susquehanna River at Harrisburg, based upon daily flow data. The minimum flow of 1600 cfs occurred on November 29, 1930, as a result of a freeze-up of the river. The minimum daily flow since the construction of a dam downstream from the water filtration plant on City Island was 1700 cfs on September 28, 1964. Average yearly flow at Harrisburg is 34,000 cfs. The following tabulation summarizes the percent of time the flow is equal to, or less than, that indicated as follows:

<u>Flow (cfs)</u>	<u>Percent of Time</u>
2,000	less than 1
3,000	2
4,000	4
5,000	6
6,000	8
7,000	11
8,000	13
9,000	18
10,000	22
20,000	50
34,000 average flow	69

Figure 2.6-5 shows a minimum flow summary of mean monthly flows for the Susquehanna River at Harrisburg. The data show mean monthly flows by months for recurrence intervals of 2, 5, 10, 20, and 50 years. The curves on Figure 2.6-5 were developed from mean monthly flow data for the period of record from 1890 to 1960. It will be noted on Figure 2.6-5 that characteristically the low flows occur in the late summer and fall, and that the minimum mean monthly flow of record, in general, follows the 50 year curve. Extrapolation of the individual monthly curves for August through December produces the following mean monthly flows for the recurrence interval of once in 100 years:

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<u>Month</u>	<u>100 Years (cfs)</u>
August	2200
September	2000
October	1800
November	2500
December	3000

A minimum flow duration summary is presented on Figure 2.6-6, which shows not only the minimum daily flows, but also their probable frequency and duration. For example, a flow of 5000 cfs or less lasting for 90 consecutive days may be expected to occur one in 5 years. It will also be noted that a minimum daily flow of 2000 cfs will be expected to occur once in 25 years, and this flow, lasting for 20 consecutive days, will occur once in 50 years.

Hydraulic studies were conducted to determine the availability of emergency and shutdown cooling water from the intake structure during periods of extreme low flow (1700 cfs), assuming a failure of the east channel dam or main dam at York Haven.

Loss of York Haven Reservoir could result from failure of the main dam, failure of the east channel dam, or failure of both dams. Failure of a dam was considered to be complete removal of the structure as a barrier to flow. Loss of the east channel dam alone would cause only a partial loss of reservoir. The reservoir would still have ample capacity to furnish water for a safe shutdown and would also furnish sufficient water for operation.

Loss of the main dam would return the river to normal open channel flow with the depth of the water, at any point, dependent upon the hydraulic continuum of the river channel systems. A hydraulic study was made of the river to determine such conditions, since the loss of the main dam is the critical condition.

A hydrographic survey was conducted over the entire reach of the river, involving horizontal and vertical control by normal methods of surveying procedure, determination of water surface elevations by an integrated gaging system, and the use of a fathometer survey to determine stream bed elevations. These results, shown as Figures 2.6-6A through 2.6-6G, gave the necessary information regarding actual channel geometry in order to enter an analytical process of flow distribution and water surface determinations.

The analytical process was carried out by using a general program of water surface profiles in a multidivided channel, using FORTRAN II language, for an IBM 1620 computer. With the channel characteristics, as determined by the hydrographic survey, inserted into the program, it became a specific program for the reach of the Susquehanna River in question. The program was run with the main dam removed and with a total flow of 1700 cfs in the river, which is the minimum daily flow on record. This represents the most severe restrictive effect possible under the existing conditions.

The results of the study show that, under the most conservative evaluations, approximately 430 cubic feet per second will be available at the service water screen and pump house based upon a minimum river flow of 1700 cfs. The quantity is many times the amount required for a safe and orderly plant shutdown and corresponds to a pool surface elevation of 271.7 ft above mean sea level in the middle channel of the river (Reference 24). The intake structure was constructed at an elevation to take water from the bottom of the river and to maintain minimum submergence on the nuclear service pumps at all times. The low-flow intake canal assures that the intake structure has continuous access to river waters at water surface elevations of 270 feet or above.

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Hydrographic surveys of the pool performed from 1974 to 1993 have shown no substantial changes in the bathymetry of the pool.

In 2005, the TMI area hydrographic survey and analysis were repeated (Reference 37). This work is a continuation of and is consistent with the above previous studies. The 2005 study shows that with river flow of 1700 cfs (minimum daily flow of record at Harrisburg); TMI intake flow of 130 cfs; postulated failure and complete removal of the York Haven Dam; approximately 914 cfs of water will be available at the service water screen and pump house. This corresponds to a river water surface elevation of 271.96' above mean sea level.

The 2005 study also shows that with river flow of 1700 cfs (minimum daily flow of record at Harrisburg); TMI intake flow of 130 cfs; postulated failure and complete removal of both the York Haven Dam and the East Channel Dam; approximately 882 cfs of water will be available at the service water screen and pump house. This corresponds to a river water surface elevation of 271.93' above mean sea level.

Historical evaluations described above demonstrated that the river channel was adequate to deliver much higher flows required if the plant was operating. After permanent defueling, the river water intake during normal conditions will be less than 30 cfs. Loss of cooling water at the river water intake would not threaten public safety. The intake structure and river water systems do not perform a safety function.

2.6.4 FLOOD STUDIES

Since the site is located on an island in the Susquehanna River, the flood conditions of the river are of prime importance in the planning of the project.

The Susquehanna River is the principal source of flooding in the Harrisburg area. The large tributaries such as the Conodoguinet, Paxton, Yellow Breeches, and Swatara overflow their banks at times; however, the major cause of flooding is the Susquehanna River. The generating station and its facilities will not have any effect on upstream conditions during the design flood.

Prior to June 1972, the 1936 flood was the flood of record. The 1936 flood was gauged at Harrisburg at 740,000 cfs and resulted from precipitation and snow melt over the entire basin.

The TMI- design flood was established based on the Probable Maximum Flood (PMF) as defined by the ACOE in 1967 (time of PSAR), which was 1,083,000 cfs at Harrisburg. The design flood established for the site is 1,100,000 cfs. The design of the dike which protects TMINS therefore was based upon a design flood of 1,100,000 cfs. The hydraulic design of plant inundation protection is based upon this design flood, with an ample margin of free board to account for wind driven waves.

In 1969, the ACOE issued a revised PMF which predicts a peak river flow of 1,625,000 cfs at TMI. The revised PMF and computations indicate that flood discharge of an unregulated river will be 1,750,000 cfs. This flood, as modified by existing upstream flood control projects, will produce a flow at Harrisburg of 1,600,000 cfs and of 1,625,000 cfs as the PMF at Three Mile Island. The probable maximum precipitation (PMI) for the Susquehanna watershed was obtained from Hydrometeorological Report No. 40. The precipitation at Lewisburg, Wilkes-Barre, and Harrisburg has been chosen as index figures. These precipitation values are shown

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in histogram form on Figure 2.6-14 and in Table 2.6-5. Figure 2.6-9 shows the resulting hydrographs of the PMF, as furnished by the Corps of Engineers.

In 2011, the validity of the PMF as determined in 1969 was reviewed. The ACOE HEC-HMS model was used to assess the Impact of the most significant changes on the PMF since 1969. The model was prepared based on a HEC-1 model of the Susquehanna watershed completed in 1975 by ACOE to learn from the observations made after Tropical Storm Agnes. The rainfall input (PMP) was derived from the National Oceanic and Atmospheric Administration (NOAA), Hydrometeorological Report 40 (HMR-40). The most significant change since the 1969 PMF was the construction of three (3) dams in the upper portions of the watershed in 1979 (Tioga, Hammond, and Cowanesque Dams). This assessment concluded that the PMF flow determine in 1969 (i.e. 1,625,000 cfs) remains the bounding hazard. (Reference 39)

An analysis of flood discharge and frequency relationship was made using data gathered by U.S. Geological Survey on past floods, dating back to 1786 (References 8, 9, 10, and 11). The flood of record occurred in June 1972. This event, according to the U.S. Geological Survey, is the highest flood to have occurred since 1784 and probably the highest since 1740 or an earlier date. The 1972 flood resulted from the blowout of tropical storm "Agnes" which covered the 25,000 square mile drainage area with an average depth of approximately eight inches of rainfall. This rainfall produced a flood at Harrisburg estimated at 1,020,000 cfs. The June 1972 flood water level remained below the top of the protective dikes and observed water elevations showed that the 1970 stage discharge analysis was a conservative assessment of the design flood. Preceding the flood of June 1972, the flood of 1936 was the flood of record. The 1936 flood was gauged at Harrisburg at 740,000 cfs and resulted from precipitation and snow melt over the entire basin.

The following table summarizes key TMI-1 flood hazard parameters (from Reference 38):

Event Definition	River Flow Rate (cfs) at TMI	Predicted Recurrence Interval (years)	Water Surface Elevation (ft.) at ISPH
	200,000	1.4	284.2
	350,000	4.5	287.4
2011 Tropical Storm Lee	600,000	45	292.1
1936 Flood	740,000	124	294.6
Flood of Record (Agnes 1972)	1,042,000	705	300.4
Design Flood	1,100,000	911	301.6
Probable Maximum Flood (PMF)	1,625,000	8670	313.3

2.6.4.1 Discharge Frequency Analysis

An analysis of flood discharge and frequency relationship was made using data gathered by U.S. Geological Survey on past floods. 119 years of peak annual stream flow records from the USGS Harrisburg Gage were used to complete the stage frequency analysis following the procedure outlined in USGS Bulletin 17B. The Log-Pearson Type III statistical analysis of stream-flow gage data was used to generate a discharge-frequency/probability curve. Figure

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2.6-7 shows the predicted discharge frequency for the Susquehanna River at TMI (from Reference 38).

2.6.4.2 Stage Discharge Analysis

The original analysis of flood discharge and frequency relationship was made using data gathered by U.S. Geological Survey on past floods, dating back to 1786 (References 8, 9, 10, and 11). Considerable study was devoted to the stretch of the Susquehanna River between the Middletown Rapids and Safe Harbor Dam to determine the profile of the design flood and to establish the elevation for flood protection facilities. Hydrographic surveys conducted over the entire reach of river under study established stream bed elevations and channel geometry for backwater computations. A computer model of the river was then established using known gaged floods and highwater marks. The rationale for using such a computerized program was based upon the premise that the profile of a known flood can be matched by a computed profile over a significant stretch of the river, and the parameters thus established can be applied to the computation of profiles for higher floods. The original stage discharge analysis used the 1936 flood as the benchmark event to calibrate the model. This analysis completed in 1970 provided the basis for the original design and licensing of TMI-1.

The original analysis was superseded by Reference 38. The current river stage discharge analysis used NAVD-88 as the reference for elevation. Typical TMI design and operating documents, and all elevation references in UFSAR section 2.6 are referenced to NGVD-29. Figures 2.6-8, 2.6-9b and 2.6-9c use NAVD-88 as the elevation reference. To convert values which are referenced to NAVD-88 to NGVD-29, add +0.8 ft.

The current river stage discharge analysis was completed as follows. A steady-state, one-dimensional hydraulic analysis was completed using the USACE Hydrologic Engineering Center's River Analysis System (HEC-RAS). The hydraulic model extends approximately 13.7 miles downstream of TMI and 5.4 miles upstream of TMI. The downstream study limit was established to minimize sensitivity to error in estimating the downstream boundary condition. The downstream study limit is located at the Marietta USGS Gage Station. Topographic information for the model was obtained from the PA DCNR LiDAR data for Dauphin, York, and Lancaster Counties. Bathymetric data (from measurements obtained and evaluated by Ocean Surveys Inc in April 2005) along the Susquehanna River around TMI upstream of the York Haven Dams was incorporated into the HEC-RAS model. Additional survey data for the top of dams and the river bottoms, representing the East and Main Channels below each dam (the "Conewago Rapids"), were incorporated into the HEC-RAS model from the 1999 York Haven Hydrostation Fish Passage Project. For all other cross sections, bathymetric data was obtained from FEMA's effective (1978) HEC-2 hydraulic model.

Additional parameters (such as Manning's 'n' values, ineffective flow areas, and blocked obstructions) were added based on review of PAMAP orthophotographs for Dauphin, Lancaster, and York Counties as well as local observations. Structures, including the East Channel York-Haven Dam, West Channel York-Haven Dam, Shocks Mill Railway Bridge, upstream TMI access bridge, and downstream TMI access bridge, were included in the HEC-RAS model. A USGS stage-discharge rating curve for the Marietta Gage Station was used to establish the downstream boundary conditions of the HEC-RAS model.

A range of discharge values were used in the hydraulic analysis to develop a stage-discharge rating curve at each location of interest along TMI. Flow along the Susquehanna River at TMI splits into four (4) channels formed by TMI, Hill Island, and Shelly Island. Each channel has

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independent hydraulic characteristics due to the varying geometry, structures, and allocation of flow. As river flow increases, some islands become inundated, which changes flow characteristics into contiguous conveyance systems. This situation results in unique split flow scenarios, depending on the river discharge.

At lower flows, the river flow was split among all four (4) channels. During intermediate flows, when some or all of the islands are inundated, the river flow was split only among the channels separated by islands not inundated. When all of the islands are inundated at high flows, the full river section (bank to bank) was included in the model, treating the entire river as a contiguous system. Because of these varying conditions, separate HEC-RAS models were developed:

- Split Flow Model for smaller flow rates (less than 500,000 cfs) - No islands are inundated (split flow around each island);
- Split Flow Model for intermediate flow rates (between 500,000 cfs and 1,500,000 cfs) - Only Shelly Island is inundated (split flow around TMI only);
- Total Flow Model for higher flow rates (greater than 1,500,000 cfs) - Shelly Island and TMI Inundated (no split flow).

The HEC-RAS model was run at the measured peak flow rate from the 1972 Tropical Storm Agnes event. The results (Figure 2.6-9b and 2.6-9c) were compared to observed flood elevations at corresponding locations. The model was calibrated by adjustment of the Manning 'n' values used. An analysis was conducted to assess the sensitivity of the computed Water Surface Elevations (WSE) to a range of overbank Manning's 'n' values for various land uses. At PMF conditions, use of minimum or maximum 'n' values obtained from Table 5-6 in Chow's Open Channel Hydraulics (1959) for various overland conditions did not change the WSE by more than 1.1 feet.

Flow pattern complexities along the Susquehanna River in the vicinity of TMI, caused by islands, dams, and bridges, raised concerns that the 1D HEC-RAS model would produce unacceptable errors in predicting flood elevations. To address these concerns, a Two-Dimensional (2D) Finite-Element River Dynamics Computer Model was used to simulate the PMF hydraulics (flow depths and velocity vectors). The 2D model reach included Three Mile Island, Shelly Island, Hill Island, the York Haven Dams, and the two (2) access bridges. The 2D model was calibrated to observed peak WSEs from Tropical Storm Agnes.

WSEs from the appropriate models were used to develop stage-discharge rating curves (Figure 2.6-8). Figure 2.6-8 presents the series of stage-discharge curves for the indicated locations between PA Turnpike Bridge and Marietta.

Table 2.6-6 presents the stage discharge relationship at the TMI-1 ISPH.

The precipitation necessary to cause a PMF type flood is given for various locations in Hydrometeorological Report No. 40, by the U.S. Weather Bureau. The precipitation at Lewisburg, Wilkes-Barre, and Harrisburg has been chosen as index figures. These precipitation values are shown in histogram form on Figure 2.6-14 and in Table 2.6-5.

A comparison of top of dike elevations and water surface elevations to define freeboard on the dikes, as well as the mean velocities, which can be expected during the design flood are shown on Figure 2.6-15. The plan of the access bridge is also shown on this figure. Elevations and sections of the access bridge are shown on Figure 2.6-16.

2.6.5 DESIGN OF HYDRAULIC FACILITIES

Designs were prepared for the structures associated with flood protection and cooling water facilities including earth dikes, channels, and pressure conduits. The design criteria are conservative and based on sound civil engineering practices. The strength parameters and other design characteristics of the foundations and of the materials which were used in constructing the facilities were determined from comprehensive field and laboratory testing programs. The facilities were constructed and will be maintained and inspected consistent with their design as integral parts of a nuclear station. The designs of the facilities were reviewed and approved by the appropriate state and federal agencies, including the Federal Power Commission, the Corps of Engineers, and the Water and Power Resources Board of the Commonwealth of Pennsylvania, where applicable.

The dikes are earth embankments, constructed of clay and silt. Approved borrow material has been excavated, hauled, placed, and compacted to produce a stable, impervious facility. The material is compacted in layers to a minimum of 95 percent of Standard Proctor density. The permeability of the dike is of the order of 10^{-6} cm/sec. The dike dimensions are: A top width of 20 ft, side slopes of two horizontal to one vertical, and of height, ranging up to 20 ft. Soil tests were conducted during construction to control the placement of material and ensure adequate and proper compaction. The dikes are protected by a layer of dumped riprap of sufficient size and thickness, and with an adequate zone of sand and gravel embedment material, to withstand wave height of 2.25 ft, and a velocity in excess of 12.0 ft/sec, on a 2-on-1 slope. The riprap was sized according to the standards of the Tennessee Valley Authority for riprap on earth dams and in power house and spillway tailraces. The riprap continues downward into natural ground for a minimum depth of 2 ft to provide a cutoff against undermining. This system of dikes will protect the site against inundation and wave action for the site design flood of 1,100,000 cfs. Dikes are provided for the site to protect the plant from wave action for the design flood, since plant grade at 304 ft is above the design flood water surface elevation. The top elevation of the protective dike at the northern tip of Three Mile Island is 310 ft, providing a freeboard of at least 6 ft above the design flood at this location. The dikes along both sides of the island descend uniformly from elevation 310 ft to elevation 305 ft, extending sufficient distances to protect the entire plant site for the design flood. The freeboard at the intake structure is at least 2 ft. Provision has been made for a cutoff at elevation 304 ft extending across the downstream end of the plant site.

The permanent access bridge (the north bridge) extends from the east shore a distance of 1600 ft to Three Mile Island, crossing the east channel and Sand Beach Island. The bridge is a combination rail- roadway structure with concrete slab and steel girders supported on 16 concrete piers and end abutments. The elevations of the deck slab are 306.3 ft at the east end of the bridge, 313.2 ft near the center, and 310 ft at the west end. Throughout the main spans of the bridge, the bottom of the supporting steel girders is approximately 9 ft below the deck. The bridge has been designed to withstand the forces of the design flood. The deck is high enough to ensure dry passage over the bridge during the design flood. The backwater effect of the bridge during the design flood is estimated to be in the order of 0.4 ft at the bridge.

A comparison of top of dike elevations and water surface elevations to define freeboard on the dikes, as well as the mean velocities, which can be expected during the design flood are shown on Figure 2.6-15. The plan of the access bridge is also shown on this figure. Elevations and

TMI-1 DSAR

sections of the access bridge are shown on Figure 2.6-16. A cross section of the dike showing riprap details is shown on Figure 2.6-17.

A Corps of Engineers' "Standard Project Flood," or roughly a repetition of the 1936 flood, is required before water can reach the elevation of the base of the dikes. A cross section of the dike showing riprap details is shown on Figure 2.6-17.

The intake structure has been based upon competent rock and is a reinforced concrete structure with wing walls for erosion protection and concrete cutoff walls to resist undercutting. The design ensures that the pumps could remain functional if the site is subjected to the maximum flood level. The structure has been designed to be stable under all conditions of loading, including maximum flood and ice jams.

After permanent shutdown, the structural stability of the Fuel Handling Building and the Spent Fuel pool as described in section 3.1.5 is sufficient to prevent a challenge to spent fuel integrity and ensure public safety in the event of a PMF.

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TABLE 2.6-1
(Sheet 1 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Total Storage (Acre-Feet)</u>	<u>Surface Area (Acres)</u>
1	Lake Williams	E. Br. Codorus Creek	York Water Company	3,440	220
2	Lake Redman	E. Br. Codorus Creek	York Water Company	5,530	290
3	Pinchot Lake	Beaver Creek	Pa. Dept. Forest & Waters	2,800	342
4	Lake Pa Ha Ga Co.	Branch of Bunch Creek	P.H. Glatfelter Co.	3,350	137
5	Lake Marburg	W. Br. Codorus Creek	P.H. Glatfelter Co.	48,500	1275
6	Indian Rock Res.	W. Br. Codorus Creek	U.S. Corp. of Engineers	28,000	1430
7	Long Arm Reservoir	Long Arm Creek	Hanover Municipal Water Works	5,220	185
8	Sheppard-Myers	Long Arm Creek	Hanover Municipal Water Works	615	46.8+
9	*	Trout Run	Shippensburg Borough Authority	640	35
10	*	Carbaugh Run	Pa. Dept. Health	*	20
11	Lake Heritage	Plum Creek	Lake Heritage, Inc.	*	146
12	Lake Mead	Mud Run	Lake Mead, Inc.	3,680	291
13	#6 Dam	S. Br. Roaring Creek	Roaring Creek Water Company	4,080	185
14	#4 Dam	Trout Creek	Roaring Creek Water Company	108	12
15	#2 Dam	S. Br. Roaring Creek	Bear Gap Water Co.	592	65
16	Clark Valley	Clark Creek	City of Harrisburg	23,600	650
17	Muddy Run Pumped Storage Pond	Muddy Run	Philadelphia Electric Company	60,500	985
18	Conowingo	Susquehanna River	Philadelphia Electric Company	*	6975
19	Holtwood Dam	Susquehanna River	Pennsylvania Power & Light Co.	*	2048
20	Safe Harbor Dam	Susquehanna River	Safe Harbor Power Co.	92,200	*

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TABLE 2.6-1
(Sheet 2 of 12)

HISTORICAL INFORMATION				
This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.				

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Total Storage (Acre-Feet)</u>	<u>Surface Area (Acres)</u>
21	Dock Street Dam	Susquehanna River	City of Harrisburg	*	*
22	*	Hammer Creek	Penna. Fish Commission	*	106.4
23	*	Conodoguinet Creek	U.S. Corp of Engineers Letterkenny Ordinance Depot	1,012	54
24	No.1	W. Br. Hammer Creek	City of Lebanon	55	*
25	No.2	W. Br. Hammer Creek	City of Lebanon	150	*
26	DAM REMOVED	-	-	-	-
27	Octoraro Lake	Octoraro Lake	Chester Municipal Authority	7,700	669
28	Blue Marsh (1)	Tulpehocken Creek	U.S. Corp of Engineers	49,000	1000(2)
29	Antietam	Stony Creek	City of Reading	310	*
30	Angelica	Angelica Creek	City of Reading	147	22.5
31	Lake Ontelaunee	Maiden Creek	City of Reading	11,900	1082
32	*	Furnace Creek	Womelsdorf- Robeson Auth.	108	*
33	*	Peters Spring Run	Maiden Creek Township	92	35
34	High Bridge Dam	Fishing Creek	City of Lebanon	1,170	55
35	Wolf Creek	Tar Run	Pottsville Water Co.	492	25.6
36	Indian Run	Indian Run	Pottsville Water Co.	52	1,470
37	Upper Dam	Tumbling Run	Borough of Schuylkill Haven	725	30
38	Lower Dam	Tumbling Run	Borough of Schuylkill Haven	530	28
39	Mud Run	Mud Run	Mahanoy Township Authority	620	54
40	No. 2 Dam	Little	Borough of Mahanoy Creek Ashland	330	13

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TABLE 2.6-1
(Sheet 3 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

				Total Storage (Acre-Feet)	Surface Area (Acres)
<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>		
41	Dam No.4	Dyers Run	Minersville Water Company	204	21.7
42	Sweet Arrow Lake	E.Br. Swatara Creek	Pennsylvania Power & Light Co.	1,100	100
43	Kauffman Dam	Kauffman Run	Pottsville Water Co.	320	19
44	Rock Run Dam	Rock Run	City of Coatsville	1,020	61
45	Liberty	N.Br. Patapsco River	City of Baltimore	132,000	3,100
46	Pretty Boy	Gun Powder Falls	City of Baltimore	61,400	1,500
47	Lock Raven	Gun Powder Falls	City of Baltimore	70,600	2,400
48	Long Pine Run Dam	Long Pine Run	Borough of Chambersburg	5,370	150
49	Chambers- burg Reser- voir	Conococheague Creek	Borough of Chambersburg	1,037	49
50	Little Buffalo Dam	Little Buffalo Creek	Pa. Dept. Forest & Waters	880	88

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TABLE 2.6-1
(Sheet 4 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Dead Storage (Acre-Feet)</u>	<u>Surface Area (Acres)</u>
1	Lake Williams	E. Br. Codorus Creek	York Water Company	0	0
2	Lake Redman	E. Br. Codorus Creek	York Water Company	0	0
3	Pinchot Lake	Beaver Creek	Pa. Dept. Forest & Waters	0	0
4	Lake Pa Ha Co.	Branch of Bunch Creek	P.H. Glatfelter Co.	0	0
5	Lake Marburg	W. Br. Codorus Creek	P.H. Glatfelter Co.	0	0
6	Indian Rock	W. Br. Codorus Creek	U.S. Corp. of Engineers	0	0
7	Long Arm Reservoir	Long Arm Creek	Hanover Municipal Water Works	0	0
8	Sheppard-Myers	Long Arm Creek	Hanover Municipal Water Works	0	0
9	*	Trout Run	Shippensburg Borough Authority		
10	*	Carbaugh Run	Pa. Dept. Health	0	0
11	Lake Heritage	Plum Creek	Lake Heritage, Inc.	0	0
12	Lake Mead	Mud Run	Lake Mead, Inc.	0	0
13	#6 Dam	S. Br. Roaring	Roaring Creek Water Company	0	0
14	#4 Dam	Trout Creek	Roaring Creek Water Company	0	0
15	#2 Dam	S. Br. Roaring	Bear Gap Water	0	0
16	Clark Valley	Clark Creek	City of Harrisburg	0	0
17	Muddy Run Pumped Storage Pond	Muddy Run	Philadelphia Electric Company	*	640
18	Conowingo	Susquehanna River	Philadelphia Electric Company	*	*
19	Holtwood Dam	Susquehanna River	Pennsylvania Power & Light Co.	*	*
20	Safe Harbor Dam	Susquehanna River	Safe Harbor Power Co.	*	*

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TABLE 2.6-1
(Sheet 5 of 12)

HISTORICAL INFORMATION
This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Dead Storage (Acre-Feet)</u>	<u>Surface Area (Acres)</u>
21	Dock Street Dam	Susquehanna River	City of Harrisburg	*	*
22	*	Hammer Creek	Penna. Fish Commission	0	0
23	*	Conodoguinet Creek	U.S. Corp of Engineers Letterkenny Ordinance Depot	0	0
24	No.1	W. Br. Hammer Creek	City of Lebanon	0	0
25	No.2	W. Br. Hammer Creek	City of Lebanon	0	0
26	DAM REMOVED	-	-	-	-
27	Octoraro Lake	Octoraro Lake	Chester Municipal Authority	0	0
28	Blue Marsh (1)	Tulpehocken Creek	U.S. Corp of Engineers	1500	*
29	Antietam	Stony Creek	City of Reading	0	0
30	Angelica	Angelica Creek	City of Reading	0	0
31	Lake Ontelaunee	Maiden Creek	City of Reading	0	0
32	*	Furnace Creek	Womelsdorf- Robesonina Auth.	0	0
33	*	Peters Spring Run	Maiden Creek Township	0	0
34	High Bridge Dam	Fishing Creek	City of Lebanon	0	0
35	Wolf Creek	Tar Run	Pottsville Water Co.	0	0
36	Indian Run	Indian Run	Pottsville Water Co.	0	0
37	Upper Dam	Tumbling Run	Borough of Schuylkill Haven	0	0
38	Lower Dam	Tumbling Run	Borough of Schuylkill Haven	0	0
39	Mud Run	Mud Run	Mahanoy Township Authority	0	0
40	No. 2 Dam	Little Mahanoy Creek	Borough of Ashland	0	0

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TABLE 2.6-1
(Sheet 6 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Dead Storage (Acre-Feet)</u>	<u>Surface Area (Acres)</u>
41	Dam No.4	Dyers Run	Minersville Water Company	0	0
42	Sweet Arrow Lake	E.Br. Swatara River	Pennsylvania Power & Light Co.	0	0
43	Kauffman Dam	Kauffman Run	Pottsville Water Co.	0	0
44	Rock Run Dam	Rock Run	City of Coatsville	0	0
45	Liberty	N.Br. Patapsco	City of Baltimore	0	0
46	Pretty Boy	Gun Powder Falls	City of Baltimore	0	0
47	Lock Raven	Gun Powder Falls	City of Baltimore	0	0
48	Long Pine Run Dam	Long Pine Run	Borough of Chambersburg	0	0
49	Chambers- burg Reser- voir	Conococheague Creek	Borough of Chambersburg	0	0
50	Little Buffalo Dam	Little Buffalo Creek	Pa. Dept. Forest & Waters	0	0

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TABLE 2.6-1
(Sheet 7 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Average Flow (CFS)</u>	<u>Distance (Miles)</u>
1	Lake Williams	E. Br. Codorus Creek	York Water Company	*	18
2	Lake Redman	E. Br. Codorus Creek	York Water Company	*	18
3	Pinchot Lake	Beaver Creek	Pa. Dept. Forest & Waters	50	10
4	Lake Pa Ha Co.	Branch of Bunch Creek	P.H. Glatfelter Co.	*	21
5	Lake Marburg	W. Br. Codorus Creek	P.H. Glatfelter Co.	*	25
6	Indian Rock	W. Br. Codorus Creek	U.S. Corp. of Engineers	97	16
7	Long Arm Reservoir	Long Arm Creek	Hanover Municipal Water Works	*	31
8	Sheppard-Myers	Long Arm Creek	Hanover Municipal Water Works	*	32
9	*	Trout Run	Shippensburg Borough Authority		50
10	*	Carbaugh Run	Pa. Dept. Health	*	43
11	Lake Heritage	Plum Creek	Lake Heritage, Inc.	*	34
12	Lake Mead	Mud Run	Lake Mead, Inc.	*	20
13	#6 Dam	S. Br. Roaring	Roaring Creek Water Company	*	49
14	#4 Dam	Trout Creek	Roaring Creek Water Company	*	46
15	#2 Dam	S. Br. Roaring	Bear Gap Water	*	48
16	Clark Valley	Clark Creek	City of Harrisburg	*	21
17	Muddy Run Pumped Storage Pond	Muddy Run	Philadelphia Electric Company	*	32
18	Conowingo	Susquehanna River	Philadelphia Electric Company	5985(3)	44
19	Holtwood Dam	Susquehanna River	Pennsylvania Power & Light Co.	6505(4)	30
20	Safe Harbor Dam	Susquehanna River	Safe Harbor Power Co.	*	24

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TABLE 2.6-1
(Sheet 8 of 12)

HISTORICAL INFORMATION					
This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.					

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Average Flow (CFS)</u>	<u>Distance (Miles)</u>
21	Dock Street Dam	Susquehanna River	City of Harrisburg	*	11
22	*	Hammer Creek	Pennsylvania Fish Commission	*	23
23	*	Conodoguinet Creek	U.S. Corp of Engineers Letterkenny Ordinance Depot	*	14
24	No.1	W. Br. Hammer Creek	City of Lebanon	*	21
25	No.2	W. Br. Hammer Creek	City of Lebanon	*	21
26	DAM REMOVED				
27	Octoraro Lake	Octoraro Lake	Chester Municipal Authority	183	43
28	Blue Marsh (1)	Tulpehocken Creek	U.S. Corp of Engineers	*	40
29	Antietam	Stony Creek	City of Reading	*	47
30	Angelica	Angelica Creek	City of Reading	*	43
31	Lake Ontelaunee	Maiden Creek	City of Reading	*	47
32	*	Furnace Creek	Womelsdorf- Robesonina Auth.	*	33
33	*	Peters Spring Run	Maiden Creek Township	*	48
34	High Bridge Dam	Fishing Creek	City of Lebanon	*	29
35	Wolf Creek	Tar Run	Pottsville Water Co.	*	48
36	Indian Run	Indian Run	Pottsville Water Co.	*	42
37	Upper Dam	Tumbling Run	Borough of Schuylkill Haven	*	47
38	Lower Dam	Tumbling Run	Borough of Schuylkill Haven	*	46.5
39	Mud Run	Mud Run	Mahanoy Township Authority	*	49
40	No. 2 Dam	Little Mahanoy Creek	Borough of Ashland	*	48

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TABLE 2.6-1
(Sheet 9 of 12)

HISTORICAL INFORMATION					
This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.					

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Average Flow (CFS)</u>	<u>Distance (Miles)</u>
41	Dam No.4	Dyers Run	Minersville Water Company	*	45
42	Sweet Arrow Lake	E.Br. Swatara River	Pennsylvania Power & Light Co.	*	35
43	Kauffman Dam	Kauffman Run	Pottsville Water Co.	*	49
44	Rock Run Dam	Rock Run	City of Coatsville	*	49
45	Liberty	N.Br. Patapsco	City of Baltimore	*	49
46	Pretty Boy	Gun Powder Falls	City of Baltimore	*	36
47	Lock Raven	Gun Powder Falls	City of Baltimore	*	48
48	Long Pine Run Dam	Long Pine Run	Borough of Chambersburg	*	43
49	Chambers-burg Reser-voir	Conococheague Creek	Borough of Chambersburg	*	42
50	Little Buffalo Dam	Little Buffalo Creek	Pa. Dept. Forest & Waters	*	30

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TABLE 2.6-1
(Sheet 10 of 12)

HISTORICAL INFORMATION
This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Direction</u>
1	Lake Williams	E. Br. Codorus Creek	York Water Company	S
2	Lake Redman	E. Br. Codorus Creek	York Water Company	S
3	Pinchot Lake	Beaver Creek	Pa. Dept. Forest & Waters	SW
4	Lake Pa Ha Co.	Branch of Bunch Creek	P.H. Glatfelter Co.	SSE
5	Lake Marburg	W. Br. Codorus Creek	P.H. Glatfelter Co.	SSE
6	Indian Rock	W. Br. Codorus Creek	U.S. Corp. of Engineers	S
7	Long Arm Reservoir	Long Arm Creek	Hanover Municipal Water Works	SW
8	Sheppard-Myers	Long Arm Creek	Hanover Municipal Water Works	SW
9	*	Trout Run	Shippensburg Borough Authority	W
10	*	Carbaugh Run	Pa. Dept. Health	WSW
11	Lake Heritage	Plum Creek	Lake Heritage, Inc.	SW
12	Lake Mead	Mud Run	Lake Mead, Inc.	WSW
13	#6 Dam	S. Br. Roaring	Roaring Creek Water Company	NNE
14	#4 Dam	Trout Creek	Roaring Creek Water Company	NNE
15	#2 Dam	S. Br. Roaring	Bear Gap Water	NNE
16	Clark Valley	Clark Creek	City of Harrisburg	N
17	Muddy Run Pumped Storage Pond	Muddy Run	Philadelphia Electric Company	SE
18	Conowingo	Susquehanna River	Philadelphia Electric Company	SE
19	Holtwood Dam	Susquehanna River	Pennsylvania Power & Light Co.	SE
20	Safe Harbor Dam	Susquehanna River	Safe Harbor Power Co.	SE
21	Dock Street Dam	Susquehanna River	City of Harrisburg	NW
22	*	Hammer Creek	Pennsylvania Fish Commission	E

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TABLE 2.6-1
(Sheet 11 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Direction</u>
23	*	Conodoguinet Creek	U.S. Corp of Engineers Letterkenny Ordinance Depot	NW
24	No.1	W. Br. Hammer Creek	City of Lebanon	ENE
25	No.2	W. Br. Hammer Creek	City of Lebanon	ENE
26	DAM REMOVED	-	-	-
27	Octoraro Lake	Octoraro Lake	Chester Municipal Authority	SE
28	Blue Marsh (1)	Tulpehocken Creek	U.S. Corp of Engineers	ENE
29	Antietam	Stony Creek	City of Reading	ENE
30	Angelica	Angelica Creek	City of Reading	ENE
31	Lake Ontelaunee	Maiden Creek	City of Reading	NE
32	*	Furnace Creek	Womelsdorf-Robeson Auth.	ENE
33	*	Peters Spring Run	Maiden Creek Township	ENE
34	High Bridge Dam	Fishing Creek	City of Lebanon	NNE
35	Wolf Creek	Tar Run	Pottsville Water Co.	NNE
36	Indian Run	Indian Run	Pottsville Water Co.	NE
37	Upper Dam	Tumbling Run	Borough of Schuylkill Haven	NE
38	Lower Dam	Tumbling Run	Borough of Schuylkill Haven	NE
39	Mud Run	Mud Run	Mahanoy Township Authority	NNE
40	No. 2 Dam	Little Mahanoy Creek	Borough of Ashland	NNE

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TABLE 2.6-1
(Sheet 12 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

Data on Reservoirs and Lakes Within a 50 Mile Radius

<u>No.</u>	<u>Reservoir</u>	<u>Stream</u>	<u>Owner</u>	<u>Distance</u>
41	Dam No.4	Dyers Run	Minersville Water Company	NNE
42	Sweet Arrow Lake	E.Br. Swatara River	Pennsylvania Power & Light Co.	NNE
43	Kauffman Dam	Kauffman Run	Pottsville Water Co.	NNE
44	Rock Run Dam	Rock Run	City of Coastville	ESE
45	Liberty	N.Br. Patapsco	City of Baltimore	S
46	Pretty Boy	Gun Powder Falls	City of Baltimore	S
47	Lock Raven	Gun Powder Falls	City of Baltimore	S
48	Long Pine Run Dam	Long Pine Run	Borough of Chambersburg	WSW
49	Chambersburg Reservoir	Conococheague Creek	Borough of Chambersburg	WSW
50	Little Buffalo Dam	Little Buffalo Creek	Pa. Dept. Forest & Waters	NW

* Information not available

(1) Proposed construction to be completed in 1971, begin about 1968.

Multi-purpose, flood control, water supply, recreation

(2) Area of recreation pool at storage capacity of 14,500 acre-feet

(3) 90% of time as recorded for 33 year period

(4) 90% of time as recorded for 31 year period

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TABLE 2.6-2
(Sheet 1 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Well Depth (Ft.)</u>	<u>Diam. (In.)</u>
1	Marysville Water Company	140,000	Stony Creek in Dauphin Lambs Gap Run (100,000 gal. pond) Trout Run (50,000 gal. pond) Sitterly Spring Run	-	-
2.	Summerdale Water Company	*	# 1 Well	*	*
3.	William Grove Park Company	9,000	# 2 Well Spring in center of lake (Covered and protected from lake water)	200	6
4.	Pennsylvania Industrial School	50,000	Riverton Consolidated Water Company Well	66	8
5.	Mechanicsburg Water Company	*	Trindle Spring		
6.	Gulf Oil Company (Turnpike Service Area)	*	# 1 Well	115	8
7.	Mechanicsburg Water Company	1,240,000	# 1 Well	348	6
8.	Riverton Consolidated Water Company	8,100,000	# 2 Well	150	6
9.	Center Square Water Company	11,400	Yellow Breeches Creek	-	-
10.	Grantham Water Company	35,000	Yellow Breeches Creek	-	-
11.	Dillsburg Borough Council	85,000	# 1 Well	*	*
			# 1 Spring	-	-
			# 2 Spring Well	500	6
			12 Springs		
			# 1 Well	179	6
			# 3 Well	300	6
			# 4 Well	130	8

TMI-1 DSAR

TABLE 2.6-2
(Sheet 2 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Well Depth (Ft.)</u>	<u>Diam. (In.)</u>
12.	Gifford Pinchot State Park	100,000	Pinchot Lake	-	-
13.	Dover Borough	*	# 1 Well	310	8
			# 2 Well	408	8
			# 3 Well	311	8
			# 4 Well	800	8
			# 5 Well	311	8
14.	Gleneagles Water Company	50,000	Well	258	8
15.	Mt. View Water Company	26,386	Well	210	6
16.	Dover Township Authority	143,000	# 1 Well	300	8
			# 2 Well	300	8
			# 3 Well	383	8
			# 4 Well	230	8
17.	West Manchester Township Authority	275,000	4 Wells	*	*
18.	Ivan R. Lehr Water Company	*	Well	312	8
19.	Penvale Water Company	15,000	Well	785	12
20.	Hallam Borough Authority	60,000	8 Spring		
			1 Well	177	8
21.	Jackson Township Water District #1	*	2 Springs		
22.	Pine Springs Water Company	25 houses	1 Well	830	8
23.	New Salem Borough Authority		# 1 Well	140	6
			# 2 Dug Well	7	*
		100,000	# 1 Spring	-	-

TMI-1 DSAR

TABLE 2.6-2
(Sheet 3 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Consumption Water Authority</u>	<u>Gallons per Day</u>	<u>Source</u>	<u>Well Depth (Ft.)</u>	<u>Diam. (In.)</u>
24.	Sylvan Hills Development Water Works	50 Homes	Well	200	8
25.	Windsor Water Authority	65,000	8 Springs 2 Wells	- *	- *
26.	Red Lion Municipal Authority	*	Cabin Creek Reservoir (50 MG)	-	-
27.	Red Lion Municipal Authority	*	Beaver Creek Reservoir (5.6 MG)	-	-
28.	Wrightsville Water Supply Company	144,000	6 Springs # 1 Well Abandoned quarry (5.5 acres, 140 MG) filled by Susquehanna River intake	- 425	- 6
29.	Dauphin Consolidated Water Company	3,500,000	Stony Creek (2 acre lake)	-	-
30.	Linglestown Water Company	25 homes	Well	150	6
31.	Sky Line View Extension	170 homes	Well	400	6
32.	Annville Water Company	450,000	Springs	-	-
33.	Cleona Water Company	25,000	2 Springs from City of Lebanon	-	-

TMI-1 DSAR

TABLE 2.6-2
(Sheet 4 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Consumption Water Authority</u>	<u>Gallons per Day</u>	<u>Source</u>	<u>Well Depth (Ft.)</u>	<u>Diam. (In.)</u>
34.	Dauphin Consolidated Water Company	1,200,000	Open Quarry (1 acre, 10 MG)	-	-
35.	Gulf Oil Company (Turnpike Service Area)	10,000	# 1 Well # 2 Well # 3 Well	242 239 214	* * *
36.	Steeltown Water Company	2,000,000	Susquehanna River	-	-
37.	Dauphin Consolidated Water Company	1,700,000	Beaver Creek (1.2 acre lake) Well	-	-
38.	Dauphin Consolidated Water Company	2,000,000	Swatara Creek (5 acre lake)	178	13
39.	Cambelltown Water Company	700 people	9 Springs 1 Well	- 350	- 8
40.	Mt. Gretna Heights Water Company	12,000	Well	165	*
41.	Mt. Gretna Camp Meeting Association	25,000	Well	200	8
42.	Quentin Water Company	10,000	#1 Well #2 Well	223 314	6 *
43.	Cornwall Borough Municipal Authority	270,000	#1 Well #4 Well White Quarry Tract Spring Freeman's Springs Saddler Run	* 360 - - - -	* 6 - - - -

TMI-1 DSAR

TABLE 2.6-2
(Sheet 5 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Consumption Water Authority</u>	<u>Gallons per Day</u>	<u>Source</u>	<u>Well Depth (Ft.)</u>	<u>Diam. (In.)</u>
44.	Highspire Water Company	314,000	# 1 Well # 2 Well # 3 Well 2 Springs	255 806 878	* * 10
45.	Carroll Acres Development	*	# 1 Well Quarry (4.5 acres, 68 MG)	253 -	8 -
46.	Olmstead Air Force Base	1,000,000	Wells	*	*
47.	-	-	-	-	-
48.	Middletown Water Company	880,000	# 1 Well # 2 Well	400 800	8 10
49.	Gulf Oil Company (Turnpike Service Area)	15,000	# 1 Well # 2 Well # 3 Well # 4 Well	475 806 350 250	* * * *
50.	Elizabethtown Water Company	500,000	7 Springs # 1 Well # 2 Well # 3 Well # 4 Well # 5 Well W. Br. Conoy Creek (1500 gal. pond) Conewago Creek (emergency)	- * 219 476 700 500 - -	- * 18 10 * 10 - -
51.	Masonic Homes	250,000	7 Springs	-	-
		#3 Well (emergency)	*	*	
		# 4 Well			
		# 5 Well		200	8
		# 6 Well		500	*

TMI-1 DSAR

TABLE 2.6-2
(Sheet 6 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Well Depth (Ft.)</u>	<u>Diam. (In.)</u>
52.	Rheems Water Company	40,000	Well	380	6
53.	Mt. Joy Water Authority	*	Spring	-	-
54.	Mt. Joy Water Authority	700,000	Little Chickies Creek	-	-
55.	Manheim Water Company	460,000	Reiffs Run Quarry (0.23 acres)	-	-
56.	Bainbridge	12,000	# 1 well	182	6
			# 2 Well	242	6
57.	Shocks Mill	7,000	1 Well	101	4
58.	East Donegal Municipal Authority	100,000	Glatfelter Springs	-	-
59.	Marietta Water Company	380,000	Dugans Run (0.2 acre, 660,000 gal. lake)	-	-
			Wildcat Reservoir (0.36 acres, 1,000,000 gal. lake)	-	-
60.	Marietta Water Company	*	# 1 Well	116	8
			# 2 Well	133	8
			Spring		
61.	Columbia	2,000,000	Susquehanna River	230	12
62.	Lancaster, City of	8,000,000	Susquehanna River	230	12
63.	Safe Harbor	25,000	Susquehanna River	-	-
64.	Holtwood	290,000	Susquehanna River	-	-
65.	Riverton Consolidated Water Company	2,000,000	Conodoguinet Creek	-	-
66.	Pa. Dept. Agriculture at Summerdale	*	# 1 Well	-	-
			# 2 Well	-	-

TMI-1 DSAR

TABLE 2.6-2
(Sheet 7 of 12)

HISTORICAL INFORMATION					
This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.					

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Yield (Ft.)</u>	<u>Pump (In.)</u>
1	Marysville Water Company	140,000	Stony Creek in Dauphin Lambs Gap Run (100,000 gal. pond) Trout Run (50,000 gal. pond) Sitterly Spring Run	*	*
2.	Summerdale Water Company	*	# 1 Well	*	40
3.	William Grove Park Company	9,000	# 2 Well Spring in center of lake (Covered and protected from lake water)	60	*
4.	Pennsylvania Industrial School	50,000	Riverton Consolidated Water Company Well	200	275
5.	Mechanicsburg Water Company	*	Trindle Spring	500	*
6.	Gulf Oil Company (Turnpike Service Area)	*	# 1 Well	500	500
7.	Mechanicsburg Water Company	1,240,000	# 1 Well	*	100
8.	Riverton Consolidated Water Company	8,100,000	# 2 Well	*	100
9.	Center Square Water Company	11,400	Yellow Breeches Creek	*	*
10.	Grantham Water Company	35,000	Yellow Breeches Creek	*	*
11.	Dillsburg Borough Council	85,000	# 1 Well	120	120
			# 2 Spring	*	*
			Well	*	30
			12 Springs	*	30
			# 1 Well	*	*
			# 3 Well	*	50
			# 4 Well	*	40
				200	200

TMI-1 DSAR

TABLE 2.6-2
(Sheet 8 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Yield (Ft.)</u>	<u>Pump (In.)</u>
12.	Gifford Pinchot State Park	100,000	Pinchot Lake	*	*
13.	Dover Borough	*	# 1 Well	*	30
			# 2 Well	*	15
			# 3 Well	*	30
			# 4 Well	*	35
			# 5 Well	*	21
14.	Gleneagles Water Company	50,000	Well	57	55
15.	Mt. View Water Company	26,386	Well	23	23
16.	Dover Township Authority	143,000	# 1 Well	112	100
			# 2 Well	201	100
			# 3 Well	70	70
			# 4 Well	50	50
17.	West Manchester Township Authority	275,000	4 Wells	*	*
18.	Ivan R. Lehr Water Company	*	Well	89	45
19.	Penvale Water Company	15,000	Well	75	75
20.	Hallam Borough Authority	60,000	8 Spring	*	*
			1 Well	*	100
21.	Jackson Township Water District #1	*	2 Springs		20
22.	Pine Springs Water Company	25 houses	1 Well	300	125
23.	New Salem Borough Authority		# 1 Well	*	19
			# 2 Dug Well	10	*
		100,000	# 1 Spring	*	*

TMI-1 DSAR

TABLE 2.6-2
(Sheet 9 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Yield (Ft.)</u>	<u>Pump (In.)</u>
24.	Sylvan Hills Development Water Works	50 Homes	Well	22	*
25.	Windsor Water Authority	65,000	8 Springs 2 Wells	* *	* *
26.	Red Lion Municipal Authority	* *	Cabin Creek Reservoir (50 MG)	* *	* *
27.	Red Lion Municipal Authority		Beaver Creek Reservoir (5.6 MG)	* *	* *
28.	Wrightsville Water Supply Company	144,000	6 Springs # 1 Well Abandoned quarry (5.5 acres, 140 MG) filled by Susquehanna River intake	* * *	* * 25 150 200
29.	Dauphin Consolidated Water Company	3,500,000	Stony Creek (2 acre lake)	*	*
30.	Linglestown Water Company	25 homes	Well	60	50
31.	Sky Line View Extension	170 homes	Well	*	75
32.	Annville Water Company	450,000	Springs	*	*
33.	Cleona Water Company	25,000	2 Springs from City of Lebanon	*	*

TMI-1 DSAR

TABLE 2.6-2
(Sheet 10 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Yield (Ft.)</u>	<u>Pump (In.)</u>
34.	Dauphin Consolidated Water Company	1,200,000	Open Quarry (1 acre, 10 MG)	*	*
35.	Gulf Oil Company (Turnpike Service Area)	10,000	# 1 Well # 2 Well # 3 Well	* * *	* * *
36.	Steeltown Water Company	2,000,000	Susquehanna River	*	*
37.	Dauphin Consolidated Water Company	1,700,000	Beaver Creek (1.2 acre lake) Well	* 753	* 700
38.	Dauphin Consolidated Water Company	2,000,000	Swatara Creek (5 acre lake)	*	*
39.	Cambelltown Water Company	700 people	9 Springs 1 Well	* 38	* 25
40.	Mt. Gretna Heights Water Company	12,000	Well	*	*
41.	Mt. Gretna Camp Meeting Association	25,000	Well	*	65
42.	Quentin Water Company	10,000	#1 Well #2 Well	* *	60 80
43.	Cornwall Borough Municipal Authority	270,000	#1 Well #4 Well White Quarry Tract Spring Freeman's Spring Saddler Run	* 150 * * * *	* 150 * * * *

TMI-1 DSAR

TABLE 2.6-2
(Sheet 11 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Yield (Ft.)</u>	<u>Pump (In.)</u>
44.	Highspire Water Company	314,000	# 1 Well # 2 Well # 3 Well 2 Springs	* * * *	100 150 250 *
45.	Carroll Acres Development	*	# 1 Well Quarry (4.5 acres, 68 MG)	* *	20 *
46.	Olmstead Air Force Base	1,000,000	Wells	*	*
47.	-	-	-	-	-
48.	Middletown Water Company	880,000	# 1 Well # 2 Well	* 575	600 435
49.	Gulf Oil Company (Turnpike Service Area)	15,000	# 1 Well # 2 Well # 3 Well # 4 Well	* * * *	* * * *
50.	Elizabethtown Water Company	500,000	7 Springs # 1 Well # 2 Well # 3 Well # 4 Well # 5 Well W. Br. Conoy Creek (1500 gal. pond) Conewago Creek (emergency)	* * * 120 * 440 * * *	* * 50 150 400 300 * * *
51.	Masonic Homes	250,000	7 Springs #3 Well (emergency) # 4 Well # 5 Well # 6 Well	* * * * 405 *	* * * * 250 400

TMI-1 DSAR

TABLE 2.6-2
(Sheet 12 of 12)

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed but is not intended or expected to be updated for the life of the plant.

DATA ON OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

<u>No.</u>	<u>Water Authority</u>	<u>Consumption Gallons per Day</u>	<u>Source</u>	<u>Yield (Ft.)</u>	<u>Pump (In.)</u>
52.	Rheems Water Company	40,000	Well	50	60
53.	Mt. Joy Water Authority	*	Spring	600	*
54.	Mt Joy Water Authority	700,000	Little Chickies Creek	*	*
55.	Manheim Water Company	460,000	Reiffs Run Quarry (0.23 acres)	*	*
56.	Bainbridge	12,000	# 1 Well	*	225
			# 2 Well	*	60
57.	Shocks Mill	7,000	# 1 Well	20	18
58.	East Donegal Municipal Authority	100,000	Glatfelter Springs	*	*
59.	Marietta Water Company	380,000	Dugans Run (0.2 acre, 660,000 gal. lake)	*	*
			Wildcat Reservoir (0.36 acres, 1,000,000 gal. lake)	*	*
60.	Marietta Water Company	*	# 1 Well	550	300
			# 2 Well	700	300
			Spring	*	50
61.	Columbia	2,000,000	Susquehanna River	*	10
62.	Lancaster, City of	8,000,000	Susquehanna River	*	100
63.	Safe Harbor	25,000	Susquehanna River	*	*
64.	Holtwood	290,000	Susquehanna River	*	*
65.	Riverton Consolidated Water Company	2,000,000	Conodoguinet Creek	*	*
66.	Pa. Dept. Agriculture at Summerdale	*	# 1 Well	*	*
			# 2 Well	*	*

*Data not available

TMI-1 DSAR

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

Table 2.6-3

[illegible]

TMI-1 DSAR

HISTORICAL INFORMATION

This information was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

Table 2.6-4

STREAM

Susquehanna River

STATION LOCATION

USGS Gage - Above Chickies Cr.
E. Donegal Twp., Lancaster Co.
(Marietta) W.V.

HSE 6526

(Lt. Bk.)

COMMONWEALTH OF PENNSYLVANIA

DEPARTMENT OF HEALTH

SANITARY ENGINEERING

PENNSYLVANIA WATER QUALITY NETWORK

REGION VI

BASIN CODE 2.02.0

STATION NUMBER 2-1

NOTE: ALL RESULTS ARE IN POUNDS EXCEPT WHERE NOTED OTHERWISE; UNITS ARE THOSE USED IN STANDARD METHODS.

DATE			STREAM FLOW (CFS)	FIELD ANALYSIS				LABORATORY ANALYSIS																								
MONTH	DAY	YEAR		TEMPERATURE (DEGREES F)	CONDUCTIVITY (MICRO-MHO/CM)	APPEARANCE (NO. UNITS)	COLOR (PCU-CO.DI)	ODOR (UNITS)	TURBIDITY (UNITS)	pH (UNITS)	ALKALINITY	ACIDITY (MICRO-MHO/CM)	ACIDITY (MICRO-MHO/CM)	HARDNESS	B.O.D.	CHLORIDE	SULFATE	NITRATE (NITROGEN)	TOTAL SOLIDS	SUSPENDED SOLIDS (TOTAL)	COPPER	IRON (TOTAL)	MANGANESE	PHENOL	ARS	PHOSPHATE (PO4)	GAGE READING	COLUMBIAN OCCASIONS MPN/100 CC	ALPHA P/C/L	BETA P/C/L		
7	12	62	3,350	80	12.9	7.6	Pale Green	35	V. Ft. Earthy	25	8.7	64	0	0	220	6.5	16	145	-	335	10	-	0.1	0	-	0.16	-	-	91	430		
10	17	62	22,300	64	7.1	6.8	Pale Yellow	35	Decided Musty	20	7.2	36	-	0	120	1.2	7	80	-	300	15	-	0.8	0.1	-	0.13	-	-	430,000	930,000		
1	23	63		32	12.3	6.8	Pale Gray	20	0	40	7.0	30	-	0	96	3.6	10	78	-	185	15	-	1.4	.6	-	0	-	-	Frozen in Transit			
4	22	63	28,030	64	7.6	7.2	Pale Yellow	15	Ft. Chem.	25	7.0	35	-	0	130	2.9	2.0	89	-	260	10	-	.4	0.6	-	0.1	-	(1)	36.4	7,300		
7	15	63	8,610	69	12.2	8.5	Yellow	10	Ft. Earthy	30	8.3	52	-	0	180	7.2	13	140	-	370	40	-	.3	.1	-	.75	-	33.0	2,400			
10	22	63	2,104	60	9.4	7.5	Pale Straw	10	0	5	7.7	68	-	0	250	2.6	23	220	-	405	0	-	0	0	-	.17	-	31.67	2,400			
1	20	64		RIVER FROZEN OVER ----- NO SAMPLES -----																												
4	8	64	65,214	43	5.7	7.0	Brown	15	0	15	8.1	53	-	0	86	2.3	.7	46	-	120	5	-	.4	.25	-	.15	-	OG 17.6	2,400			
7	6	64		79	9.9	8.7	V. Sl. Brown	10	V. Ft. Musty	10	9.2	47.0	-	0	190	3.8	16	150	-	350	10	-	0	0	-	.08	-	OG 17.6	< 2.2			
9	29	64		60	9.0	7.4	V. Sl. Brown	15	Dist. Earthy	10	7.5	61	-	0	315	1.6	33	248	-	460	10	-	0	0	-	.15	-	OG 17.6	430			
12	22	64		34	13.8	7.0	Clear	20	Dist. Musty	15	7.2	31	-	0	164	2.8	19	90	-	284	5	-	0	1.0	-	.15	-	OG 17.6	2,400			
12	15	65		45	11.8	6.7	V. Sl. Brown	10	0	10	7.7	27	-	0	76	3.5	7	50	-	142	5	-	0.8	0.25	0	.05	-	OG 17.6	46,000			
6	29	65		82	9.0	8.0	Sl. Yellow	15	0	15	8.3	58	-	0	144	5.2	15	126	-	282	10	-	0.2	0.25	0	0.1	-	OG 17.6	2,400			
9	15	65	3,008	L-68	L-8.0	L-7.6	Sl. Yellow	10	Decided Grassy	20	7.8	57	-	0	185	6.1	21	180	-	375	20	-	0.4	0.1	0	.08	-	OG 17.6	< 1	7+ 3		
12	15	65		160			V. Sl. Yellow	15	Paint Musty	10	7.8	37	-	0	90	3	9	64	-	140	10	-	0.4	0.33	0	0	-	OG 17.6	< 1	3		
12	30	AM	11,435	42	12.2	7.1	Yellow	15	Paint Musty	10	7.8	37	-	0	140	3	9	64	-	140	10	-	0.4	0.33	0	0	-	OG 17.6	< 1	3		
3	21	66		140			Yellow	15	Paint Musty	10	7.8	37	-	0	140	3	9	64	-	140	10	-	0.4	0.33	0	0	-	OG 17.6	< 1	3		
2	45	PM	55,115	48	11.2	7.0	Clear	10	Paint Musty	15	7.4	25	-	0	50	1.2	5	36	-	100	15	-	0.6	0.25	0	0	-	OG 17.6	< 1	3		
5	31	66		180			V. Sl. Yellow	5	0	10	7.7	30	-	0	76	4.0	6	50	-	140	10	-	0.6	0.2	L-0 R-0	0	-	OG 17.6	< 1	3		
3	15	66	36,230	68	9.6	7.6	Yellow	5	0	10	7.7	30	-	0	76	4.0	6	50	-	140	10	-	0.6	0.2	L-0 R-0	0	-	OG 17.6	< 1	3		
9	8	66		525			Yellow	5	Decided Musty	10	7.3	45	-	0	224	3.0	22	245	-	426	5	-	0.1	0	LO-02 RO-0	0	-	OG 17.6	< 1	3+ 3		
3	00	PM	1,860	72	9.2	7.6	Clear	10	Dist. Musty	10	7.3	45	-	0	224	3.0	22	245	-	426	5	-	0.1	0	LO-02 RO-0	0	-	OG 17.6	< 1	3+ 3		
12	1	66		200			Clear	10	Dist. Musty	10	7.3	45	-	0	224	3.0	22	245	-	426	5	-	0.1	0	LO-02 RO-0	0	-	OG 17.6	< 1	3+ 3		
6	3	PM	83,465	40	11.6	6.8	Brown	30	Dist. Musty	35	6.9	29	-	0	62	8.0	6	34	-	208	40	-	4.0	0.6	RT-0 LT-0	0	-	OG 17.6	< 1	4+ 3		

REMARKS:

(1) Rating table starts at 35.5', so reading of 3.64 should be 36.4 or stream flow of 28,030 cfs per Rhodes Memo of 6-27-63

(2) See letter of 1-27-65 from Italy in WQW File. There is a 10' correction

* June 12, 1964 2-1 sampling point changed to US30 Bridge @ Columbia

GPU Nuclear

TMI Unit-1

Update -8

7/8/81

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TABLE 2.6-5
(Sheet 1 of 3)

TABULAR PRECIPITATION FOR PMF – HYDROMET NO. 40

TOTAL DURATION (HOURS)			6	12	18	24
PERIODS (HOURS)			6	6	6	6
WILKES-BARRE WARNING: 18 HOURS 8.4" RAIN 24 HOURS 9.6" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR 12 HOUR 24 HOUR 48 HOUR 72 HOUR	4.9	1.9 6.8	1.6	1.2 2.8 9.6
LEWISBURG WARNING: 18 HOURS 10.1" RAIN 24 HOURS 11.5" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR 12 HOUR 24 HOUR 48 HOUR 72 HOUR	6.2	2.3 8.5	1.6	1.4 3.0 11.5
HARRISBURG WARNING: 18 HOURS 6.6" RAIN 24 HOURS 7.8" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR 12 HOUR 24 HOUR 48 HOUR 72 HOUR	3.4	1.8 5.2	1.4	1.2 2.6 7.8

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TABLE 2.6-5
(Sheet 2 of 3)

TABULAR PRECIPITATION FOR PMF – HYDROMET NO. 40

TOTAL DURATION (HOURS)			30	36	42	48
PERIODS (HOURS)			6	6	6	6
WILKES-BARRE WARNING: 18 HOURS 8.4" RAIN 24 HOURS 9.6" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR	1.0	0.8	0.8	0.6
		12 HOUR		1.8		1.4
		24 HOUR				3.2
		48 HOUR				12.8
		72 HOUR				
LEWISBURG WARNING: 18 HOURS 10.1" RAIN 24 HOURS 11.5" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR	1.1	0.9	0.8	0.6
		12 HOUR		2.0		1.4
		24 HOUR				3.4
		48 HOUR				14.9
		72 HOUR				
HARRISBURG WARNING: 18 HOURS 6.6" RAIN 24 HOURS 7.8" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR	1.1	0.9	0.7	0.7
		12 HOUR		2.0		1.4
		24 HOUR				3.4
		48 HOUR				11.2
		72 HOUR				

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TABLE 2.6-5
(Sheet 3 of 3)

TABULAR PRECIPITATION FOR PMF – HYDROMET NO. 40

TOTAL DURATION (HOURS)			54	60	66	72
PERIODS (HOURS)			6	6	6	6
WILKES-BARRE WARNING: 18 HOURS 8.4" RAIN 24 HOURS 9.6" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR	0.5	0.4 0.9	0.3	0.2
		12 HOUR				0.5
		24 HOUR				1.4
		48 HOUR				14.2
		72 HOUR				
LEWISBURG WARNING: 18 HOURS 10.1" RAIN 24 HOURS 11.5" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR	0.6	0.4 1.0	0.3	0.2
		12 HOUR				0.5
		24 HOUR				1.5
		48 HOUR				
		72 HOUR				16.4
HARRISBURG WARNING: 18 HOURS 6.6" RAIN 24 HOURS 7.8" RAIN	CUMULATIVE RAINFALL IN INCHES	6 HOUR	0.6	0.4 1.0	0.3	0.2
		12 HOUR				0.5
		24 HOUR				1.5
		48 HOUR				
		72 HOUR				12.7

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TABLE 2.6-6
(Sheet 1 of 1)

SUSQUEHANNA RIVER LEVEL VERSUS FLOW AT TMI ISPH

<u>Flow (cfs)</u>	<u>Level (feet)</u>
125000	282.4
200000	284.2
250000	285.4
300000	286.4
350000	287.4
400000	288.3
450000	289.2
500000	290.0
550000	291.0
600000	292.1
650000	293.0
700000	293.9
750000	294.8
800000	295.8
850000	296.8
900000	297.8
950000	298.8
1000000	299.8
1050000	300.5
1100000	301.6
1150000	302.6
1200000	303.5
1250000	304.4
1300000	305.3
1350000	306.2
1400000	307.6
1450000	309.3
1500000	311.0
1550000	311.9
1600000	312.8
1625000	313.3
1650000	313.6
1700000	314.2
1750000	314.8

2.7 ENGINEERING GEOLOGY AND FOUNDATION CONSIDERATIONS

2.7.1 GENERAL

Three Mile Island is located approximately 2.5 miles south of Middletown, Pennsylvania, at longitude 76°43'30" west and at latitude 40°8' north. The reactor vessel coordinates are N300.827; E2,286,878, based on the Pennsylvania State coordinate system (UTM coordinates, Zone 18, 4,446,102 meters north, 353,062 meters east). It is one of the largest of a group of several islands in the Susquehanna River and is situated about 900 ft from the east bank. It is elongated parallel to the flow of the river, with its longer axis oriented approximately due north and south. The island is about 11,000 ft long and 1700 ft wide. This unit is located in the northern one third of the island.

The southeasterly flowing Susquehanna River makes a sharp change in direction, to nearly due south, in the vicinity of Middletown. After this directional change just north of Three Mile Island, the channel widens to approximately 1.5 miles.

The island on which the site is located is basically composed of fluvially deposited sand and gravel of adequate density to support moderately heavy loads. The underlying rock is a sedimentary sequence of interbedded sandstone, shaley siltstone, and shaley claystone which belongs to the Gettysburg Formation of Triassic Age. Below the weathered surface, bedrock is capable of safely bearing loads imposed by the heaviest structures (Reference 16).

A program of test borings and a seismic refraction survey, supplemented by field geology, photogeology, and a tectonic evaluation, have been employed to determine and evaluate the suitability of the geology at the site to support a nuclear powered generating station.

Groundwater occurs above the bedrock-soil interface and varies in response to fluctuating river levels. Foundation design includes the effect of such hydrostatic variations.

The site is not considered to be deleteriously affected by faulting, and it is concluded that regional tectonic elements are inactive and present no threat to the structural integrity of local geology.

Historically, earthquakes in Pennsylvania have been infrequent and of low intensity. The resultant intensity at the site of past quakes greater than 50 miles distance was between III and IV. The highest recorded intensity within a 50 mile radius of the site was modified Mercalli VI which was rapidly attenuated with distance from the epicenter. Assuming the closest future earthquake activity would occur 5 to 6 miles north of the site at the Triassic Border Fault, which to this time has produced a maximum epicentral intensity of VI, the intensity would be attenuated to V at the site. If the focal depth is greater than the previous earthquakes, the intensity at the station site might approach the epicentral intensity and not be rapidly attenuated. Therefore, based on intensity at the site to be a low intensity VI, a ground acceleration of 0.04g should be expected, according to Figure 4 as referenced in Reference 21. The design is conservatively based on a basic ground motion of 0.06g maximum.

Acceleration response spectra for the design earthquake were partially developed using records from the March 1957 San Francisco earthquake normalized to a basic ground motion of 0.06g. Data from this earthquake were recorded by an instrument located on rock in Golden Gate Park. The instrumented records provide valuable data on the attenuation of a moderate earthquake occurring a short distance from the recording station. These field conditions are considered to

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best approximate those at the Three Mile Island site. The acceleration response spectra were further developed on the basis of the spectra for the 1940 El Centro earthquake, again normalized to a basic ground motion of 0.06g. The resultant spectra (Figure 2.7-1) therefore are controlled in the low frequency region by the El Centro Spectra.

Without qualification, the site is considered to be geologically competent to safely support a nuclear-powered generating station.

2.7.2 FIELD INVESTIGATION

Supplemented by a ground surface survey team, aerial photographs, and a geologic surface investigation, a subsurface exploration program consisting of 21 borings in conjunction with a seismic refraction survey was initiated in August and September of 1966 to establish the general range of soil depths, types, and engineering properties, and to examine and evaluate the condition of the bedrock and depth of weathering (Figure 2.7-2). This work was done to ascertain the capabilities of the site to safely and economically support a nuclear-powered generating station.

The borings and seismic profiles covered the entire length of the island. From the initial interpretation of the longitudinal seismic line and accompanying borings, the general area of the island appearing to offer the best subsurface conditions was selected, and the seismic cross lines and additional borings were concentrated in that area.

In January 1967, 20 borings were added in the plant site area for further soil investigation and to obtain reliable water table information.

The Phase I drilling program consisted of continuous sampling of overburden using the Standard Penetration Method and coring (NX core) a minimum of 10 feet into sound rock. Phase II drilling, in all but a few holes, consisted of continuous split-barrel soil sampling of overburden to bedrock and the installation of 3 inch slotted standpipes in drilled holes, comprising a system of groundwater observation points.

Forty eight additional borings were then taken in Phase III: 15 at the cooling towers, 12 at the Reactor Building, and 21 at other structure locations. These borings substantiated the conclusions drawn from the Phase I drilling program. Phase III drilling consisted of continuous split-barrel soil sampling of overburden to bedrock and core drilling into bedrock to various depths; the deepest hole penetrating 128 feet into rock.

A seismic survey was conducted (Reference 17). A twelve trace seismic refraction system with a recording oscillograph was used. Seismic energy was generated with explosives. Data were collected using two 100 foot long geophone spreads, with 20 foot geophone intervals and with three 10 foot intervals on both ends of the geophone spreads. The standard continuous profiling technique was used. In addition to end shots, two intermediate shot point determinations were made on each spread.

Seismic results were interpreted (Reference 17) and incorporated in the text of this report and on the soils profile, Figure 2.7-3. The location of the seismic lines and borings are shown on the boring location plan, Figure 2.7-4.

2.7.3 GEOLOGY

2.7.3.1 Geographic Setting

The site lies within the Gettysburg Basin section of the physiographic division known as the Piedmont Province. The topography of the area immediately surrounding Three Mile Island is of a slightly undulating nature with maximum relief of about 200 feet and highest elevation seldom above 500 feet. From the east, drainage is largely represented by the southwesterly flowing Swatara Creek which has its mouth near Middletown, and by the more westerly flowing Conewago Creek which empties into the Susquehanna River at the south end of Three Mile Island. Fishing Creek flows into the Susquehanna west of the site, and the northeasterly flowing Conewago Creek terminates at York Haven. Three Mile Island has very little relief, with elevations ranging from about 280 feet at the water's edge to slightly more than 300 ft in the north central portion.

2.7.3.2 Regional Geologic History

A geologic history of the Three Mile Island area was prepared (Reference 18), which summarized the regional structural, geologic, and tectonic aspects of this locale. The site is located in the Triassic lowland of Pennsylvania, one of a series of long narrow basins of Triassic deposits which extend in broken patches from Connecticut to North Carolina. The Triassic lowland in the vicinity of the site is referred to as the Gettysburg Basin. North and west of the Triassic lowland are the folded and thrust faulted Paleozoic rocks which comprise the Appalachian Mountains. Southeast of the Triassic lowland is the Piedmont, of Pre-Cambrian and Early Paleozoic Age, composed of granites, gneisses, and schists.

Reference 14 states after the late Paleozoic mountain making during which compression had been relieved by folding and overthrusting, large blocks of the region settled along nearly vertical joint planes, resulting in normal faulting and tilting of blocks, which formed a vast sunken basin or trough with a northeast trend and a highland to the southeast. In the following Triassic period torrential rains washed the waste from the disintegrated crystalline rocks of this highland into the trough, and arkosic sand, argillaceous sand and fine sandy mud accumulated. Much of this sediment was of a pronounced red color, owing to the fact that the iron content of the rocks was highly oxidized when they disintegrated, apparently under acid conditions. After the accumulation of a thickness of many thousands of feet of these red rocks, which now form the Newark group, further adjustment of earth blocks by tilting and normal faulting, took place on a still larger scale than before, and in places mountains were formed on the northwest side of the basin by differential uplift of several thousand feet. Catoclin Mountain in Southern Pennsylvania is an example.

The Newark group is believed to have a thickness of approximately 16,000 feet and tilts toward the northwest. This is due to subsidence and faulting along the northwest border.

Later, intrusion of Triassic diabase dikes and sills cut through the Newark group and metamorphism took place along the edges of these intrusions. Since Triassic times an erosional period has been prevalent except for glacio-fluvial deposition from Pleistocene glacial advances. The farthest advance occurred during the Illinoian stage when a lobe of the glacier extended down the Susquehanna Valley to near Fisher's Ferry, approximately 45 miles north of Three Mile Island. Glacial meltwater was responsible for the deposition of some alluvial terrace deposits now present in the region.

The tectonic history of the area (Reference 18) indicates that the region has been extremely stable for at least the last 10,000 to one million years, and that studies did not establish the

existence of minor faults or fractures at the site. Reference 18 concludes, "In the event that such inconspicuous features do exist, the probability of future movement along such zones is so small as to be negligible."

2.7.3.3 Depositional History of Three Mile Island

Three Mile Island, along with Shelley Island and other small islands in the vicinity, were formed as a result of fluvial deposition by the Susquehanna River. The carrying capacity of the river sharply decreases as the channel width increases after the stream crosses an east-west trending, very resistant diabase dike just downstream from Middletown. At this point the channel has been incised into the more easily erodable Gettysburg shale, resulting in a conspicuously wider channel. In addition, a change of flow direction occurs after the stream cuts through the diabase dike which crosses the river at that point (represented by Hill Island) and is directed along N 10°E striking vertical joints until it is again restricted and deflected by a second diabase dike just south of Three Mile Island.

Boulders carried by the glacial meltwater or transported downstream by ice rafts were first deposited in this wide-channel, low-velocity section of the river and became the nuclei for subsequent deposition of smaller materials. This gradual accretion of river sediment resulted in the growth of most of the islands in this area. Three Mile Island is made up of two such nuclei which eventually merged. This area between the two nuclei is presently represented by fine-grained deposits which were deposited in a low energy fluvial environment between the two growing depositional islands.

2.7.3.4 Bedrock Geology

The site is underlain by the sedimentary rocks of the Gettysburg shale, a part of the Newark group of Triassic Age. Two easterly trending diabase intrusions cut through the Gettysburg shale, one approximately 1 mile upstream from Three Mile Island and a second one about 0.2 miles downstream from the island. These more erosion resistant diabase dikes and the metamorphosed shales along their edge have caused restrictions in the river at Hill Island, north of the site and at Conewago Falls, downstream of the site.

The bedrock surface at the site is essentially flat and lies at approximately elevation 277 ft. Lithologic types vary from red to brown, interbedded, fine to medium-grained sandstone, shaley siltstone, and shaley claystone which range from medium-hard to hard, possessing seismic velocities in a range from 8500 to 11,500 ft/sec. There is 1 to 3 ft of weathered rock at the overburden-bedrock interface.

The regional dip of the Gettysburg shale is to the northwest at 10° to 50°. Near the site, the attitude of these beds as measured in surface outcrop, varies from N 65° to 80° E with 35° to 70° northerly dips. This variation in inclination of strata was not apparent from drill cores, as a fairly consistent dip of 37.5° to 45° was observed in the subsurface of Three Mile Island.

Well developed, near vertical jointing is conspicuous along a N10°E trend. Drill hole data substantiated surface observations regarding jointing, and revealed that healing of some joints has occurred, while others were altered by oxidation.

2.7.3.5 Faulting

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No evidence of faulting which transects the island was seen in the field from available rock exposures along the east bank of the river, or along the western periphery of the island. Aerial photographs give no suggestion of faulting through the island.

Reference 14 has mapped a hypothetical NE trending fault which they projected through Three Mile Island approximately 0.5 miles south of the site area. The substance for their structural geology was obtained by large scale surface mapping which was completed in 1929. Based on the data collected to date, this hypothetical fault does not exist. Likewise, these studies show no evidence to substantiate the projections of other hypothetical faults across the river immediately north and south of the island. The Triassic depositional basins were subjected to an appreciable degree of diastrophism during the Palisades disturbance. Most significant, however, is the fact that structural features resulting from such diastrophism are as much as 185 million years old, and that area has been extremely stable for the last 10,000 to 1 million years. In addition, rock velocities obtained from the seismic survey are normal to Triassic Gettysburg Formation rock types and indicate that the bedrock is not in a state of stress, thereby eliminating any latent threat of future strain.

A comprehensive evaluation of major tectonic elements in south central Pennsylvania was prepared as in Reference 18. It is concluded that the site is not deleteriously affected by faulting, and further, that regional tectonic elements are inactive and present no threat to the structural integrity of the local geology.

2.7.3.6 Soils

The island, as a whole, is composed of fluvially stratified subrounded to rounded sand and gravel containing varying amounts of silt, clay, and occasional lenses of clean sand. Density values range from loose to very dense, as established by Standard Penetration Tests. Boulders are presented at depth and are mainly confined to the lower portions of the soil zone on the north end of the island. Soil depths vary from approximately 6 ft at the south end of the island to a maximum of 30.0 ft near the axial intersection of the island. Depth of soil is relatively constant at about 20 ft in the vicinity of the plant site.

The overburden is made up primarily of two units. Directly overlying the sedimentary rocks is a layer of coarse sand and gravel which, at the north end of the island, contains numerous boulders and cobbles and ranges from medium dense to very dense.

Above the coarse sand and gravel is a layer of loose to medium dense, fine-grained granular material which varies from a fine silty sand and gravel, north of Station 13 + 00 (north) to a very stiff, clayey silt on the south end of the island.

Local variations do occur, and in some drill holes at the plant site, the amount of fines appeared slightly greater below a depth of about 10.5 ft. This variation is believed to be a function of a mechanical breakdown of individual shale and siltstone pebbles contained in the soil. From one half to one foot of topsoil, composed of sandy silt with much organic material, covers the island. The island had been cultivated and planted into corn fields in past years.

2.7.4 GROUNDWATER HYDROLOGY

2.7.4.1 General

Three Mile Island Nuclear Station is located on a large island in the Susquehanna River. Purely from the standpoint of geography, its groundwater conditions are unique, but relatively constant, and predictably controlled by the Susquehanna River itself. The drainage area of the Susquehanna River extends from its source at Otsego Lake, Cooperstown, New York, for a distance of 444 miles, across the states of Pennsylvania and Maryland, and terminates in the Chesapeake Bay. This major water course collects surface runoff and groundwater seeps as well as their respective contaminants from a watershed area of approximately 27,000 square miles. Three Mile Island is located in the lower third course of the river and therefore is influenced by the quantity and quality of that portion of the water from the Susquehanna River watershed upstream from the island. Average annual rainfall at the site is 40 inches.

2.7.4.2 Mode of Occurrence

The bedrock underlying the general area is composed of shales, sandstones, and siltstones belonging to the Gettysburg shale of Triassic Age. A wide range in yields occurs within the formation, with the sandstone facies normally being the best aquifers. However, in closely jointed or fractured shales, relatively high yields can occur. The alluvial deposits are not believed to be a major source of groundwater in this area.

2.7.4.3 Site Studies

Groundwater was studied at the site by means of:

- a. Standpipes to record elevation and fluctuation of water levels,
- b. A pumping-out test to determine permeability of the saturated soil,
- c. A falling head permeability test to determine permeability of soil above the water table.

Groundwater at the site occurs under water table conditions. The water table reaches its maximum elevation at the highest topographic point in the center of the island and falls off toward both the east and west shores. A variation of only about 5 ft occurs from either side to the center producing a gradient of approximately 0.6 percent toward the river.

At twenty observation points in and surrounding the plant area, water levels occurred generally at a depth in excess of 15 ft and ranged from 14 to 19 ft. The groundwater level occurred at a maximum of 6.2 ft above the top of rock with less than 1 ft of head existing above the soil-rock interface at one point of observation.

The water level of the Susquehanna River, which normally flows at elevation 277 ft, controls Three Mile Island groundwater levels. A rapid rise in the river in response to a heavy rain and thawing of ice jams during January 1967 was seen to produce significant rises in the water levels of observation wells 200 ft from the water's edge. Since a positive head exists on the island, any movement of groundwater from the plant site would be toward either channel of the river and would eventually enter the stream. The river would act as a natural boundary, limiting the dispersal of groundwater from the island to the river. Two factors are important in

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considering the possibility of infiltration of groundwater into the underlying Gettysburg shale and transmission to onshore water supplies:

- a. A maximum positive head of 6 ft exists above the impervious (relative to the soils) Gettysburg formation.
- b. Groundwater levels are higher on either shore of the river, with hydraulic gradients sloping toward the river.

In order for groundwater to move from Three Mile Island to the mainland, it would be necessary to reverse the hydraulic gradient on the mainland, which would necessitate partial dewatering of the Susquehanna River.

Further, it is unlikely that river water would ever significantly infiltrate rocks on either shore, except under sustained high capacity pumping, creating induced infiltration. A natural condition of river water flowing into the ground would not be normal to this climate and geography.

The length of time required for the groundwater to move to the river is problematical. Two types of tests were run, in an attempt to establish permeability of the soils overlying the bedrock. A pumping-out test was run on the eastern side of the island and the results determined indicated the permeability of the saturated soil zone to be on the order of 10^{-2} cm/sec. A falling head test was run in the center of the island which showed a coefficient of approximately 10^{-3} cm/sec. These results confirmed those permeabilities estimated from examining the continuous samples from the test borings. Some change in slope of the water table will occur due to the variations of the river level, and temporary minor reversal of groundwater movement will probably result during periods of high water or during periods of extended drought. Any additional impoundment of water in either channel would necessarily have an effect on the direction of groundwater movement and on the slope of the water table, but in any event the final discharge point would still be into the river.

2.7.4.4 Area Water Well Data

Most of the data on groundwater wells in the Gettysburg shale came directly from the U.S. Geologic Survey water supply paper on the groundwater resources of Harrisburg International Airport at Middletown (Reference 13). The airport is located approximately two and one half miles north of Three Mile Island.

The average strike of the formation in this area is N 43° E with dips at angles varying from 19 to 38°. The rock consists of alternating layers of fine to coarse-grained sandstone, siltstone, and shale.

Groundwater occurs under artesian conditions at the base. In 20 wells ranging from 300 to 800 ft in depth, specific capacities of the aquifers tapped ranged from 0.33 to 15.0 gpm/foot drawdown and transmissibilities varied from 1200 gpd per ft to somewhat less than 50,000 gpd per foot. The pH of wells tested ranged from 6.6 to 8.1, hardness from 137 to 826 ppm and dissolved solids from 200 to 1340 ppm. The Brunswick shale (the name applied to the eastern counterpart of the Gettysburg shale) and the Gettysburg shale were combined and included together in groundwater tabulations (Reference 15). Of 112 wells surveyed, the depths ranged from 18 to 500 feet, with an average depth of 157 feet. Yields ranged from zero to 300 gallons per minute with an average yield of 41. Six analyses were included showing the range of total dissolved solids to be between 201 and 786 ppm, hardness from 152 to 499 ppm, and iron from

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0.5 to 4.9 ppm. Since these are fairly deep rock aquifer wells where water chemistry is a function of mineralogy of the reservoir rocks, one should not expect the alluvial groundwater to be of similar chemical makeup.

2.7.5 FOUNDATION CONSIDERATIONS

As set forth in Section 2.7.3.3, the subsurface of Three Mile Island indicates that it formed first as two islands which were later joined as a result of the fluvial cycle of the Susquehanna River. Both nuclei have subsurface conditions which are favorable for safe, economic foundation design; however, the larger thickness of the uniformly dense to very dense sand and gravel layer at the north end of the island provides a better opportunity to utilize these dense soils as bearing materials and a greater opportunity to remain above the groundwater table. A plant built on the south end of the island would involve expensive rock excavations and definite dewatering costs. Also, a greater variation in the bedrock surface was detected by the seismic survey on the south end of the island, and inferences of greater variations in intensity of weathering were also noted in this area. The type of foundation, base elevation, and foundation medium for all Class I structures are listed in Table 2.7-1.

2.7.5.1 Soils

There is a high degree of uniformity of density and of soil types at the plant location, within the upper silty sand layer and the lower sand and gravel layer.

Both layers are capable of supporting foundation loads which will greatly increase with increasing depth. The actual bearing capacity of each soil layer is dependent upon the depth, size, and spacing of footings, and is greatly influenced by the elevation of the ground water table. The bearing capacity of footings placed in the lower sand and gravel layer can conservatively be permitted within a range of 5 to 8 kips per square foot (ksf) but can be considerably increased depending upon the above variables. For instance, a square footing 10 ft wide at a depth of 10 ft can be considered to safely support 14 ksf if it remains a sufficient distance above the water table, and half of that value at the water table.

Field explorations have established that the top of dense to very dense granular overburden material (in excess of 30 blows per foot (bpf) - Standard Penetration Test (SPT) at the plant site ranges from elevation 284.5 to 289.5 ft, while at the cooling towers the elevation of this horizon varies from 283.0 to 294.0 ft.

The top of the granular material with blow counts in excess of 20 bpf - spt varies from 288.6 to 295.4 ft at the cooling towers and at the plant site from 295.0 to 298.7 ft.

2.7.5.2 Rock

The fact that the bedrock surface is differentially weathered necessitates a variation in the surface bearing capacity and points out the need for assuring that any structure supported by rock be firmly established on sound rock. For preliminary planning and estimating purposes, the sound rock surface was assumed to be represented by the seismic top of rock line, defined by a compressional wave velocity of 10,000 ft/sec, shown on Figure 2.7-3, and the bearing capacity of rock to support any given foundation at a constant elevation was assumed to be equal to a maximum of 30 ksf.

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Since bedding dips at approximately 45 degrees to the horizontal, the heavily loaded foundations (those in excess of 10 ksf) were not placed up dip from sides of excavations. The shearing strength of rock parallel to unconfined bedding planes is significantly smaller than the compressive strength of the rock perpendicular to bedding planes. Also, due to the combination of vertical jointing and moderately dipping bedding planes which afford potential shear surfaces, the high differentially loaded foundations were not placed at shallow depths.

Under conditions of even loading at constant elevation, the unweathered rock at Three Mile Island is capable of safely sustaining maximum loading of 30 ksf.

At the cooling tower locations, the top of rock elevation varies from 270.5 ft to a maximum elevation of 276.2 ft. At the Reactor Building and other main structures, the top of rock varies from elevation 275.0 to 279.5 ft.

All Class I structures founded on bedrock were excavated to sound rock. Wherever necessary concrete fill was placed on top of the rock to the base elevation listed in Table 2.7-1.

The following test data were obtained on BX core samples which were sealed in cheesecloth and wax to preserve natural water until laboratory tests could be performed.

Five representative samples were tested in unconfined compression and one sample in consolidated, undrained triaxial compression. The triaxial compression test was performed with a cell pressure of 50 psi. All compression test samples, having a diameter of approximately 1.65 ± 0.02 in were trimmed into lengths of 3.03 ± 0.02 in by means of a diamond saw prior to testing.

Three unconfined compression tests were carried to failure using a rate of strain of 0.025 in/min. The remaining unconfined compression tests and the triaxial compression test were terminated at a compressive load usually in the order of 100 tons per square foot (tsf). Unload-reload cycles were carried out in four of the tests with stress excursions below the elastic limit.

The pertinent physical, strength, and elastic property test data are tabulated on the summary of laboratory test results in Table 2.7-2. It is noted that the First Load, Secant Modulus (E_c) of the test specimens averaged 0.6×10^3 ksi and that the Second Load (reload), Secant Modulus (E) averaged 1.35×10^3 ksi, based on five and four tests, respectively. The average unconfined compressive strength of the three test specimens which were loaded to failure was 87.4 tsf.

No significant difference was noted between an unconfined and triaxial test performed on specimens trimmed from the same core sample. This finding is noted to be consistent with the relatively small confining pressure utilized during triaxial testing.

The allowable bearing capacity used for the design of the foundations on rock has been determined based upon the average unconfined compressive strength of the rock.

TMI-1 DSAR

The factors of safety used to determine the allowable bearing stress were as follows:

30 ksf, for normal load:	F.S. = 6
60 ksf, for factored load:	F.S. = 3
90 ksf, for extreme load:	F.S. = 2 (airplane impact)

The soils and rock are capable of safely supporting the anticipated design loads at the proposed elevations.

2.7.5.3 Groundwater

Groundwater levels were recorded in detail around the plant site and vary from 279.0 to 282.0 feet with elevation and distance from the river. Assuming the river is maintained at elevation 277 feet, the groundwater levels should vary little from those presently observed. However, flood studies indicate a mean annual flood at the 286 feet elevation with 5, 10, 25, and 30 year flood predicted at 288.0, 290.0, 291.5, and 292.5 feet, respectively.

Deep excavations were dewatered by pumping from shallow sumps. Hydrostatic pressures were considered in the foundation design.

TMI-1 DSAR

TABLE 2.7-1
(Sheet 1 of 2)

FOUNDATIONS FOR CLASS I STRUCTURES

<u>Structure</u>	<u>Type of Foundation</u>	<u>Base Elevation</u>	<u>Foundation Medium</u>
Reactor Building	Mat	270 ft	Bedrock
Auxiliary Building	Mat	278 ft	Bedrock
Fuel Handling Building	Mat	276 ft 6 in	Bedrock
Control Building	Continuous footings under walls. Square footings under columns.	278 ft	Bedrock
Diesel Generator Building	Mat	303 ft	Compacted Backfill
Intermediate Building	Continuous footings under walls.	277 ft	Bedrock
Intake Building	Mat	259 ft 6 in 262 ft 6 in	Bedrock
Heat Exchanger Vault	Mat	267 ft 6 in	Bedrock
Access Tunnel Vault to Auxiliary Building	Mat	279 ft	Bedrock
Air Intake Structure	Mat	266 ft 278 ft	Bedrock

TMI-1 DSAR

TABLE 2.7-1
(Sheet 2 of 2)

FOUNDATIONS FOR CLASS I STRUCTURES

<u>Structure</u>	<u>Type of Foundation</u>	<u>Base Elevation</u>	<u>Foundation Medium</u>
Foundations for Storage Tanks:			
Borated Water Sodium thiosulfate (1) Sodium hydroxide (1)	Mat	300 ft	Compacted Backfill
Emergency Feedwater System Condensate	Mat	300 ft 11 in	Compacted Backfill
Underground Diesel Fuel	Mat	283 ft 6 in	Compacted Backfill

(1) – The Sodium Thiosulfate and Sodium Hydroxide Tanks are isolated from the DH and BS Systems.

TMI-1 DSAR

TABLE 2.7-2
(Sheet 1 of 1)

SUMMARY OF LABORATORY TEST RESULTS

Comprehensive Boring Sample	RB-1BX/1	RB-1BX/2	RB-1BX/3	RB-1BX/2A	RB-1BX/3A
Sample Depth (ft)	30.0-30.7	33.4-34.1	44.6-45.0	33.4-34.1	44.6-45.0
Water Content (%)	0.85	0.82	0.92	0.90	0.94
Dry Density (psf)	168.3	167.9	167.0	166.0	166.1
Elastic Modulus					
$E_{(psi)}$	0.67×10^6	0.50×10^6	0.62×10^6	0.90×10^6	0.30×10^6
$E_{(psi)}$	1.6×10^6	1.2×10^6	1.3×10^6	1.3×10^6	-
Strength (tsf)	90.7	91.6	82.9	-	-
Type Failure	Chisel- Shattered	Chisel- Shattered	Chisel- Shattered	-	-

2.8 SEISMOLOGY

A seismicity and response spectra were conducted and is represented in Reference 19.

2.8.1 SEISMICITY

The seismicity analysis indicated that Pennsylvania is relatively inactive seismically, based upon 200 years of historical data and 40 years of instrumental data. Earthquakes in the greater Pennsylvania area, which have or might have affected the site, were studied and their intensity at the site was determined by attenuation of the earthquake with distance.

Earthquakes which have affected the site were studied in two categories, those within a 50 mile radius of the site, and those beyond this radius. Nearly all the earthquakes considered were felt over very limited areas, which are generally elliptical in shape, and aligned with the general structural trend of the area. The high attenuation of these earthquakes indicates that their foci must have been close to the earth's surface.

An estimate of the maximum expected intensity of an earthquake was predicated on the assumption that the activity which would affect the site would originate along the border fault of the Triassic Lowland, five to six miles north of the site. The highest intensity earthquake to occur on this fault has been modified Mercalli VI. The intensity of such an earthquake at the site would be V, based upon the rapid attenuation of similar earthquakes in the area and along the fault. Consideration was given, however, to the future occurrence of an earthquake at a greater depth in the fault, with a conservative assumption that the resulting intensity at the site could approach the epicentral intensity, and not be rapidly attenuated.

The conservative estimate of the maximum earthquake intensity to be expected at the site is a low intensity VI. Using relationships published in Reference 21, this intensity corresponds to a ground acceleration of 0.04g. The design is conservatively based on a basic ground motion of 0.06g maximum.

2.8.2 ACCELERATION RESPONSE SPECTRA

The objective of this study was to establish an acceleration response spectra corresponding to a possible earthquake of low intensity VI (Modified Mercalli), as established by the seismicity analysis.

Figure 2.7-1 indicates the average smooth response spectra derived from the ground motions of the 1957 Golden Gate Park, San Francisco earthquake together with the revised acceleration spectra reflecting the greater response at lower frequencies based upon the 1940 El Centro spectra. The resulting acceleration response spectra (Figure 2.7-1) were developed as described in Subsection 2.7.1.

2.9 GEOLOGIC GLOSSARY

2.9.1 GENERAL

Alluvial	Stream transported
Aquifer	A water producing geologic unit
Artesian water	Groundwater that rises above the level at which it is encountered
Bedrock	Rock underlying soil, sand, clay, boulders, etc.
Diastrophism	The process of deformation of the earth's crust
Dike	A tabular body of igneous rock that cuts across the structure of adjacent rocks or cuts massive rocks
Decomposition	Breakdown and decay of rock to soils condition
Dip	The angle at which a stratum or any planar feature is inclined from the horizontal
Fault	A break in rock along which there has been displacement of the two sides relative to one another parallel to the break
Fluvial	Produced by river action
Folds	Bends in the strata
Footwall	Rock below a fault plane
Fractures	Breaks in rock due to folding or faulting
Groundwater	Subsurface water in the zone of saturation
Hanging wall	Rock above a fault plane
Ice raft	Floating ice which transports rock particles and other materials
Intrusion	An invasion of older rock by igneous rock
Joint	A fracture or parting along which little or no movement has occurred parallel to the fracture
Lithology	Composition and texture of rock
NX	Core size - 2.155 inches
Overburden	Soil, sand, gravel, boulders, etc., that overlie bedrock

TMI-1 DSAR

Overthrust	In faulting, the process of thrusting the hanging wall (relatively) over the footwall
Sandstone	Sandstones are composed of cemented grains of sand-size material
Shale	A laminated sediment, composed predominantly of particles of clay size
Siltstone	A very fine grained clastic rock composed predominantly of particles of silt size
Tectonics	The study of the structure of the earth in response to earth movements
Water table	The upper surface of a zone of saturation
Weathering	Mechanical and chemical action on rocks which eventually reduces them to a soil state

2.9.2 HARDNESS CRITERIA

Rock hardness as indicated on the boring logs is a qualitative property determined by field classification and is used as an aid in analyzing general subsurface conditions. Actual rock hardness is difficult to define; therefore, an arbitrary method of hardness determination was defined by the following criteria:

Very soft	Rock disintegrates to touch; exhibits destructive alteration; can be considered as hard to very hard soil.
Soft	Rock is coherent but is broken very easily by thumb pressure at sharp edges and crumbles with firm hand pressure. Exhibits partial alteration.
Medium hard	Rock appears to be relatively hard, but small pieces can be broken by considerable hand pressure. Usually exhibits high degree of oxidation.
Hard	Rock cannot be broken by thumb pressure. May exhibit minor oxidation (i.e., some oxide staining or discoloration, etc.).
Very hard	Rock broken with difficulty by hammer. Exhibits no effects from weathering.

TMI-1 DSAR

2.10 REFERENCES

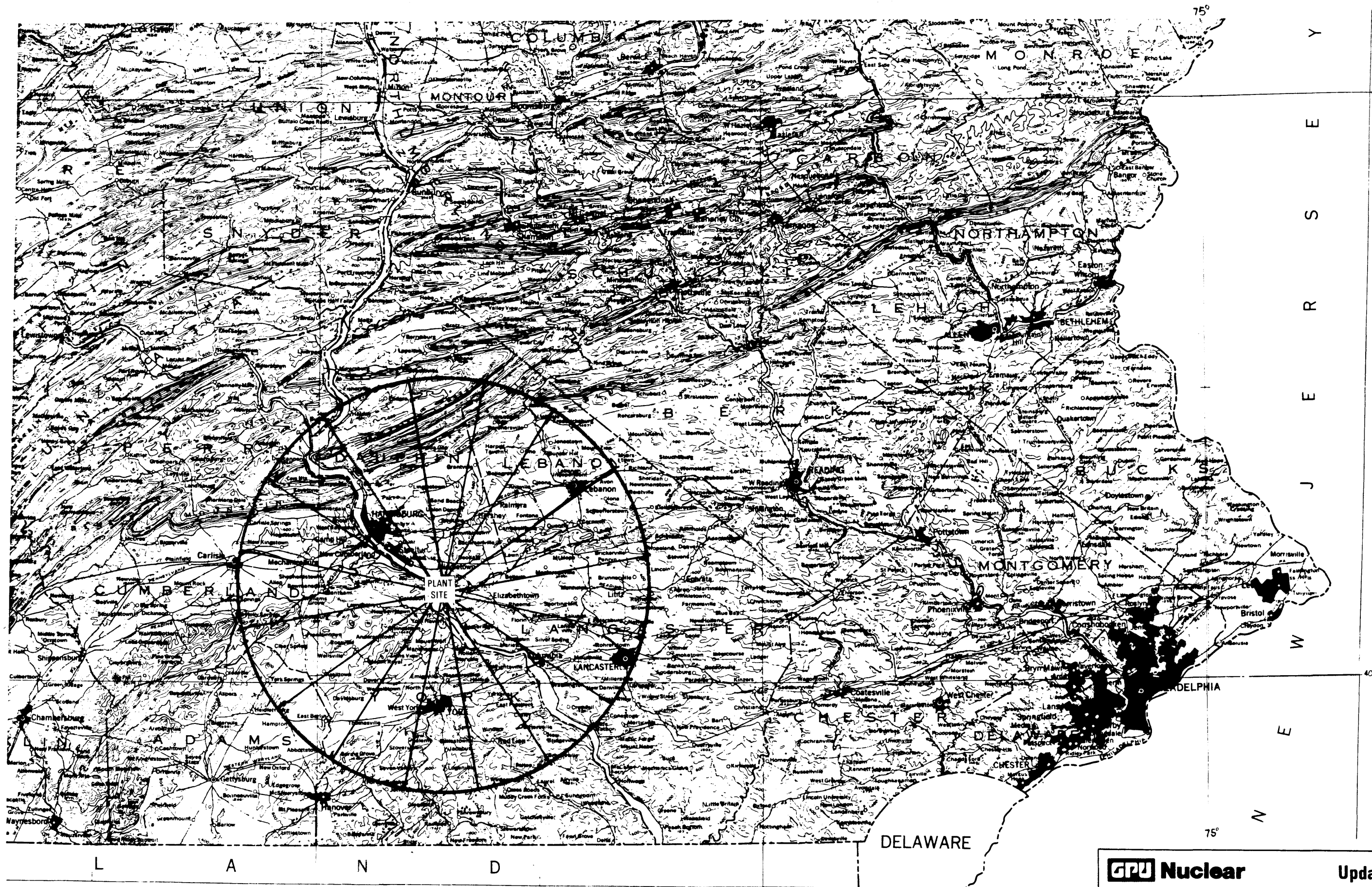
1. Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations, National Transportation Safety Board, calendar years 1978 through 1987.
2. U.S. Weather Bureau, Office of Climatology, Tornado Listings by Year, 1917-1963, and U.S. Department of Commerce, ESSA, Storm Data, 1966-June 1969.
3. Thom, H.C.S., "Distributions of Extreme Winds in the United States," ASCE Proceedings, April 1960.
4. Huss, P.O., "Relation Between Gusts and Average Wind Speeds," Daniel Guggenheim Airstrip Inst., Report No. 140, Akron, Ohio, 1946.
5. Gifford, F.A., "Consequences of Activity Release," Nuclear Safety, 2 (2), p. 57 (1960).
6. Pasquill, F., "Estimation of the Dispersion of Windborne Material," Meteorology Magazine 90, (1963) pp. 33-49, February 1961.
7. Slade, D.H., "Estimates of Dispersion from Pollutant Releases of a Few Seconds to 8 Hours in Duration," Environmental Science Services Administration, Technical Note 39 - ARL-3, April 1966.
8. Deleted.
9. Deleted.
10. Deleted.
12. Deleted.
12. Commonwealth of Pennsylvania, Department of Forests and Waters, Water Resources Bulletin No. 1, Pennsylvania Streamflow Characteristics, Low Flow Frequency and Flow Duration, prepared in cooperation with U.S. Department of Interior, Geological Survey, April 1966.
13. Meisler, H. and Longwill, S.M., Groundwater Resources of Olmstead Air Force Base, Middletown, Pennsylvania, U.S. Geological Survey Water-Supply Paper 1539-H, 1961, U.S. Government Printing Office, Washington, D.C., 34 pp.
14. Stose, G.W. and Jonas, A.E., Geology and Mineral Resources of the Middletown Quadrangle, Pennsylvania, U.S. Printing Office, Washington, D.C., 86 pp.
15. Hall, G.M., Groundwater in Southeastern Pennsylvania, Pennsylvania Geological Survey Bulletin W-2; 1934, 255 pp.
16. Gray, C., Geologic Map of Pennsylvania, Pennsylvania Geological Survey Map #1, 1960.

TMI-1 DSAR

17. Sexton, T.F., Seismic Survey, Three Mile Island, Pennsylvania, Weston Geophysical Engineers, Inc., Weston, Massachusetts, September 9, 1966; USAEC Docket No. 50-289, Three Mile Island Nuclear Station, Unit 1 Preliminary Safety Analysis Report.
18. Kauffman, M.E., Geologic Investigation of Three Mile Island and Vicinity, January 14, 1967; USAEC Docket No. 50-289, Three Mile Island Nuclear Station, Unit 1 Preliminary Safety Analysis Report.
19. Holt, R.J., Seismicity and Response Spectra Analysis, Proposed Nuclear Power Plant, Three Mile Island, Susquehanna River, Pennsylvania, Weston Geophysical Research, Inc., January 19, 1967; USAEC Docket No. 50-289, Three Mile Island Nuclear Station, Unit 1 Preliminary Safety Analysis Report.
20. Snyder, F.F., "Hydrology of Spillway Design: Large Structures - Adequate Data," Journal of the Hydraulics Division, Proceedings of the ASCE, Volume 90, No. HY3, May 1964, Part I, p. 239.
21. U.S. Nuclear Regulatory Commission, Nuclear Reactors and Earthquakes, TID 7024
22. Annual Review of Aircraft Accident Data, U.S. General Aviation, National Transportation Safety Board, calendar years 1986 and 1987.
23. Jane's, All the Worlds Aircraft, 1967-68 and 1990-91 Editions.
24. Gilbert/Commonwealth, Inc., Report 2661, Review of Low River Flow Hydraulic Analyses, June 1987.
25. Deleted.
26. U.S. Army Corps of Engineers (Baltimore District), Emergency Plan for Raystown Lake, Susquehanna River Basin, Raystown Branch Juniata River, September 1986.
27. Pennsylvania Dept. of Environmental Resources, Water Resources Bulletin, No. 6, December 1970.
28. USGS Water Resources Data - Pennsylvania, Water Year 1987, Volume 2.
29. Evacuation Travel Time Estimates for the Three Mile Island Nuclear Generating Station Plume Exposure Pathway Emergency Planning Zone, Final Report February 1994.
30. Deleted.
31. Deleted.
32. PLG File No. G98TM-4L, Accident X/Q Values for TMI-1, Keith Woodard to Ken Boughton, dated July 23, 1998.
33. GPU Nuclear Calculation C-1101-826-E540-020, Atmospheric Transport X/Q's Using ARCON96 Code – Control Room Habitability.

TMI-1 DSAR

- 34. Deleted.
- 35. Three Mile Island Unit 1 Nuclear Generating Station Easement License and Attachment Agreement by and Among Jersey Central Power & Light Co., Metropolitan Edison Co., PA Electric Co., as Sellers, and AmerGen Energy Co., LLC, as Buyer, Dated December 20, 1999.
- 36. Deleted.
- 37. Analysis of Three Mile Island Area Hydrographic Survey, TMI Report No. 90-3071 (Parsons E&C Report No. 53753307-CE-DR-1), November, 2005, prepared by: J.L. Caves, Spotts, Stevens and McCoy, Inc. and approved by C. Rickett, Worley Parsons.
- 38. C-1101-122-E410-003, Rev 0, "River stage discharge and discharge frequency analysis."
- 39. 990-3093, Susquehanna River PMF Sensitivity Evaluation, AMEC Letter 10-14-2011.



DSAR-2.FIG-1 REV. 2

GPU Nuclear

TMI Unit-1

General Area Map

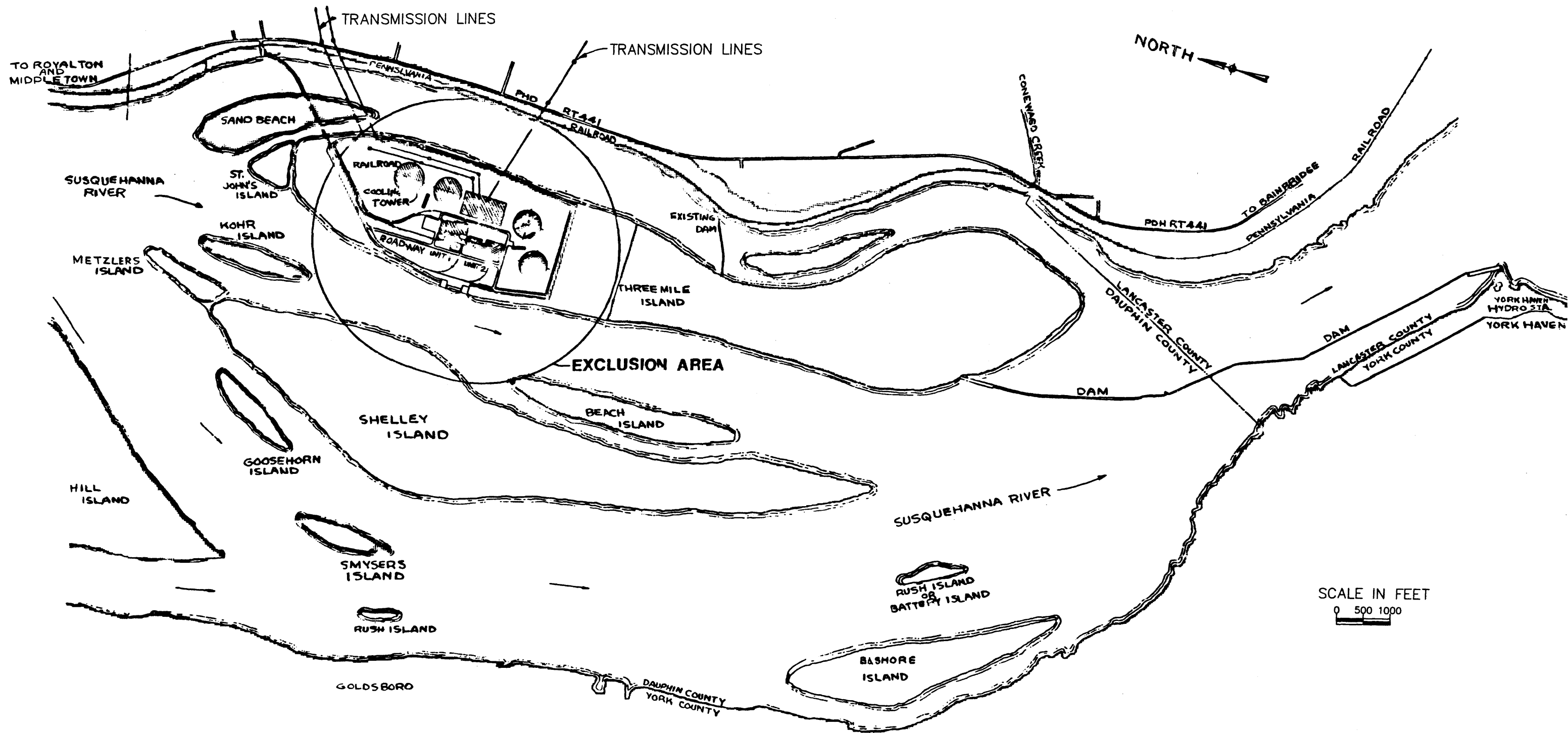
Update -1
7/82

Fig. 2.1-1



CONTOUR INTERVAL 20 FEET
DATUM IS MEAN SEA LEVEL

GPU Nuclear	Update - 1
TMI Unit 1	7/82
Site Topography	
5 Mile Radius	
	Fig. 2.1-2



DSAR-2.FIG-3 REV. 2

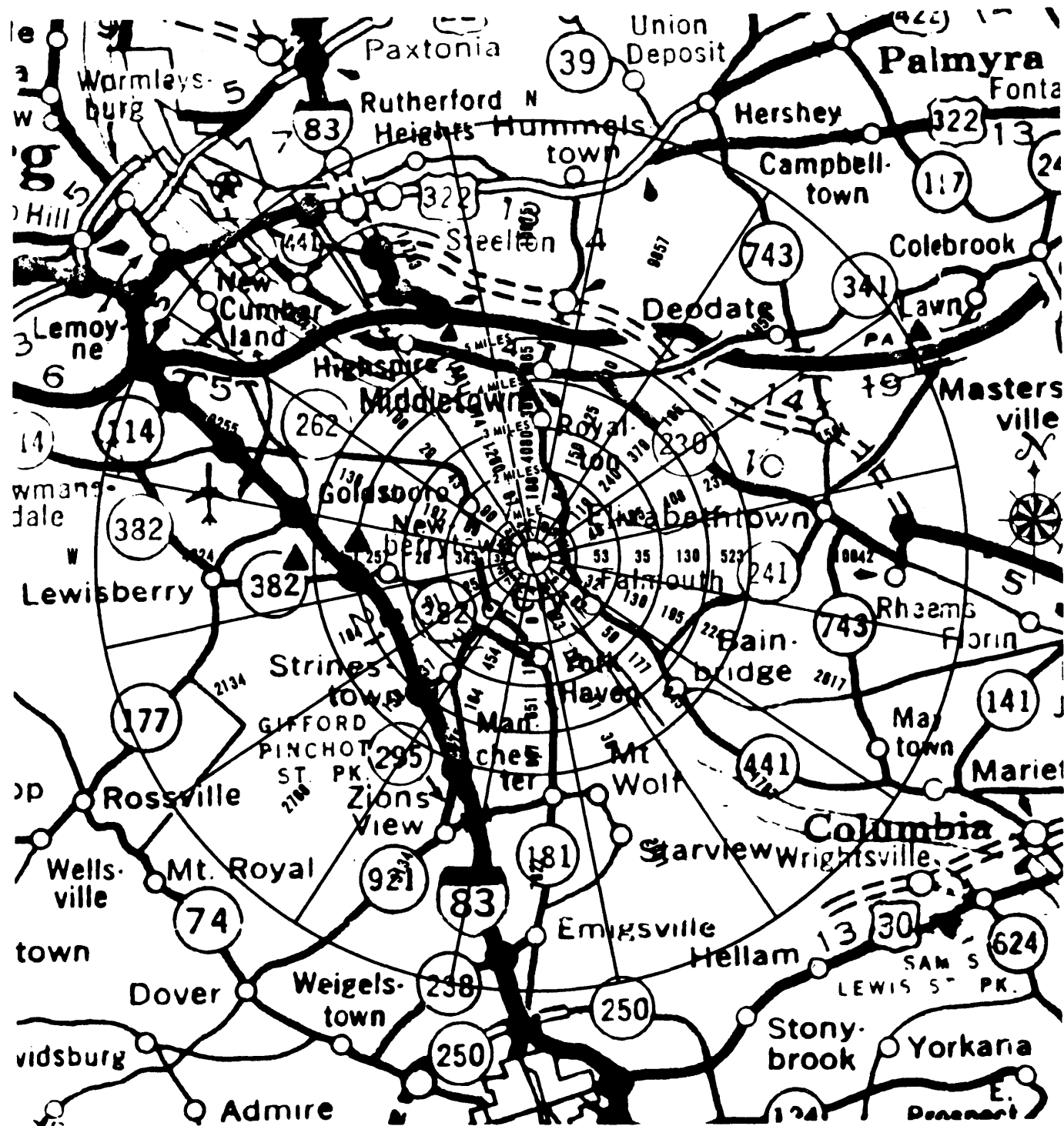
TMI Unit 1

Update - 16
4/02

Extended Plot Plan - TMINS

CAD FILE: 6633R15.DWG

Fig. 2.1-3



DISTANCE	TOTAL	CUM. TOTAL
0-1 MI.	590	
1-2 MI.	1,790	2,380
2-3 MI.	7,304	9,684
3-4 MI.	8,938	18,620
4-5 MI.	7,378	25,998
5-10 MI.	113,603	139,596

* EXCLUSION RADIUS

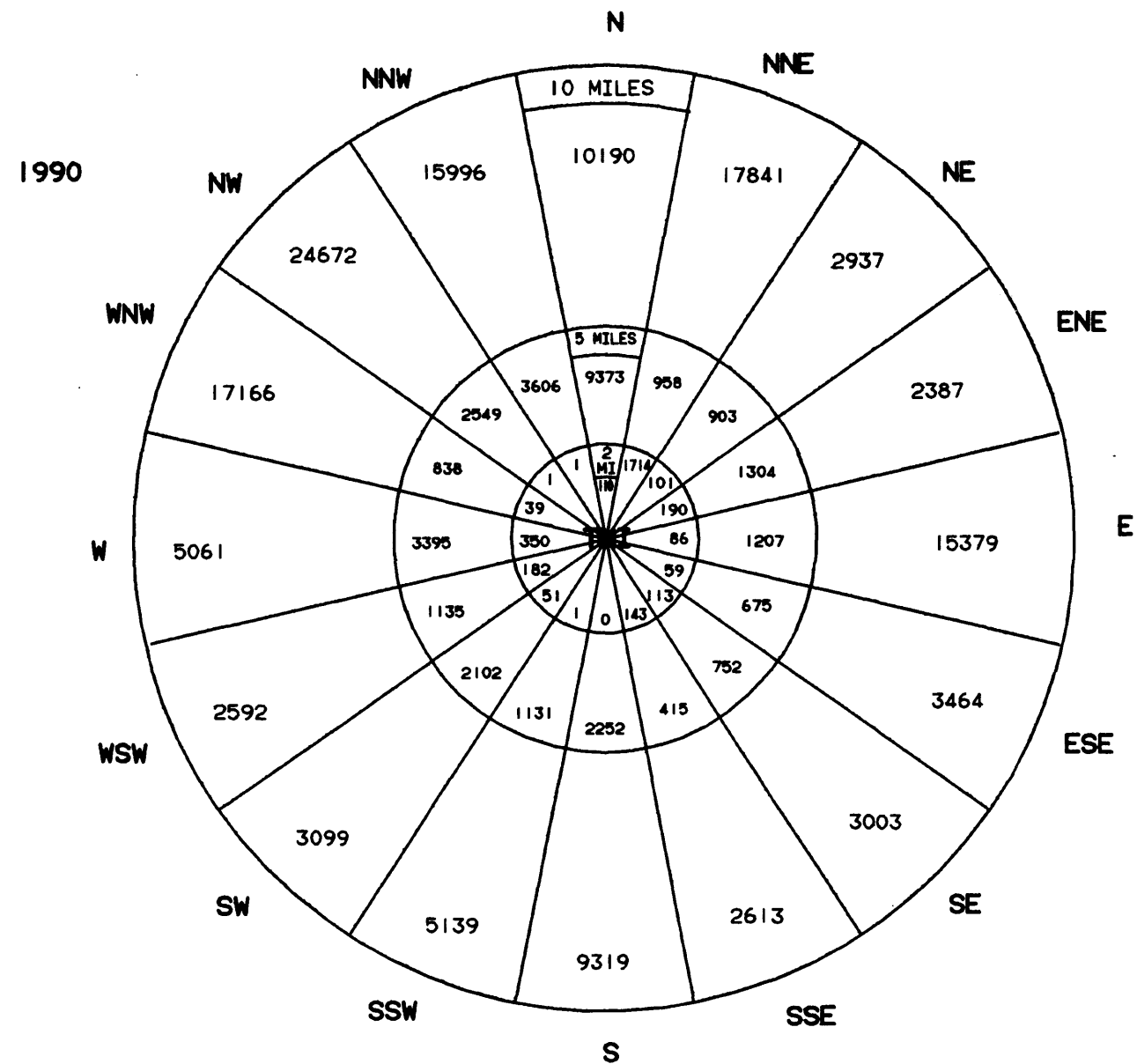
GPU Nuclear

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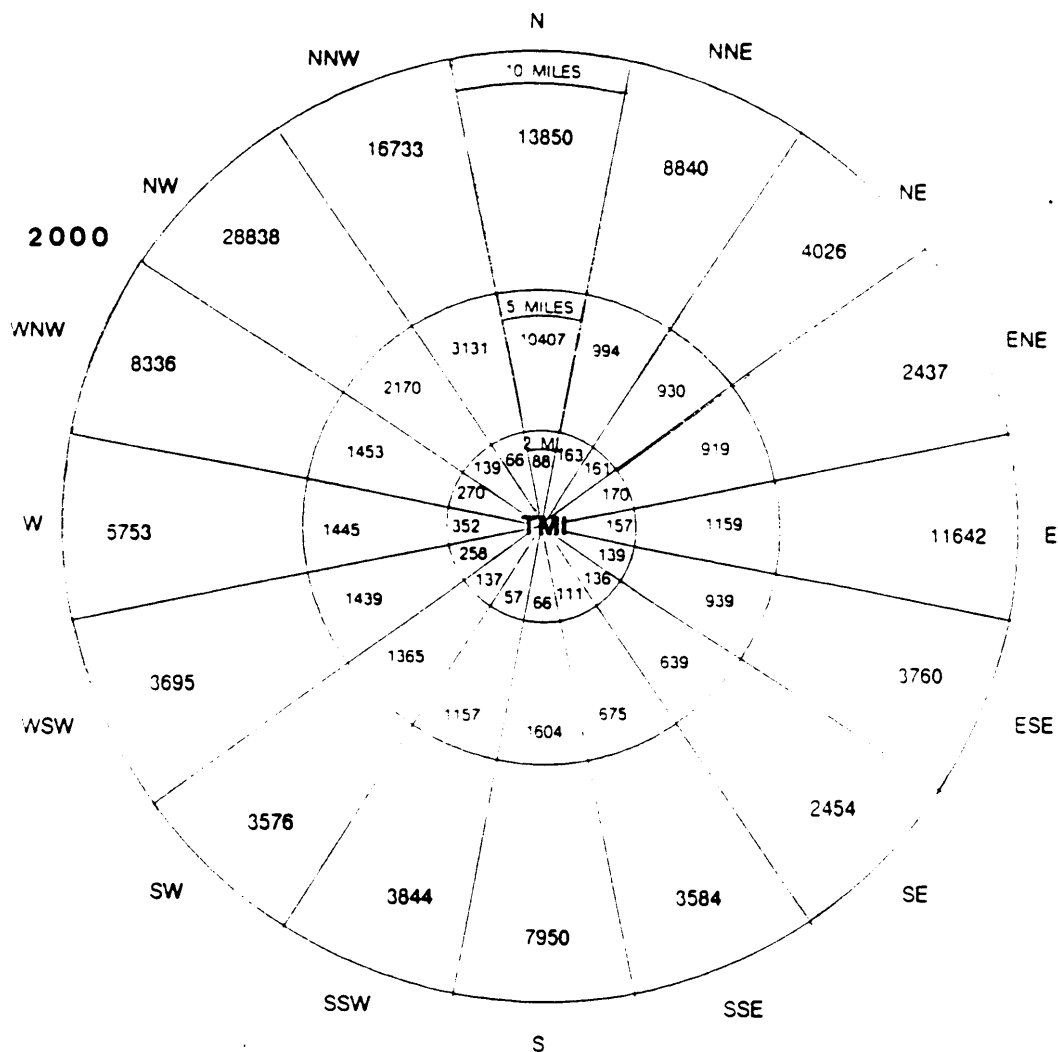
TMI Unit-1

7/82

Population Distribution 0 to
10 miles - 1970



1990 POPULATION WITHIN 10 MILES OF TMI-1 BY RING AND SECTOR (POPULATION BASED ON 1990 CENSUS DATA)																	
MILES FROM TMI-1	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	RING TOTAL
0-1	5	58	42	84	57	13	12	12	0	0	0	0	0	0	0	0	283
1-2	105	113	59	106	29	46	101	131	0	1	51	182	350	39	1	1	1315
2-3	3418	137	174	126	74	140	210	214	172	441	244	169	137	62	40	1358	7116
3-4	5515	474	423	581	516	222	135	167	827	276	903	501	1665	302	190	1062	13759
4-5	440	347	306	597	617	313	407	34	1253	414	955	465	1593	474	2319	1186	11720
5-10	10190	17841	2937	2387	15379	3464	3003	2654	9319	5139	3099	2592	5061	17166	24672	15996	140899
0-10	19673	18970	3941	3881	16672	4198	3868	3212	11571	6271	5252	3909	8806	18043	27222	19603	175092



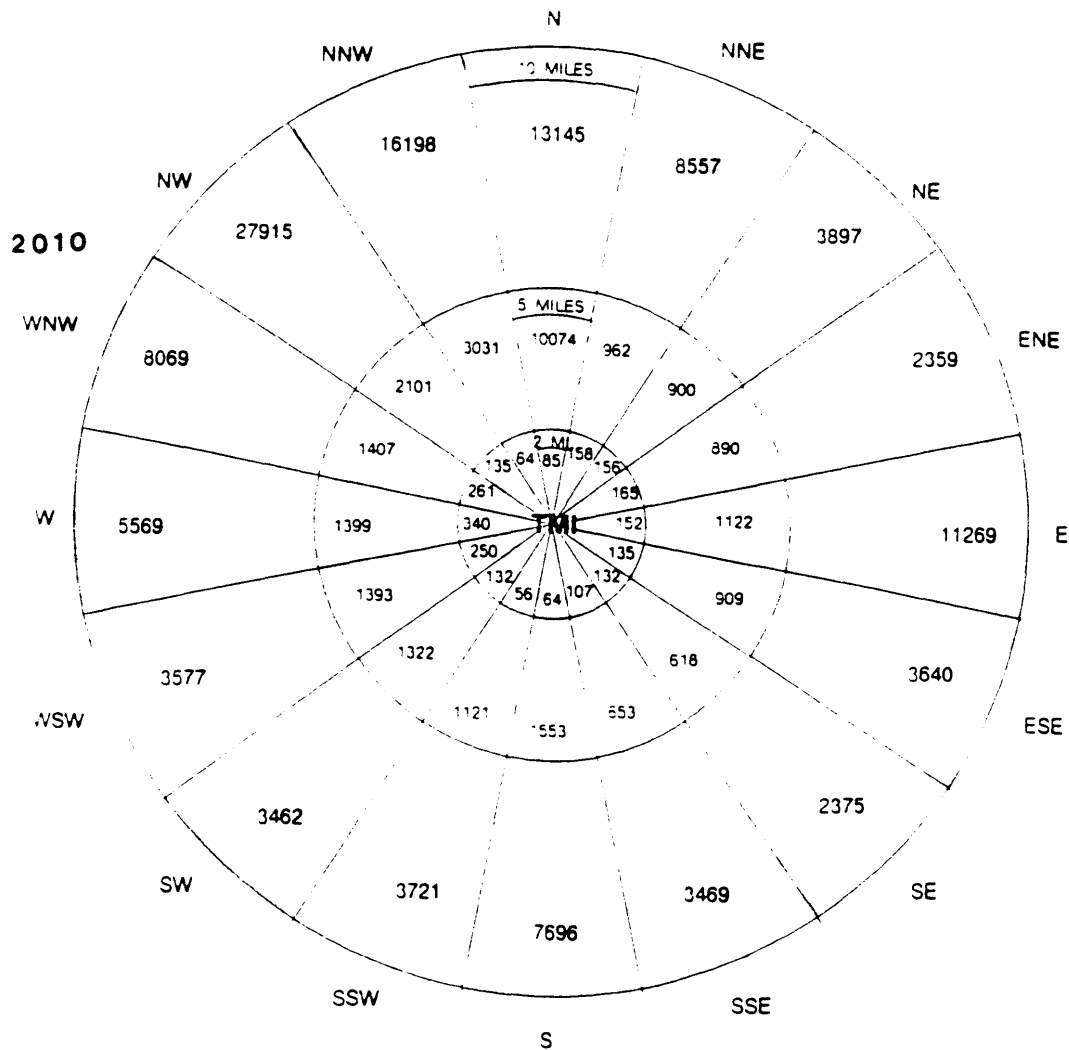
GPI Nuclear

TMI Unit 1

**Population Distribution 0 to
10 miles - 2000**

Update - 8

7/89



2010 POPULATION WITHIN 10 MILES OF TMI-1 BY RING AND SECTOR (POPULATION PROJECTION BASED ON 1980 CENSUS DATA)																		RING TOTAL
MILES FROM TMI-1	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW		
0-1	7	29	27	38	39	36	33	24	29	15	15	15	18	15	13	9		356
1-2	78	129	129	129	113	99	99	83	35	41	117	235	322	248	122	55		2029
2-3	2956	262	214	214	214	151	151	156	428	316	331	331	331	317	171	1220		7761
3-4	5418	314	300	300	414	277	206	293	663	428	465	465	465	476	254	783		11519
4-5	1700	386	386	376	494	481	281	204	462	377	526	597	603	614	1676	1028		10170
5-10	13145	8557	3897	2359	11269	3640	2375	3469	7696	3721	3462	3577	5569	8069	27915	16198		125182
0-10	23566	9677	4953	3414	12543	4684	3125	4229	9313	4898	4916	5220	7308	9737	30151	19293		157027

GPU Nuclear

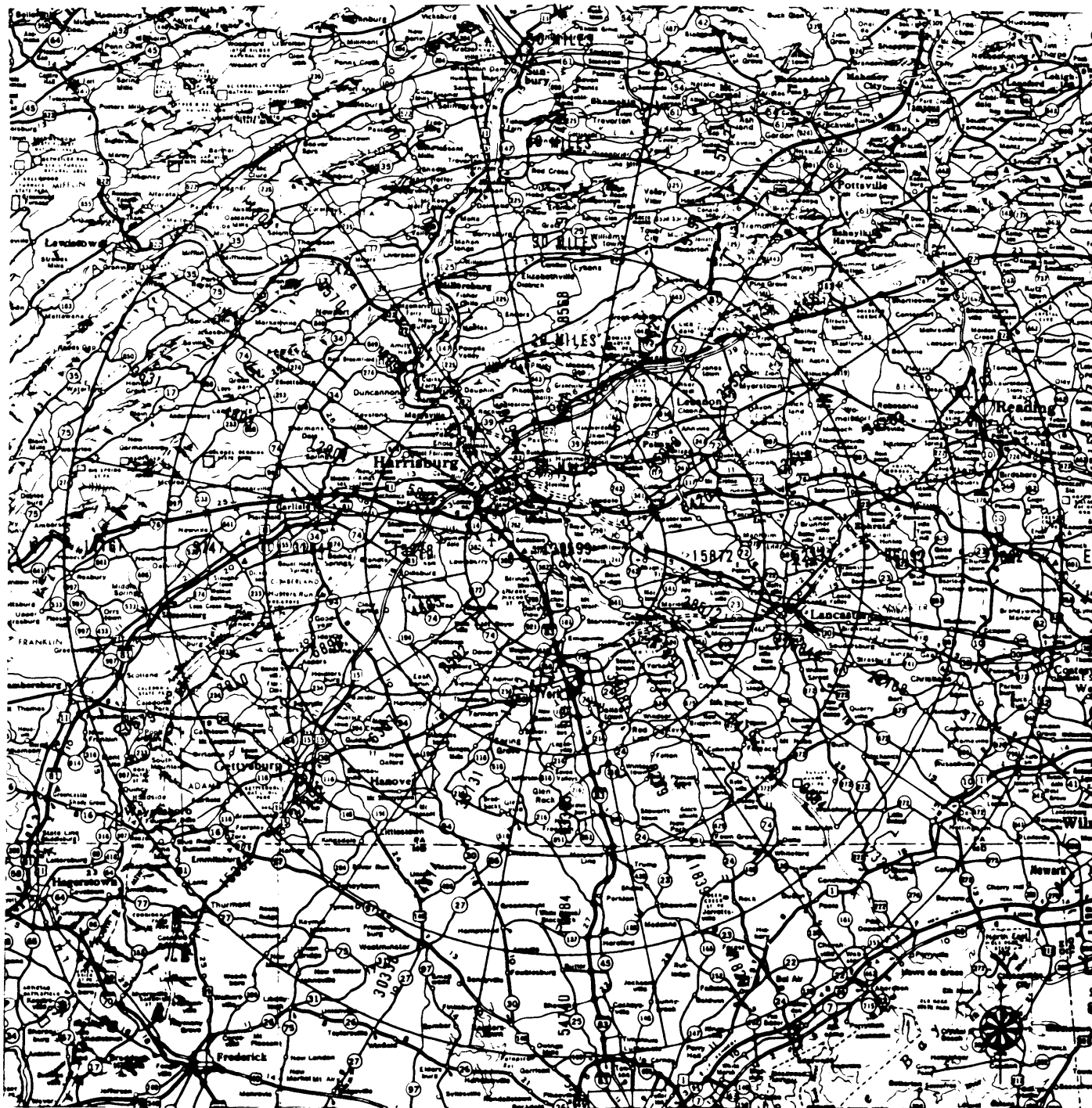
TMI Unit 1

**Population Distribution 0 to
10 miles - 2010**

Update - 8

7/89

Fig. 2.2-5



DISTANCE	TOTAL	CUM. TOTAL
0-10 MI.	139,599	
10-20 MI.	485,609	625,208
20-30 MI.	373,072	998,280
30-40 MI.	238,281	1,236,511
40-50 MI.	659,206	1,865,717

DSAR-2.FIG-9 REV. 2

GPU Nuclear

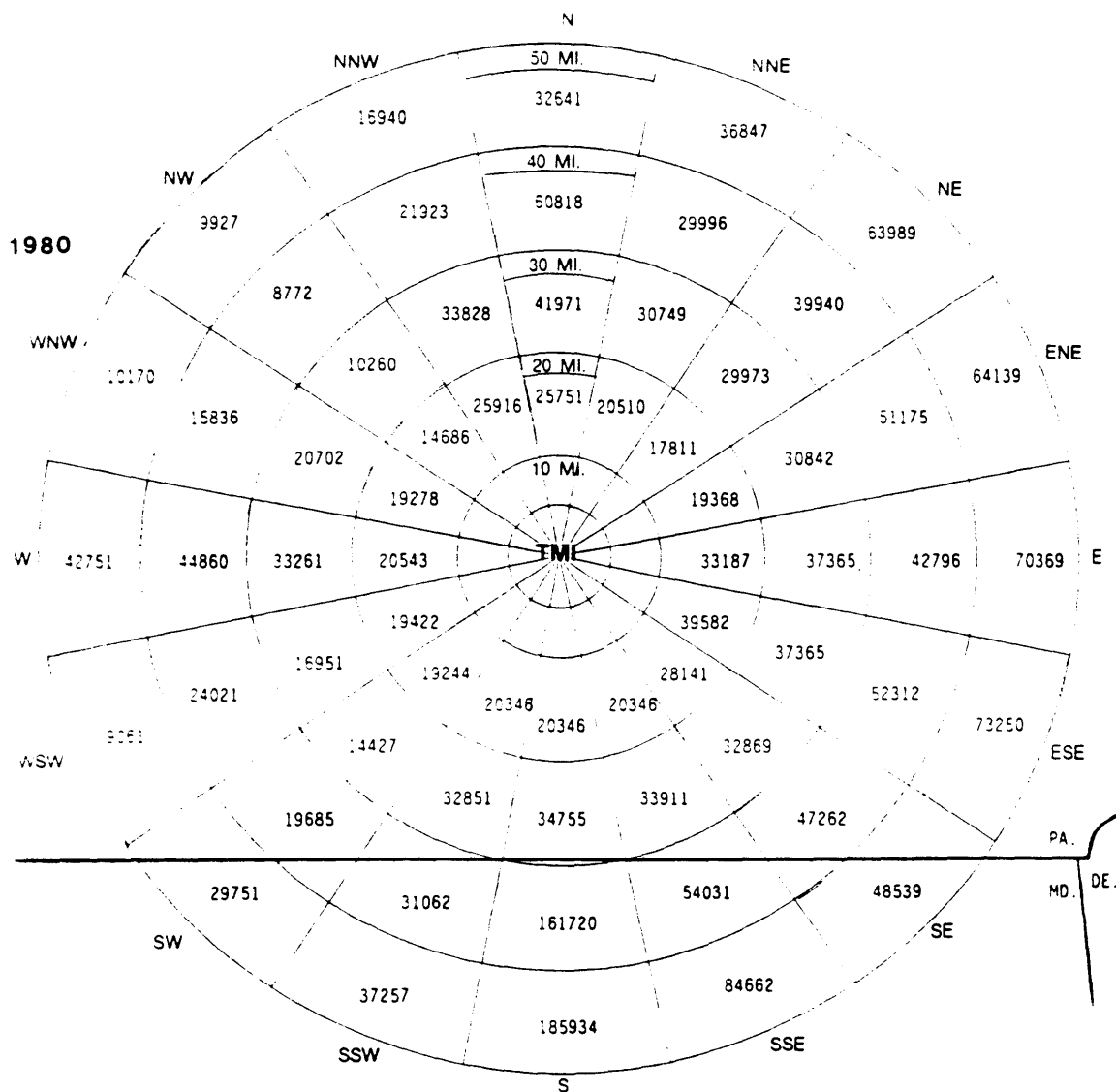
Update - 1

TMI Unit-1

7/82

Population Distribution 0-50 miles
1970

Fig. 2.2-6



1980 POPULATION WITHIN 50 MILES OF TMI -1 BY RING AND SECTOR (BASED ON 1980 CENSUS DATA)																	
MILES FROM TMI -1	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	RING TOTAL
0-10	25095	10305	5274	3637	13356	4967	3328	4503	9916	5213	5235	5558	7782	10370	32107	20544	167210
10-20	25751	20510	17811	18388	33187	39582	28141	20346	20346	20346	18244	18422	20543	18278	14888	25816	364478
20-30	41971	30748	29873	30842	37365	37365	32889	33811	34755	32851	14427	18851	33281	20702	10280	33828	472080
30-40	60818	29996	39940	51175	42796	52312	47262	54031	161720	31062	19685	24021	44860	15636	8772	21923	708208
40-50	32641	36847	63989	64139	70369	73250	48539	84662	185934	37257	29751	9061	42751	10170	9827	18840	816227
0-50	186275	126406	156887	169161	197073	207496	160139	187453	412671	126728	86343	75012	148189	76355	75753	118151	2528201

GPU Nuclear

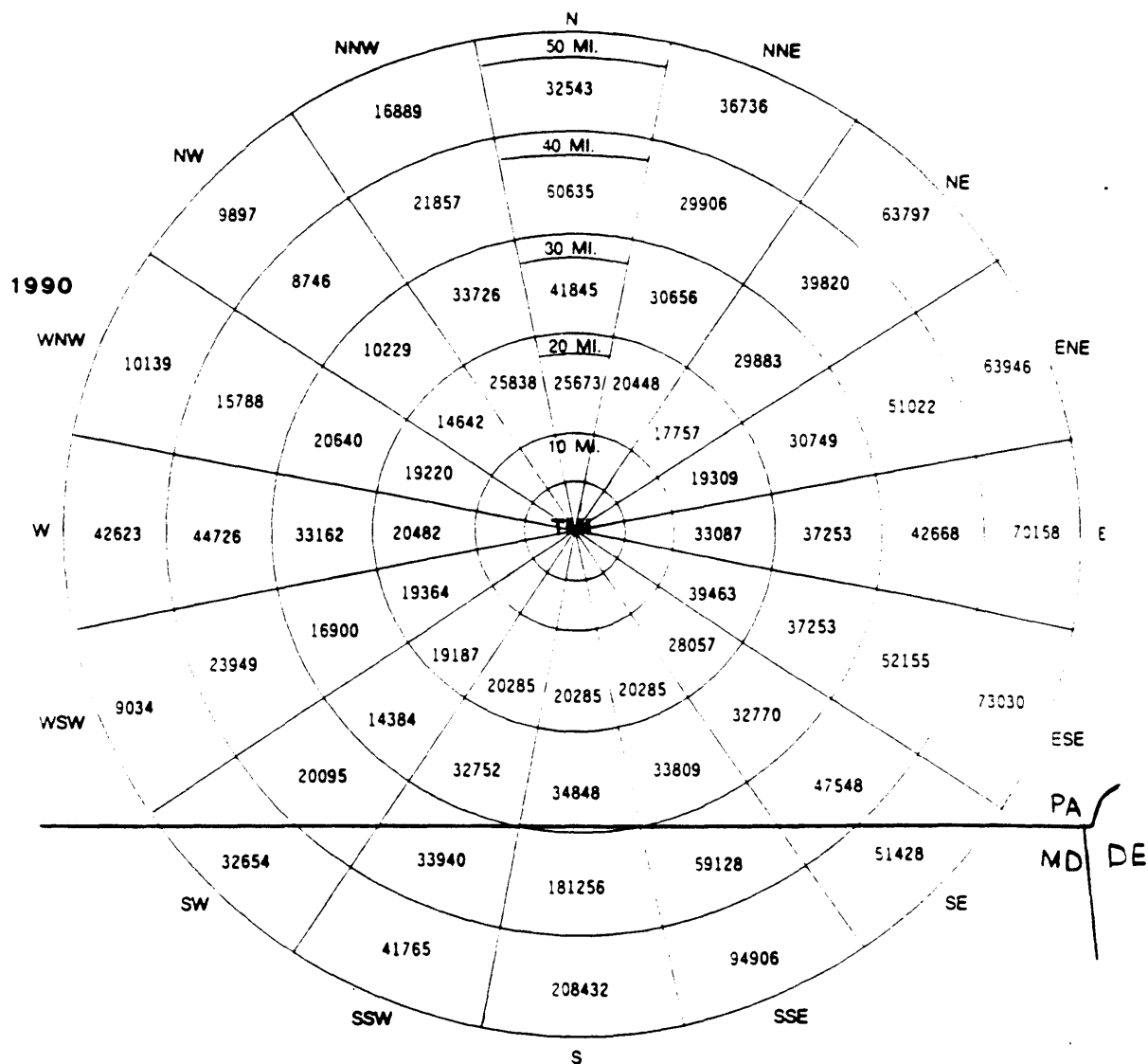
TMI Unit 1

**Population Distribution 0 to
50 miles - 1980**

Update - 8

7/89

Fig. 2.2-7



1990 POPULATION WITHIN 50 MILES OF TMI-1 BY RING AND SECTOR (POPULATION PROJECTION BASED ON 1980 CENSUS DATA)																	
MILES FROM TMI-1	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	RING TOTAL
0-10	25020	10275	5280	3625	13317	4872	3318	4490	9887	5198	5218	5542	7758	10338	32011	20482	166712
10-20	25673	20448	17757	18308	33087	39463	28057	20285	20285	20285	19187	19384	20482	19220	14642	25838	363382
20-30	41845	30858	29883	30749	37253	37253	32770	33808	34848	32752	14384	16900	33162	20640	10229	33726	470880
30-40	80635	29808	39820	51022	42888	52155	47548	58128	181256	33940	20085	23949	44726	15788	8746	21857	733238
40-50	32543	36736	63797	63946	70158	73030	51428	94906	208432	41785	32854	9034	42623	10139	9897	16889	857878
0-50	185718	128022	158517	168852	196483	208873	163122	607526	454708	133940	91537	74788	148751	78125	75526	118793	2987078

GPU Nuclear

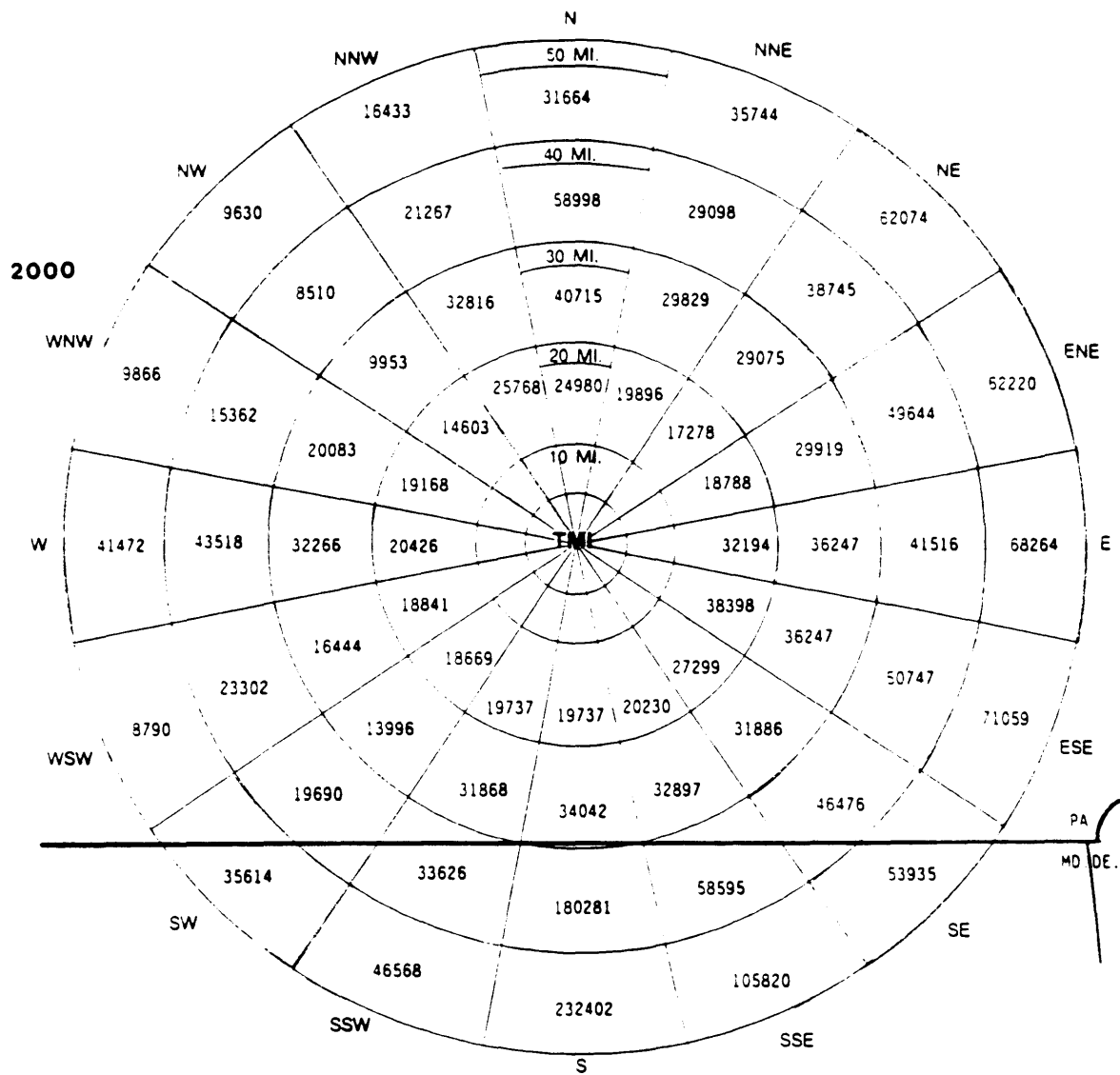
TMI Unit 1

**Population Distribution 0 to
50 miles - 1990**

Update - 8

7/89

Fig. 2.2-8



MILES FROM TMI-1	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	RING TOTAL
0-10	24345	9887	5117	3528	12958	4838	3229	4370	9820	5058	5078	5382	7550	10058	31147	19928	162214
10-20	24880	19888	17278	18788	32194	38388	27299	20230	19737	19737	18888	18841	20426	19188	14803	25788	358012
20-30	40715	29829	29075	29819	38247	38247	31886	32887	34042	31888	13888	18444	32288	20083	9953	32816	458281
30-40	58888	29088	38745	48844	41516	50747	48478	58595	180281	33828	19880	23302	43518	15382	8510	21287	718375
40-50	31884	35744	62074	62220	68264	71059	53835	105820	232402	48388	35814	8790	41472	9888	9830	16433	881585
0-50	180703	124584	152289	164087	191179	201288	182828	221813	478083	138857	93048	72788	145233	74537	73843	118214	2587438

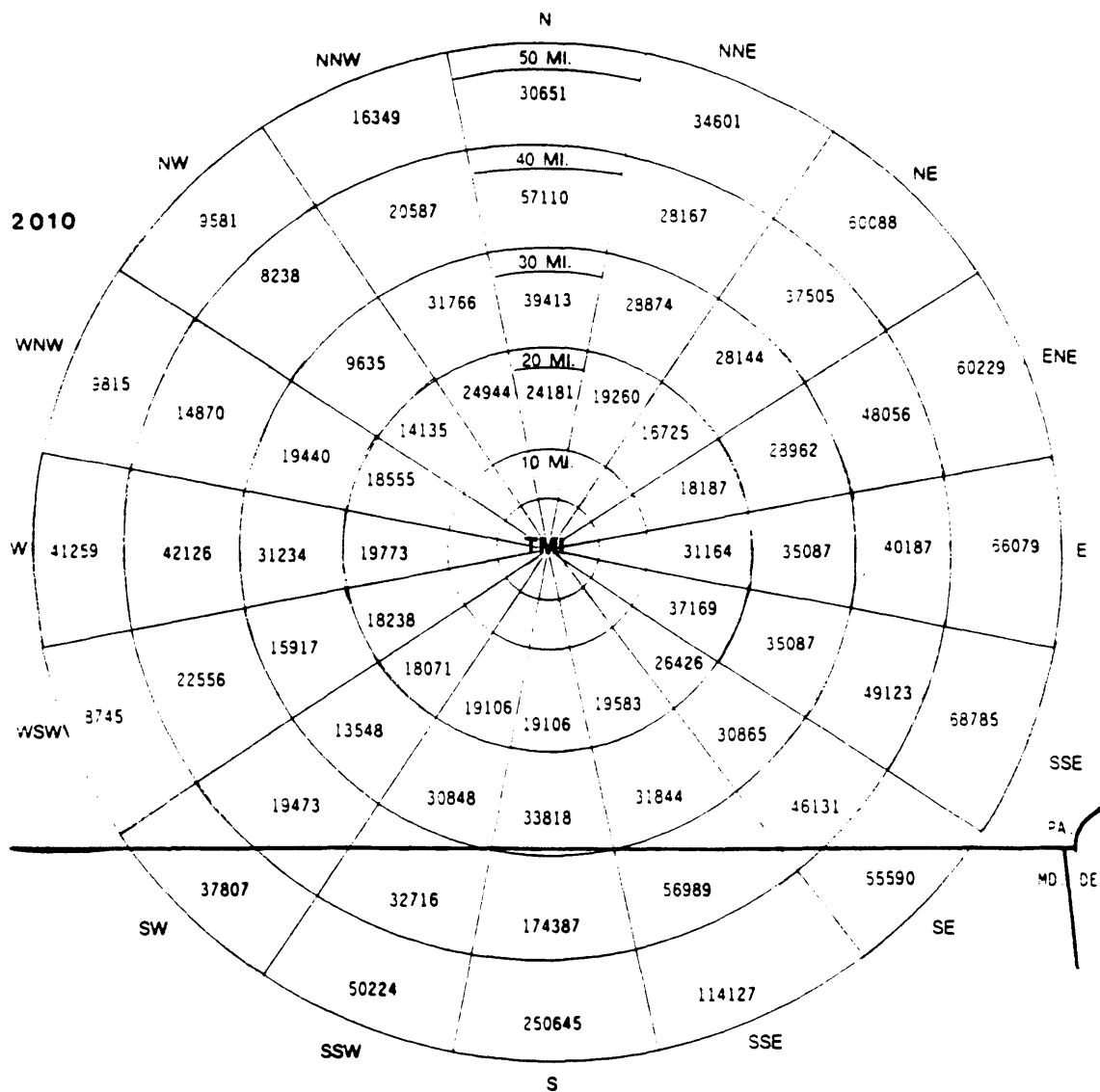
GPO Nuclear

TMI Unit 1

**Population Distribution 0 to
50 miles - 2000**

Update - 8

7/89



2010 POPULATION WITHIN 50 MILES OF TMI-1 BY RING AND SECTOR (POPULATION PROJECTION BASED ON 1980 CENSUS DATA)																		RING TOTAL
MILES FROM TMI-1	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW		
0-10	23588	9877	4853	3414	12543	4684	3125	4229	9313	4898	4816	5220	7308	9737	30151	19293		157027
10-20	24181	19280	16725	18187	31164	37188	26426	19583	19106	19106	18071	18238	19773	18555	14135	24944		344618
20-30	38413	28874	28144	28882	35087	35087	30865	31844	33818	30848	13548	15917	31234	19440	9635	31766		444482
30-40	57110	28167	37505	48056	40187	48123	48131	58888	174387	32716	18473	22556	42126	14870	8238	20587		688219
40-50	30651	34801	60088	60229	66079	68785	55590	114127	250645	50224	37807	8745	41259	9815	9581	16349		914573
0-50	174820	120578	147416	158847	185080	194848	182137	226772	487299	137782	93814	70676	141889	72416	71738	112937		2558822

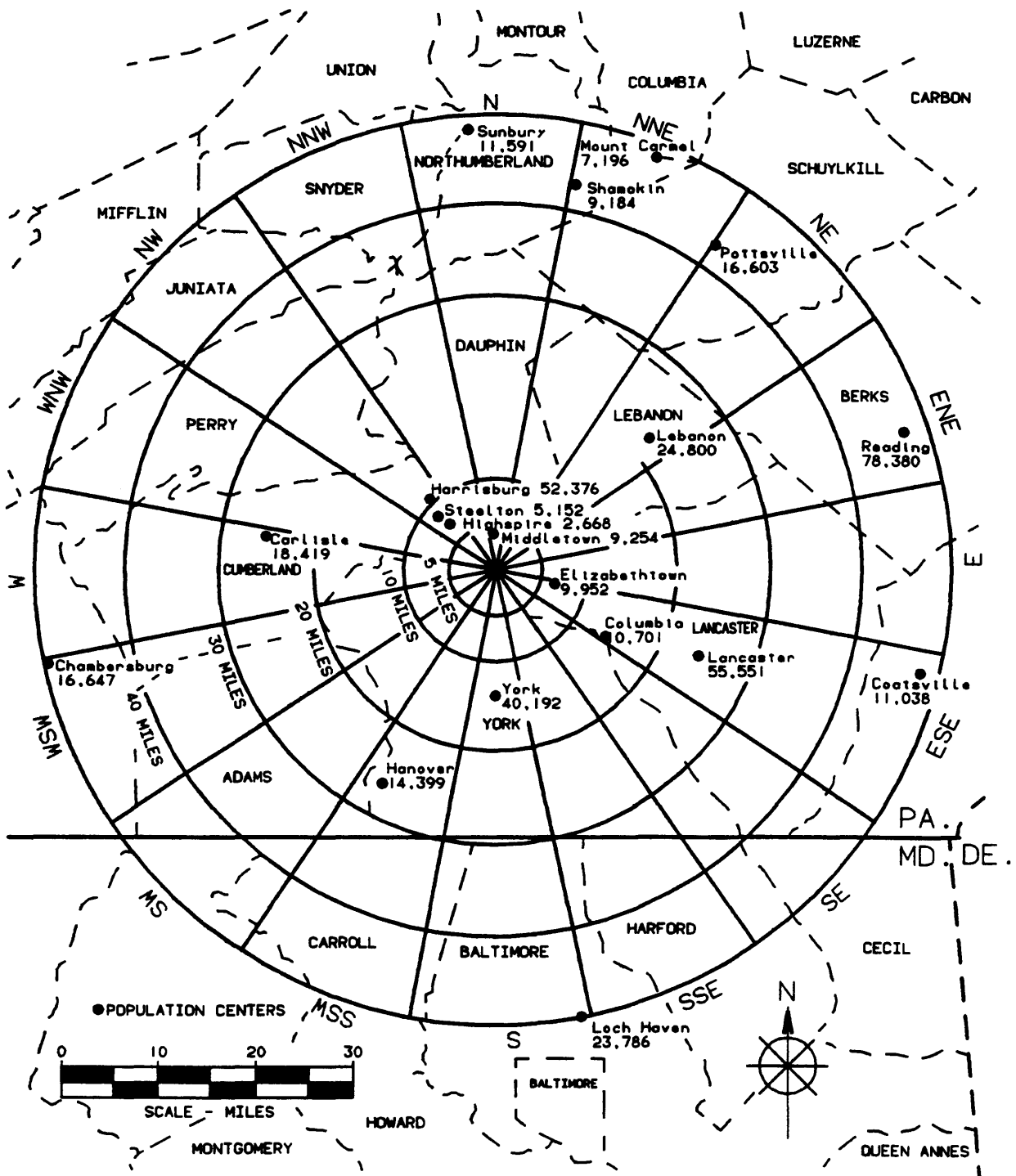
GPU Nuclear

TMI Unit 1

**Population Distribution 0 to
50 miles - 2010**

Update - 8

7/89



TMI DSAR

Figure 2.2-12

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CONTROL INTERVAL 20 FEET
DATUM IS MEAN SEA LEVEL

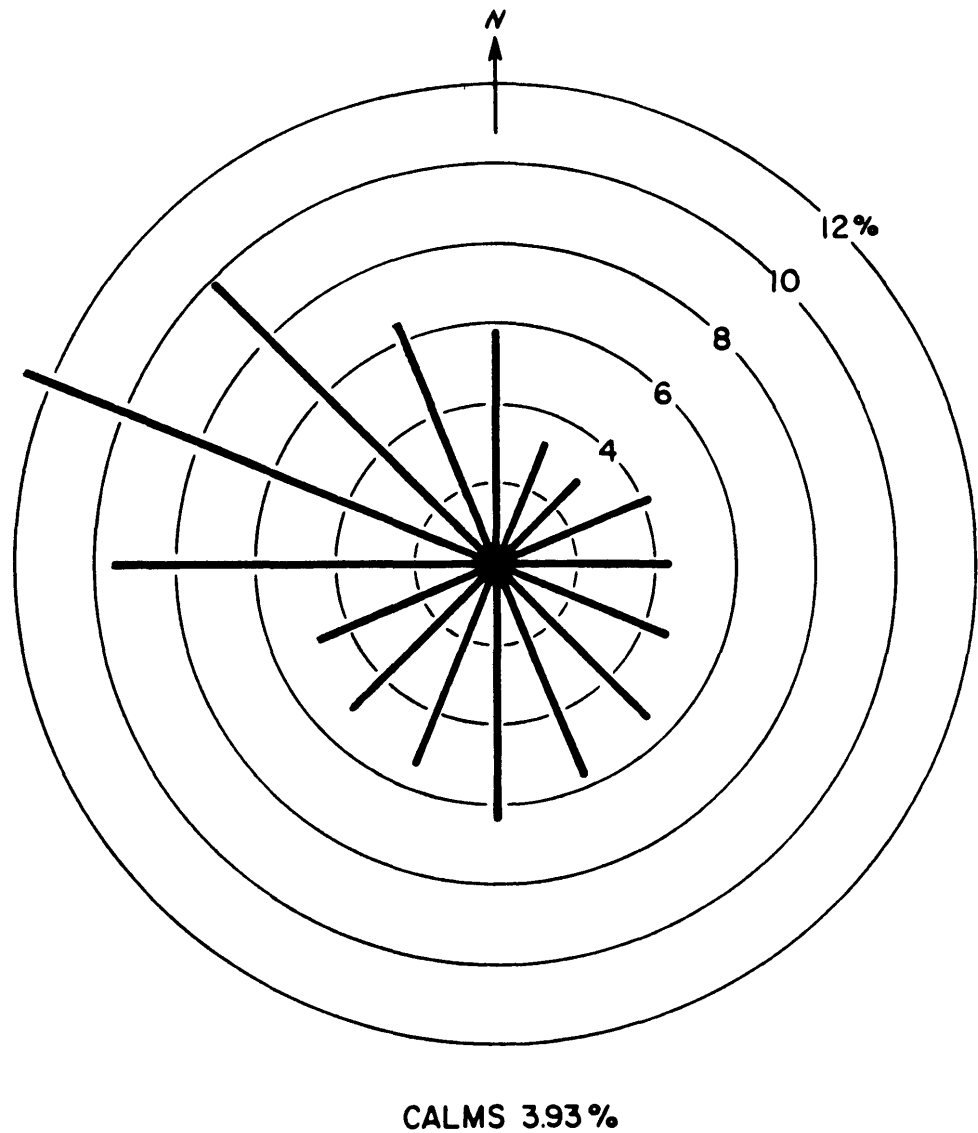
DSAR-2.FIG-2

TMI DSAR

Figure 2.4-2

Deleted

WIND ROSE
THREE MILE ISLAND TOWER
5/15/67 thru 5/14/69



GPU Nuclear

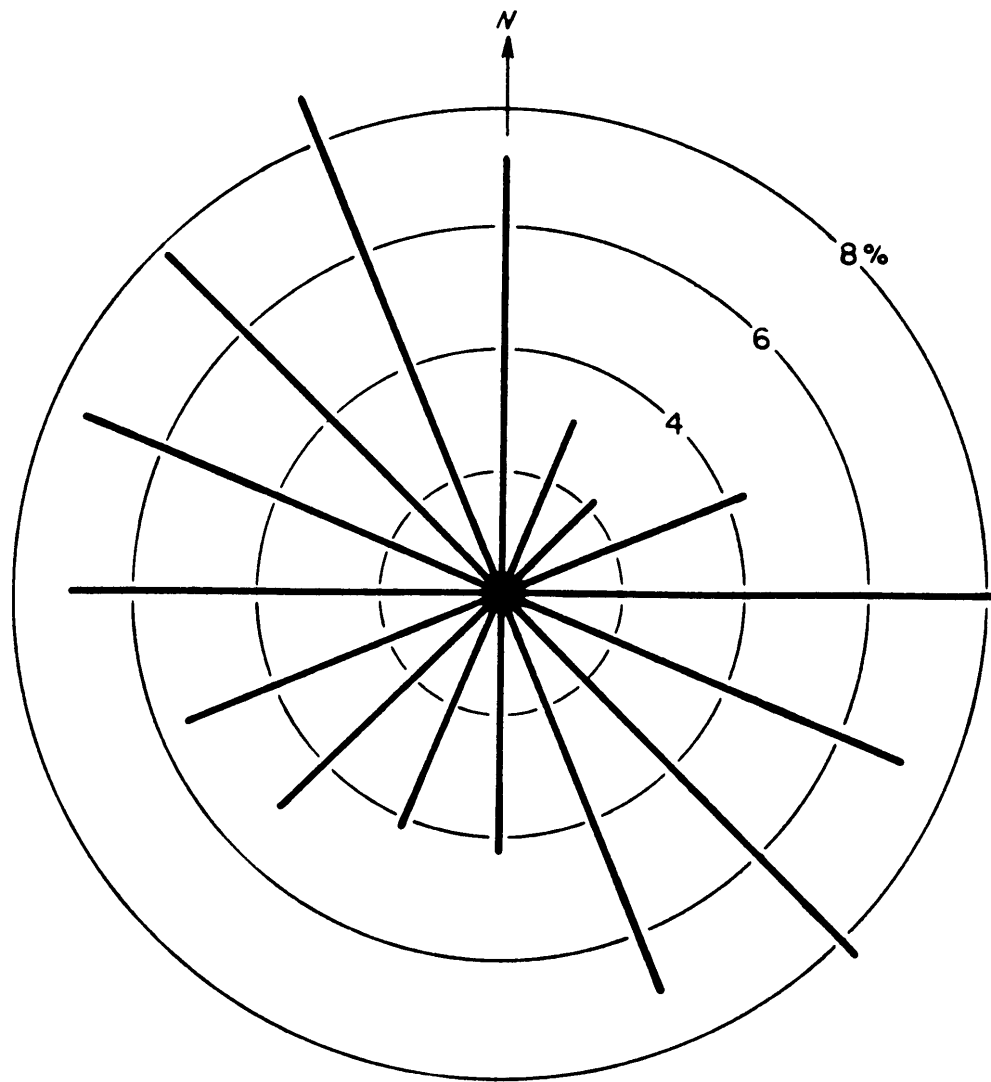
TMI Unit-1

Update - 1

7/82

Wind Rose Three Mile Island Tower

WIND ROSE
CRAWFORD STATION
5/15/67 thru 5/14/68



CALMS 1.17 %

GPU Nuclear

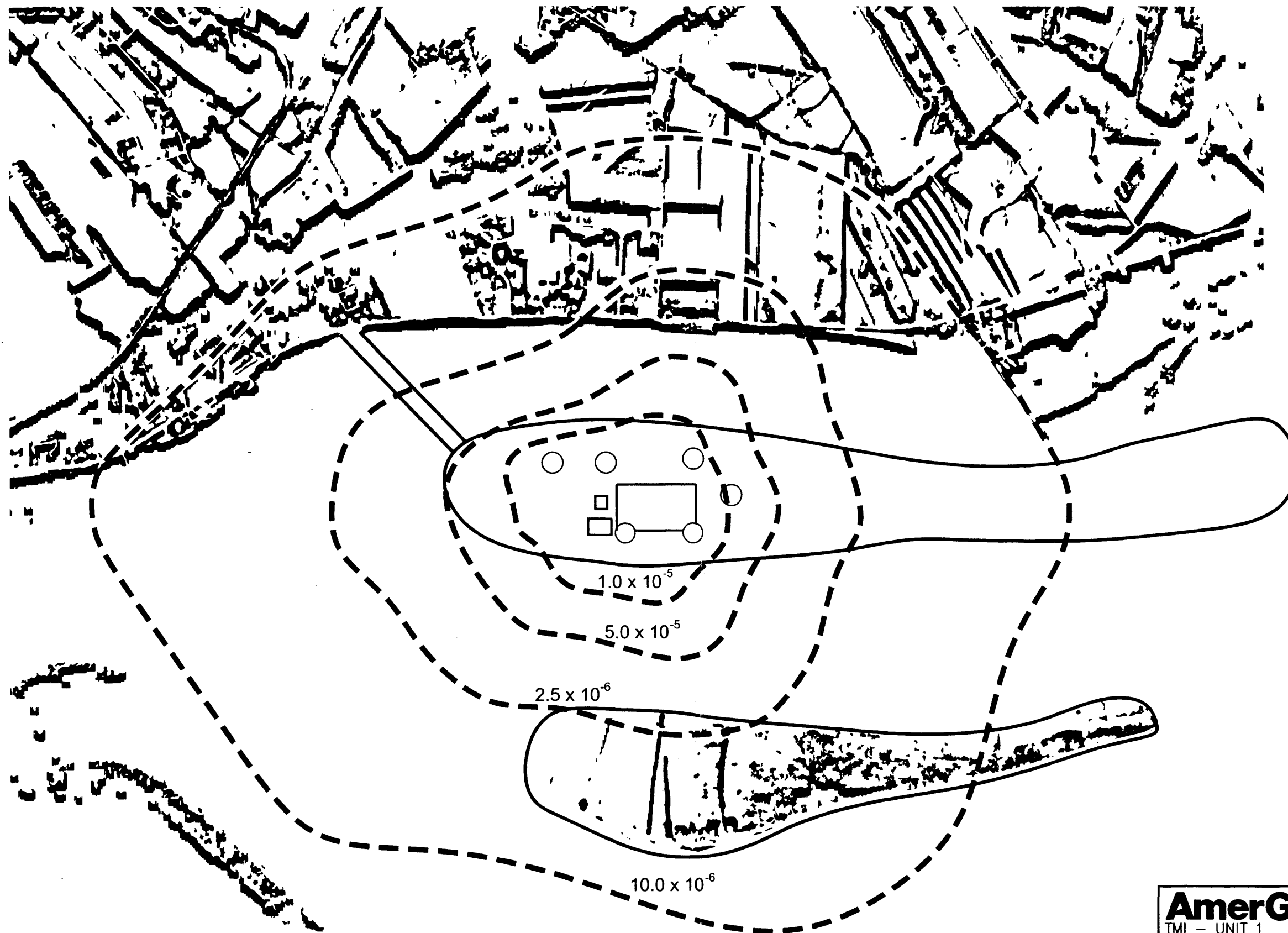
TMI Unit-1

Wind Rose Crawford Station

Update - 1

7/82

Fig. 2.5-2



DSAR-2.FIG-20 REV. 2

AmerGen TMI - UNIT 1	UPDATE - 15
	4/00
ANNUAL AVERAGE ATMOSPHERIC DISPERSION ESTIMATE, χ/Q ISOPLETH (1967-1969)	
CAD FILE: 6648R15.DWG	FIG 2.5-3

TMI DSAR

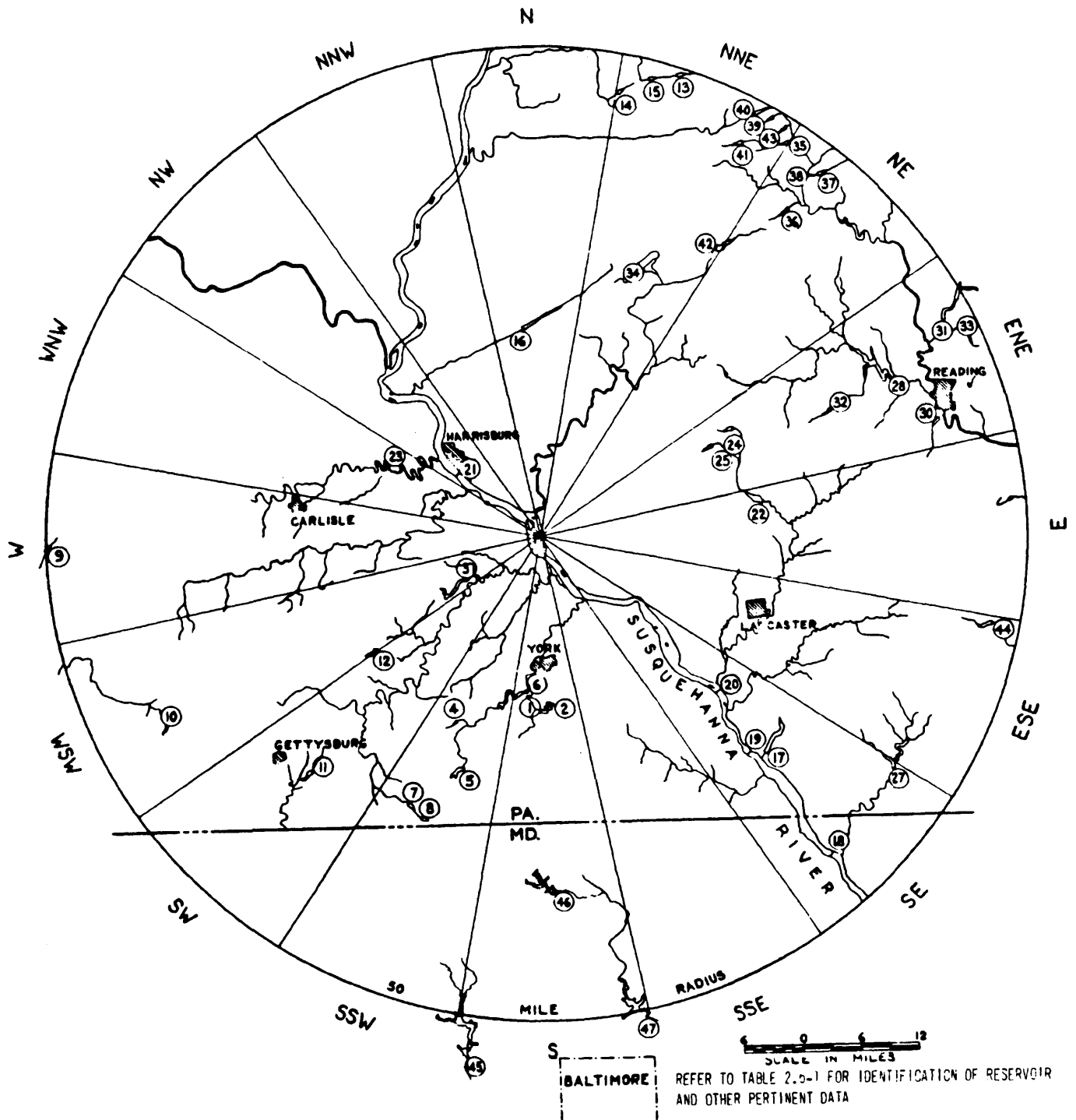
Figure 2.5-4 through 2.5-6

Deleted

TMI DSAR

Figure 2.5-8 through 2.5-28

Deleted



REFER TO TABLE 2.6-1 FOR IDENTIFICATION OF RESERVOIR
AND OTHER PERTINENT DATA

NOTE:

1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

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TMI - UNIT 1

UPDATE - 15

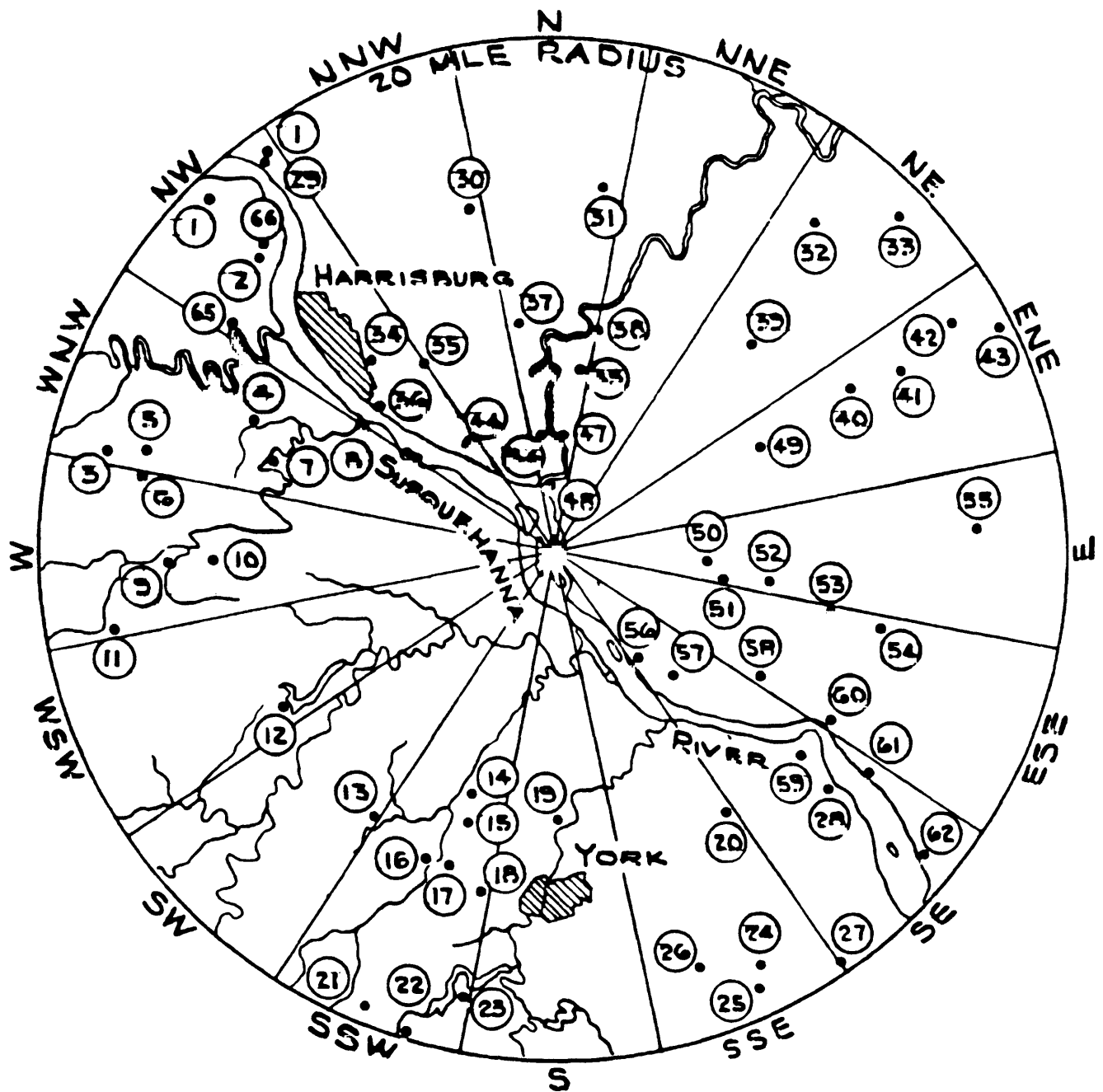
4/00

RESERVOIRS AND LAKES WITHIN A 50 MILE
RADIUS

DSAR-2.FIG-24 .REV. 2

CAD FILE: 6674R15.DWG

FIG 2.6-1



0 3 6
SCALE IN MILES

REFER TO TABLE 2.6-2 FOR IDENTIFICATION OF WATER SUPPLY

NOTE:

1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

DSAR-2.FIG-25 REV. 2

AmerGen

TMI - UNIT 1

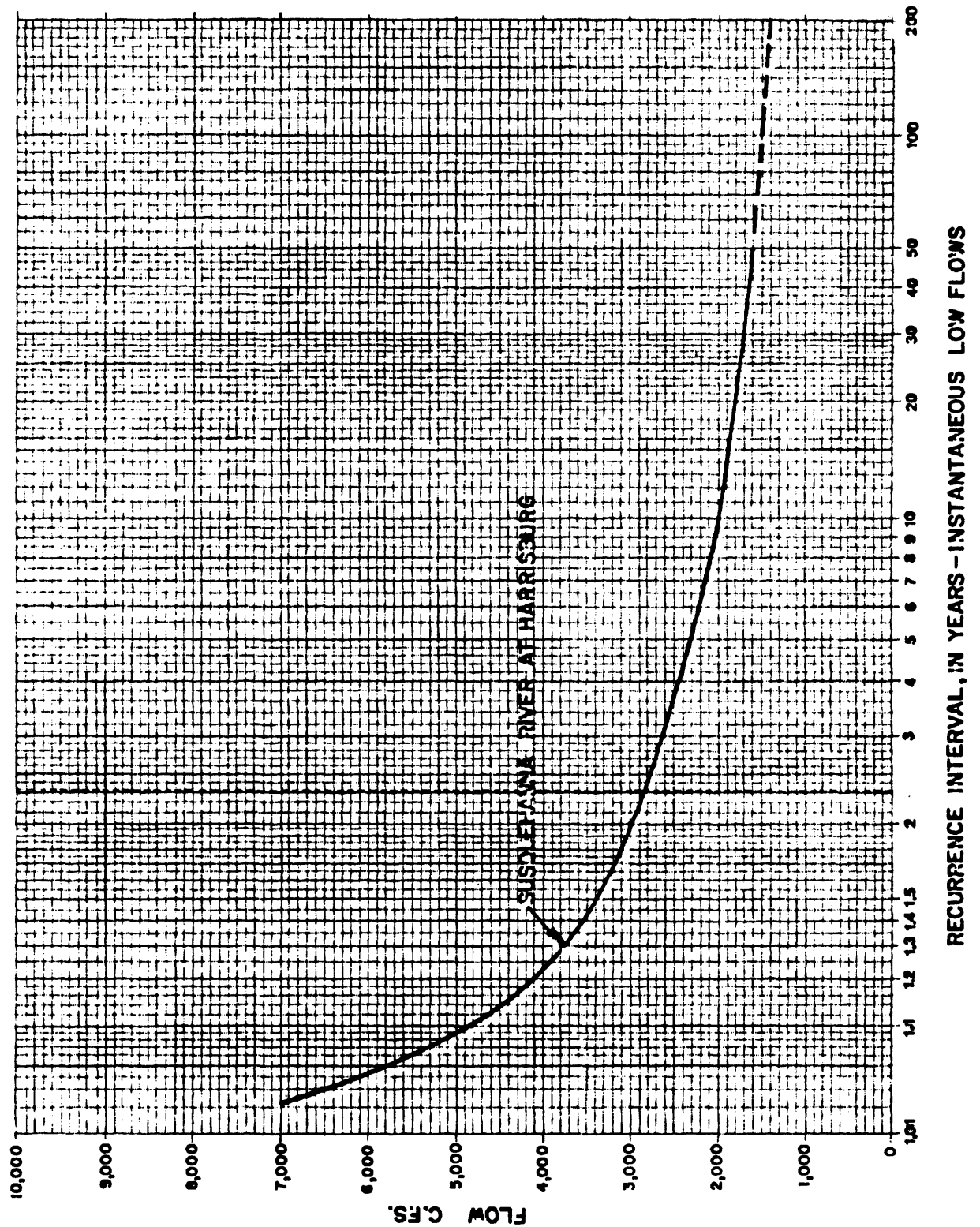
UPDATE - 15

4/00

OTHER WATER SUPPLIES WITHIN A 20 MILE RADIUS

CAD FILE: 6675R15.DWG

FIG 2.6-2



NOTE:

1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

AmerGen

TMI - UNIT 1

UPDATE - 15

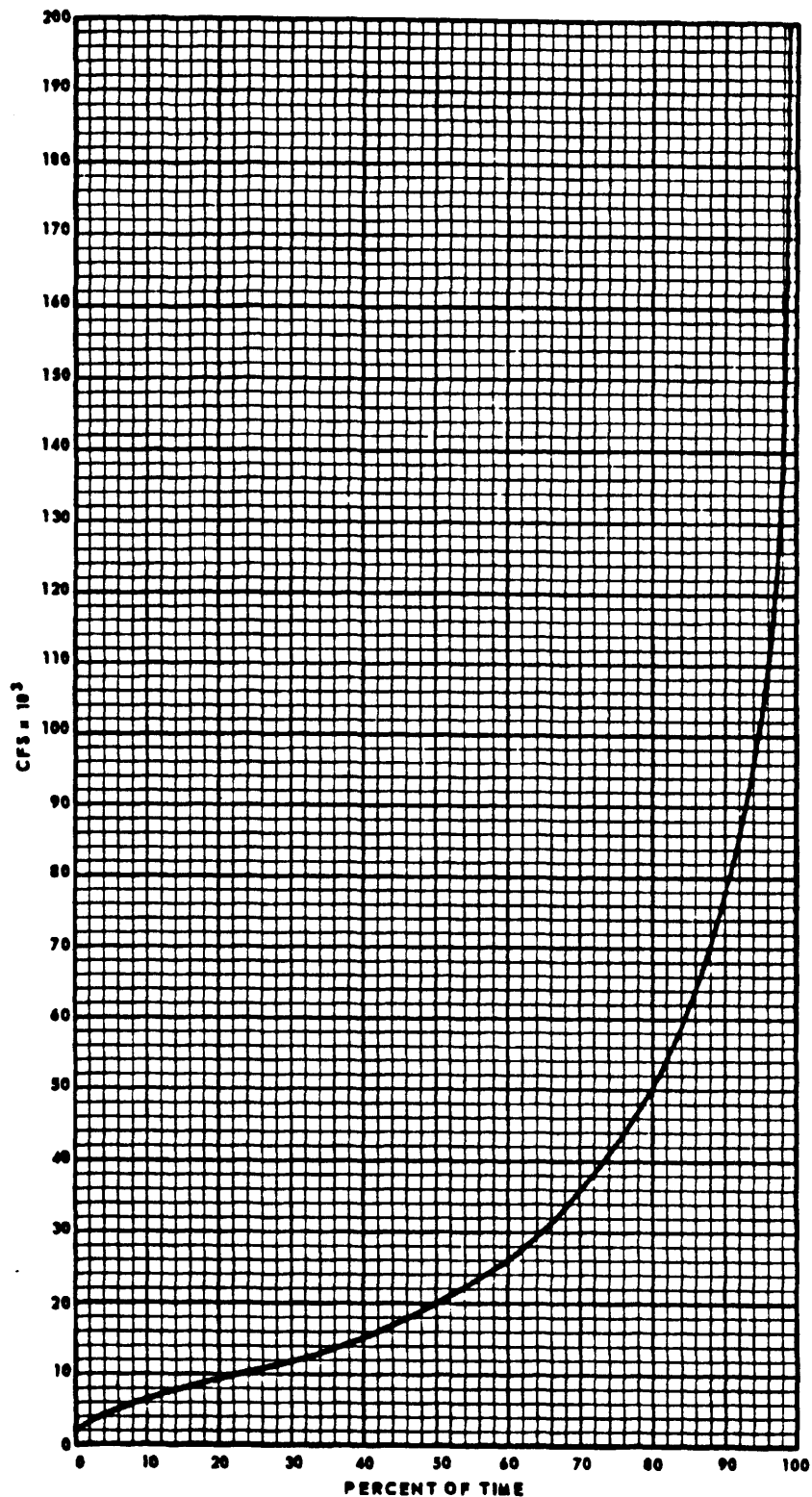
4/00

INSTANTANEOUS LOW FLOW FREQUENCY
CURVE OF SUSQUEHANNA RIVER AT
HARRISBURG

CAD FILE: 6676R15.DWG

FIG 2.6-3

DSAR-2.FIG-26 REV.2



NOTE:

1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

DSAR-2.FIG-27 REV. 2

AmerGen

TMI - UNIT 1

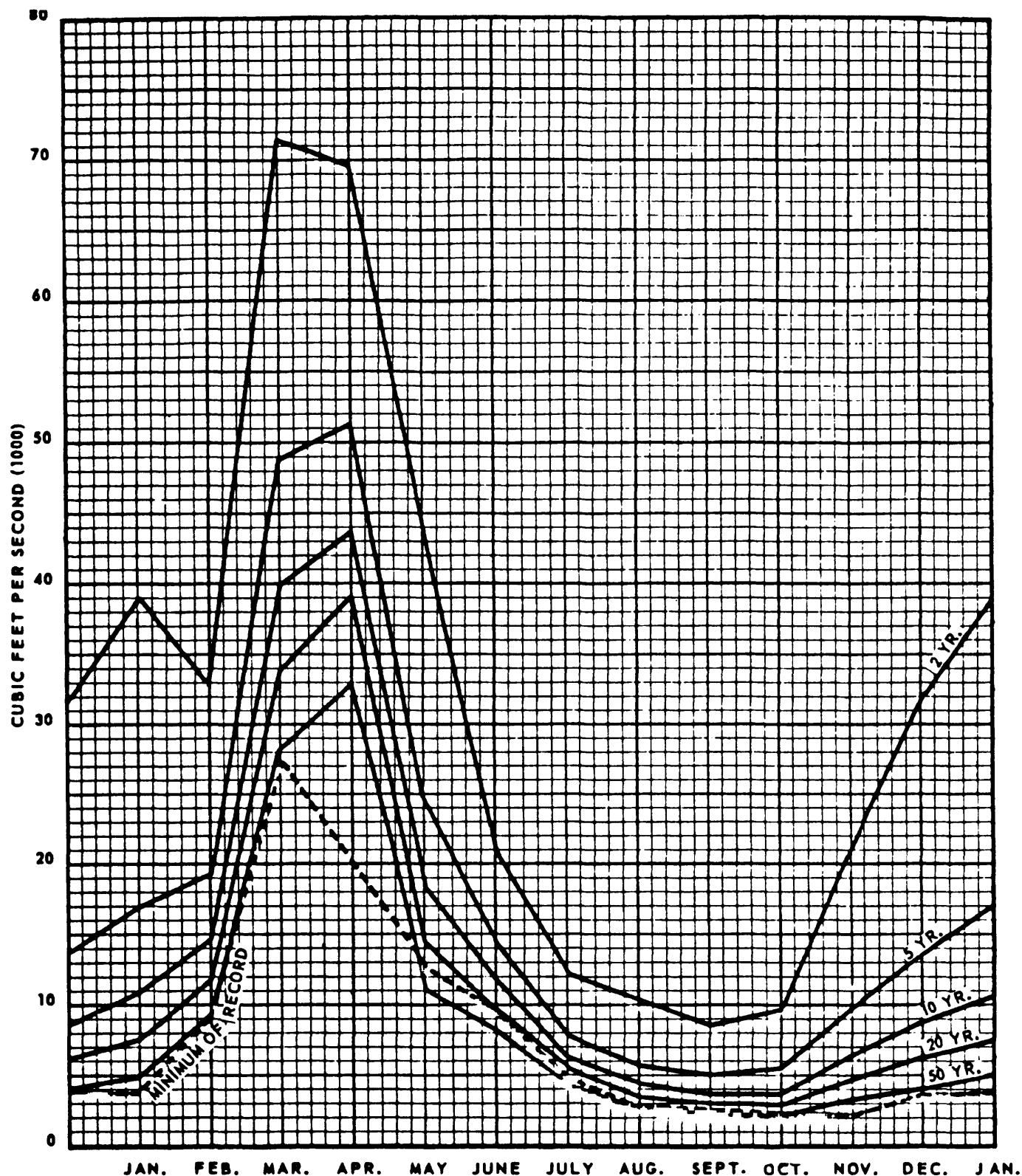
UPDATE - 15

4/00

SUSQUEHANNA RIVER AT HARRISBURG, PA,
DISCHARGE DURATION CURVE

CAD FILE: 6677R15.DWG

FIG 2.6-4



NOTE:

1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

DSAR-2.FIG-28 REV. 2

AmerGen

TMI - UNIT 1

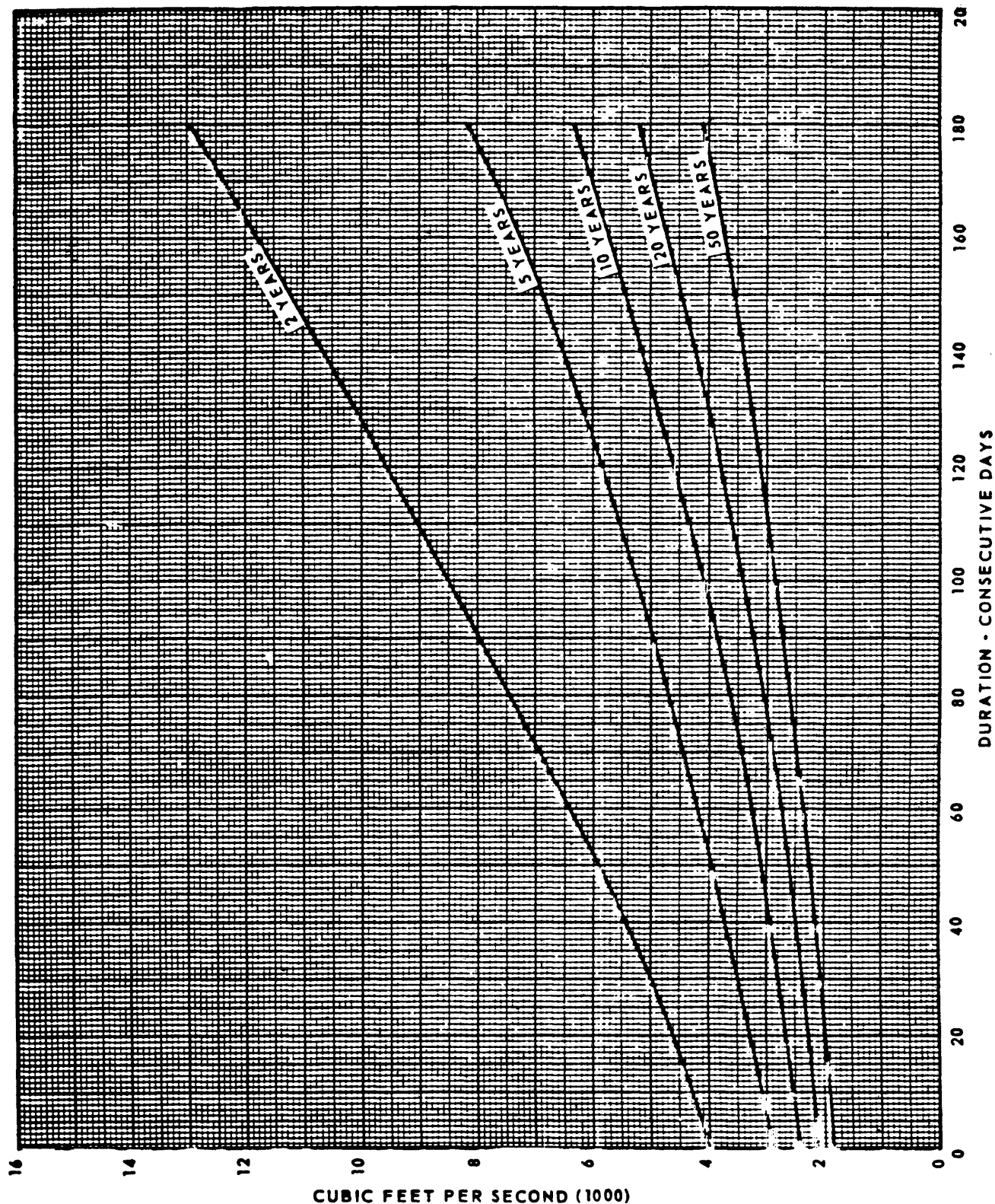
UPDATE - 15

4/00

SUSQUEHANNA RIVER AT HARRISBURG, PA,
MEAN MONTHLY FLOW SUMMARY

CAD FILE: 6678R15.DWG

FIG 2.6-5



NOTE:

1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

DSAR-2.FIG-29 REV. 2

AmerGen

TMI - UNIT 1

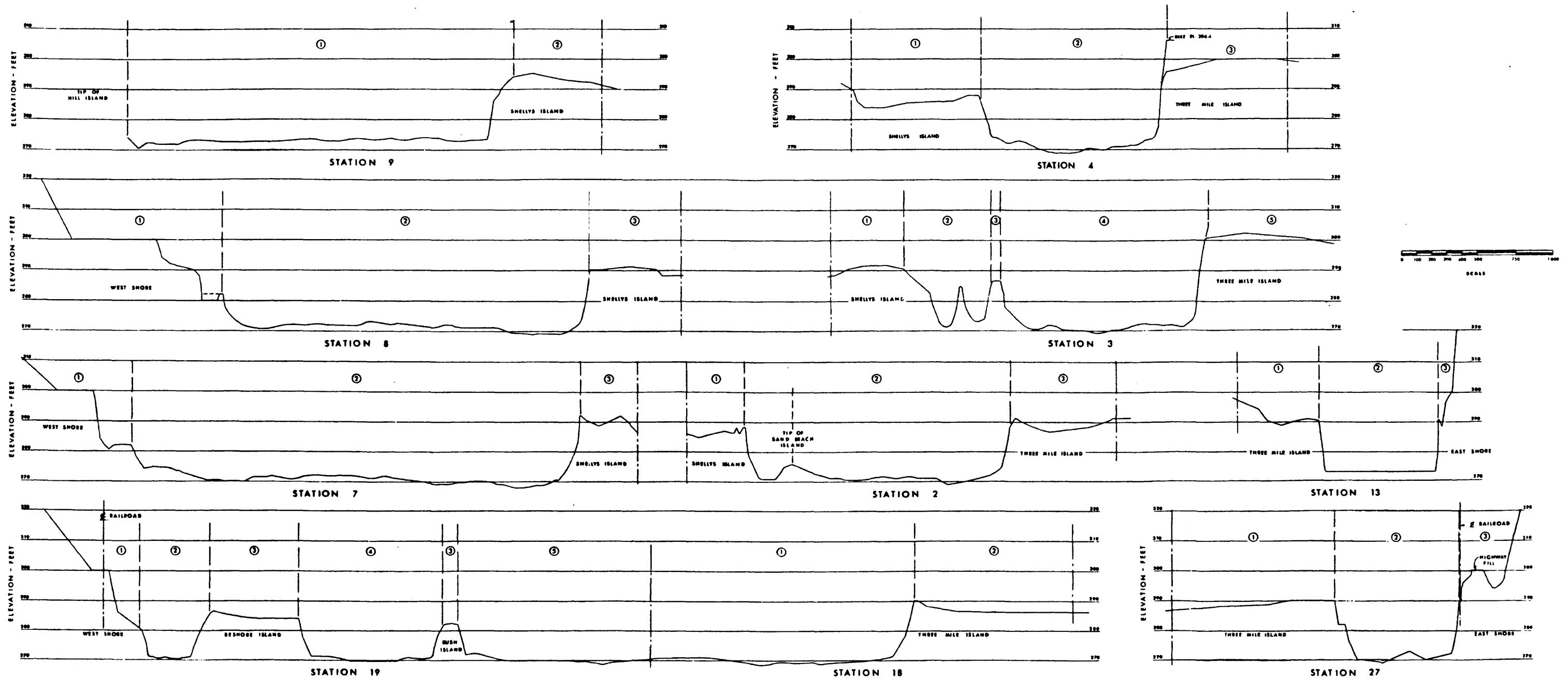
UPDATE - 15

4/00

"SUSQUEHANNA RIVER AT HARRISBURG, PA,
MINIMUM DAILY FLOW DURATION SUMMARY"

CAD FILE: 6679R15.DWG

FIG 2.6-6



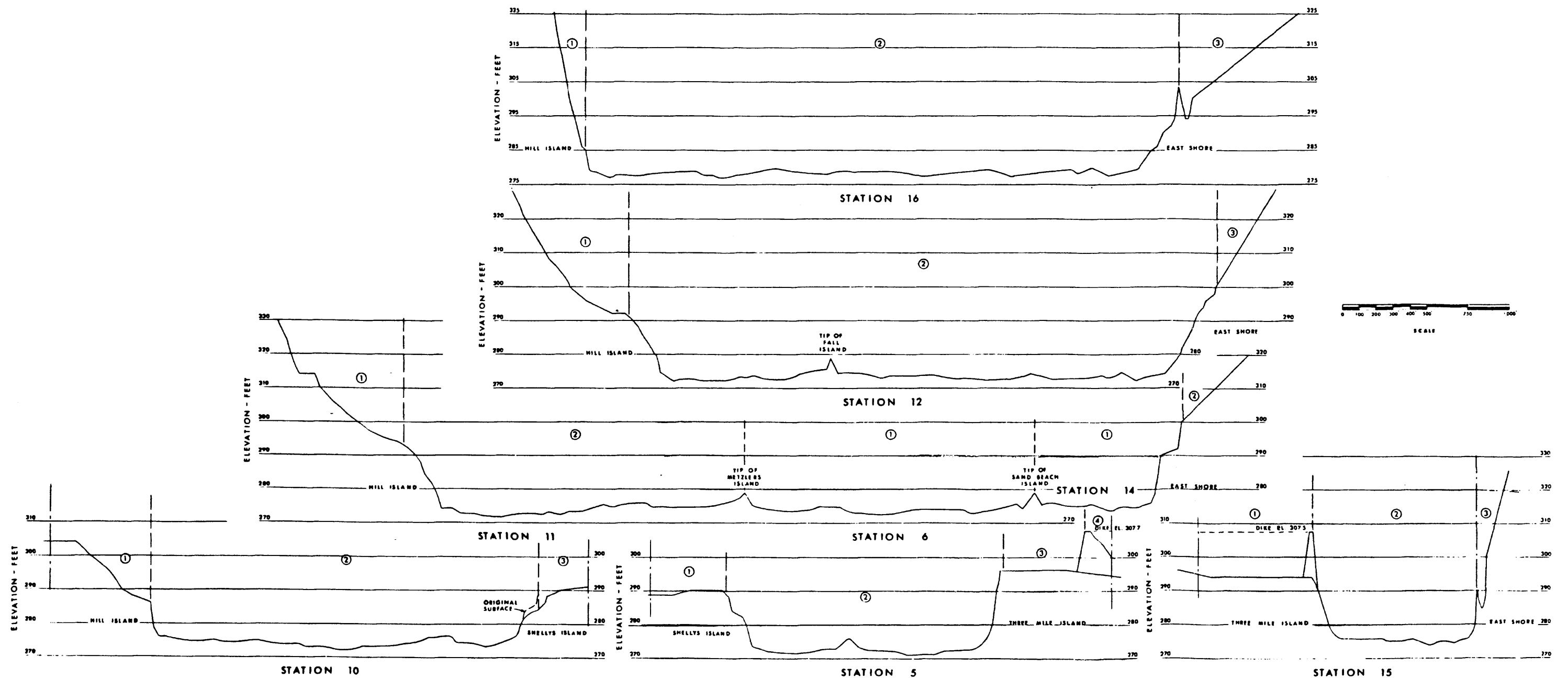
NOTE:

1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

**TMI-1
Updated Final Safety Analysis Report
RIVER CROSS SECTIONS USED IN LOW
FLOW STUDIES**

Figure 2.6-6A

Rev. 22, 04/14

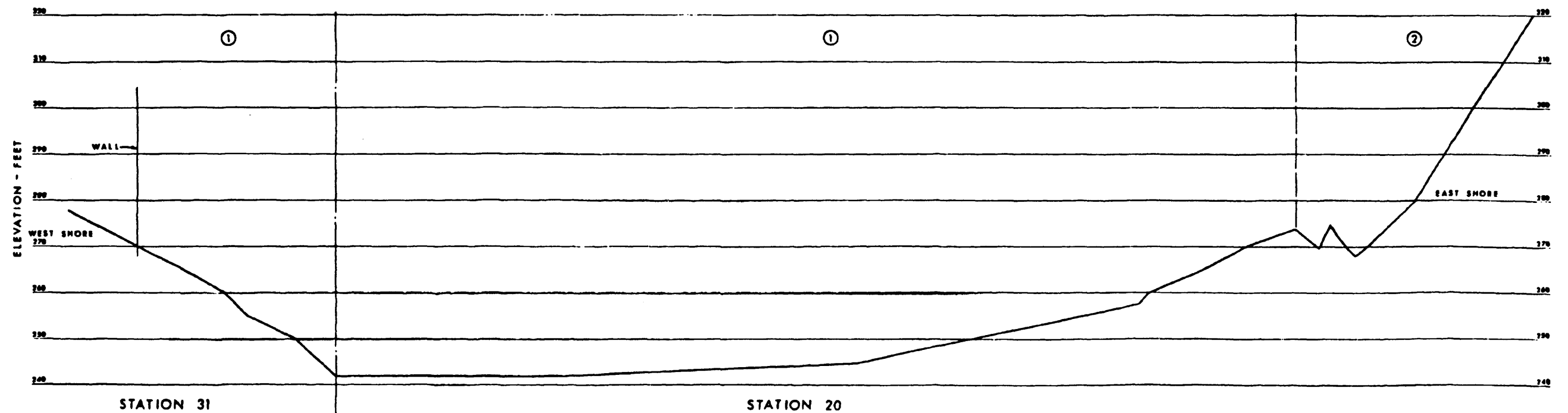
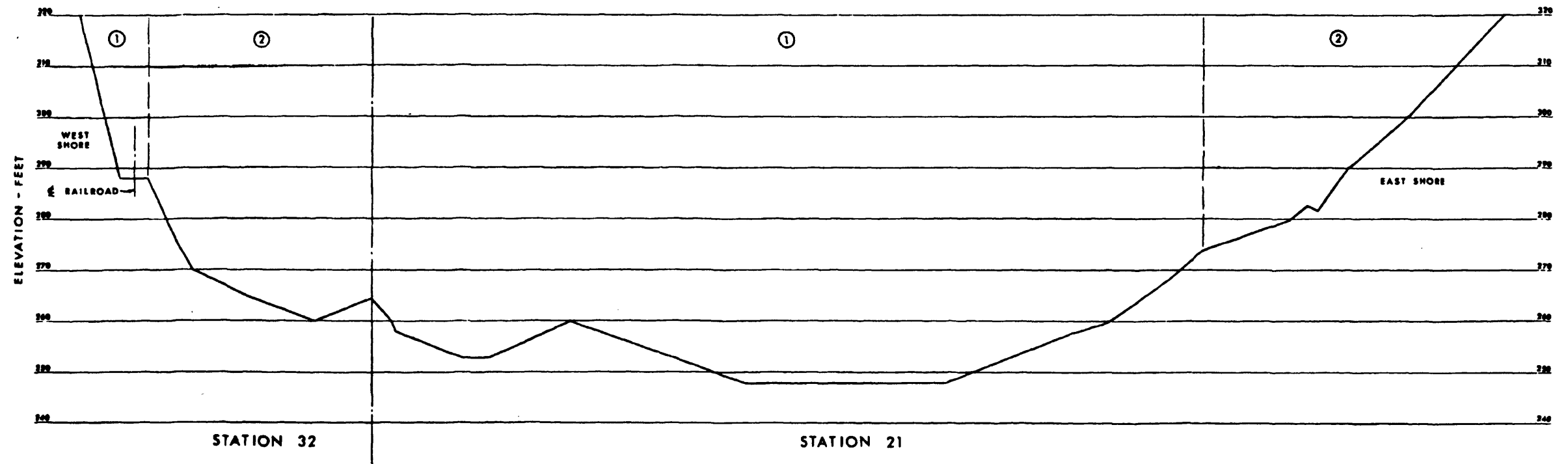


NOTE:
1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

TMI-1
Updated Final Safety Analysis Report
RIVER CROSS SECTIONS USED IN LOW
FLOW STUDIES

Figure 2.6-6B

Rev. 22, 04/14

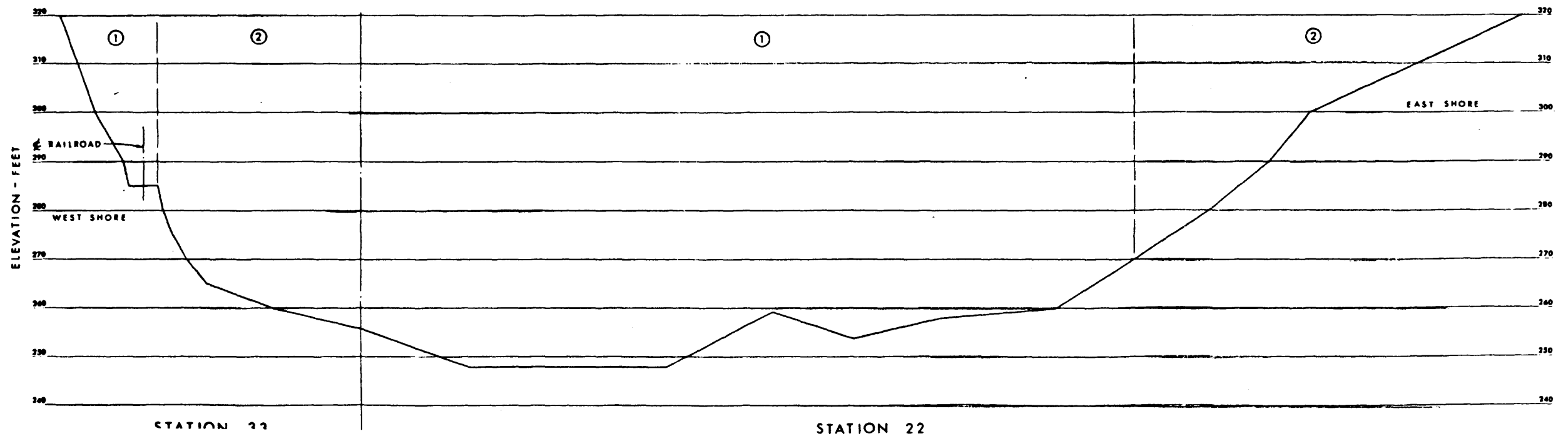
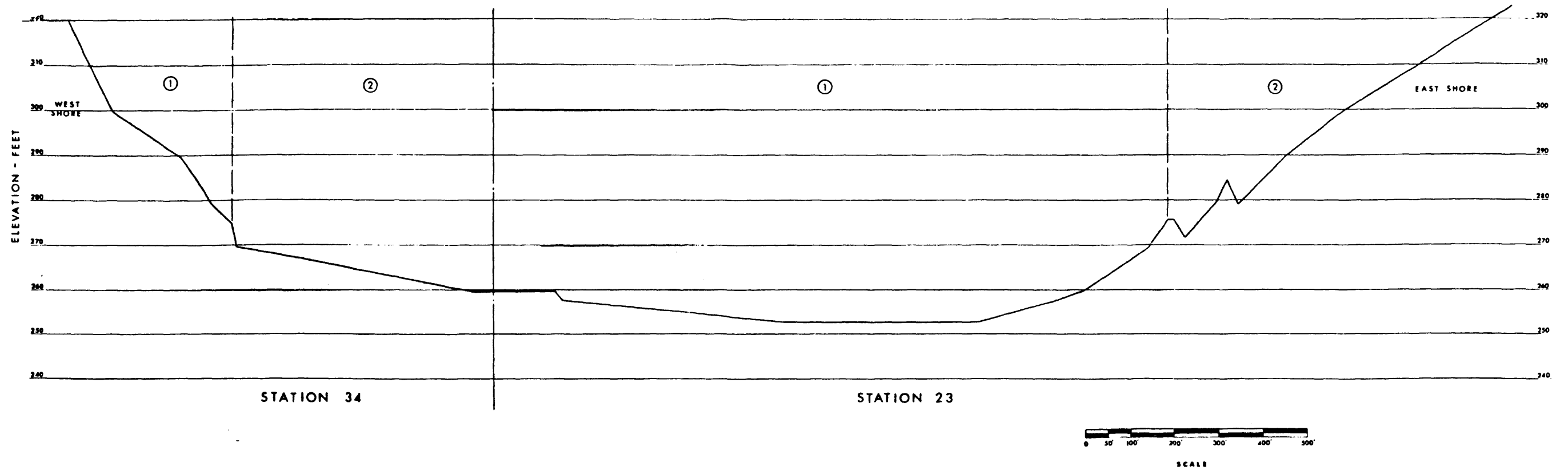


NOTE:
1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

TMI-1
Updated Final Safety Analysis Report
RIVER CROSS SECTIONS USED IN LOW FLOW STUDIES

Figure 2.6-6C

Rev. 22, 04/14

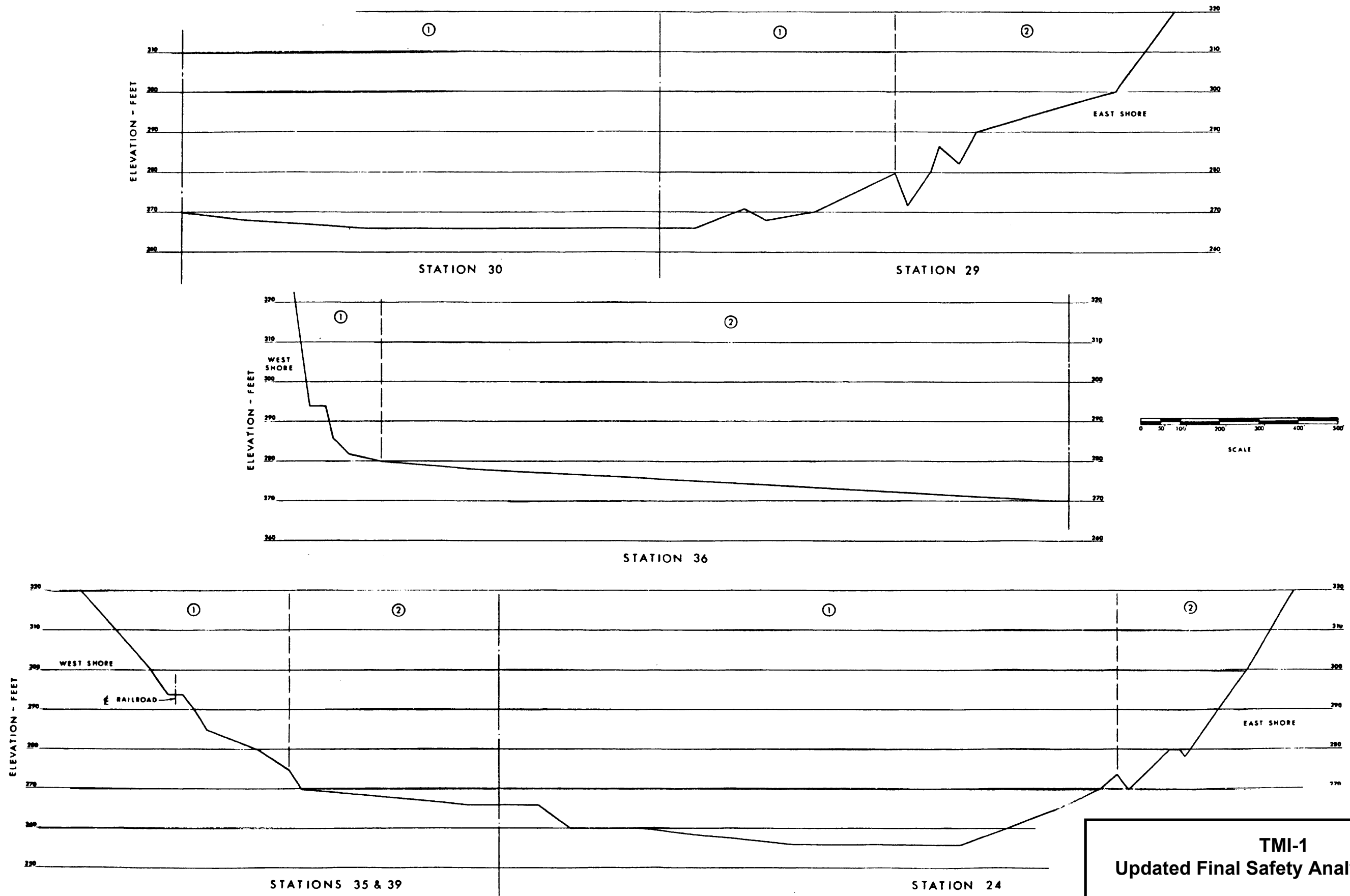


NOTE:
 1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

DSAR-2.FIG-33 REV. 2

TMI-1
Updated Final Safety Analysis Report
RIVER CROSS SECTIONS USED IN LOW
FLOW STUDIES

Figure 2.6-6D Rev. 22, 04/14



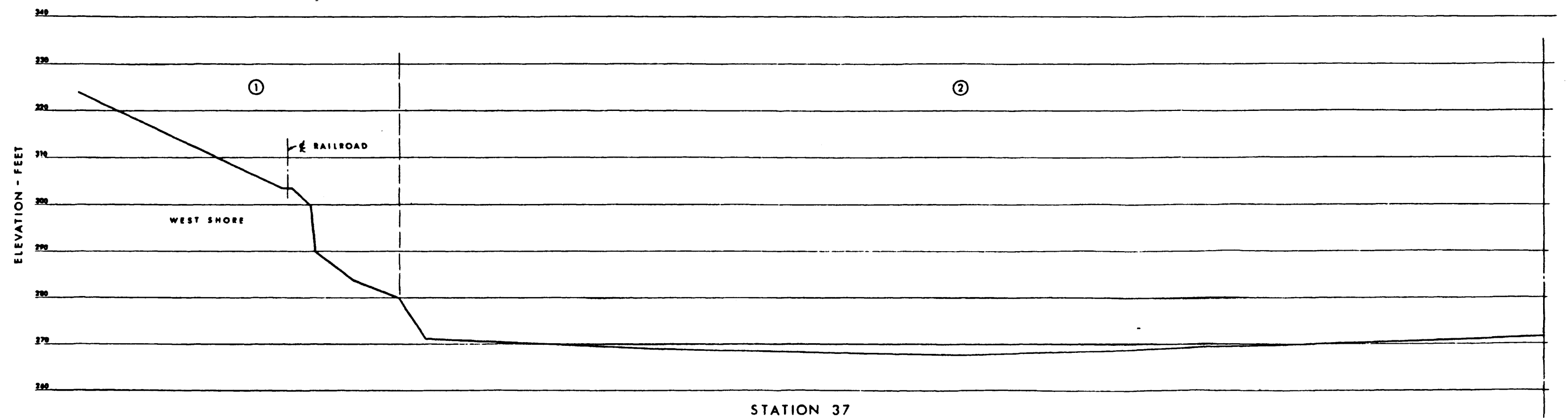
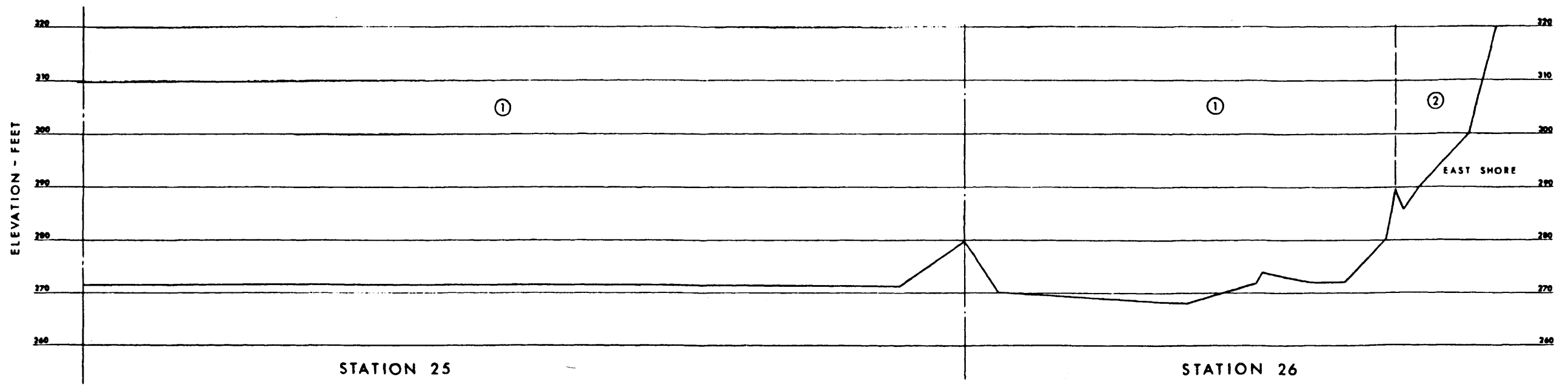
NOTE:
 1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

TMI-1
Updated Final Safety Analysis Report

RIVER CROSS SECTIONS USED IN LOW FLOW STUDIES

Figure 2.6-6E

Rev. 22, 04/14



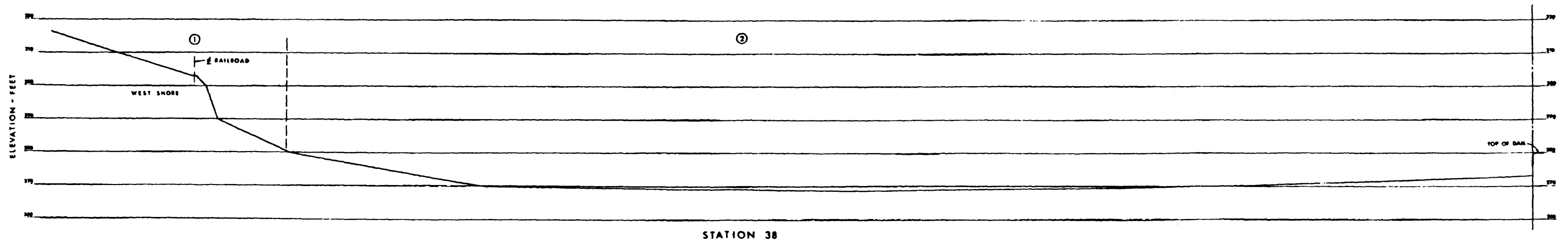
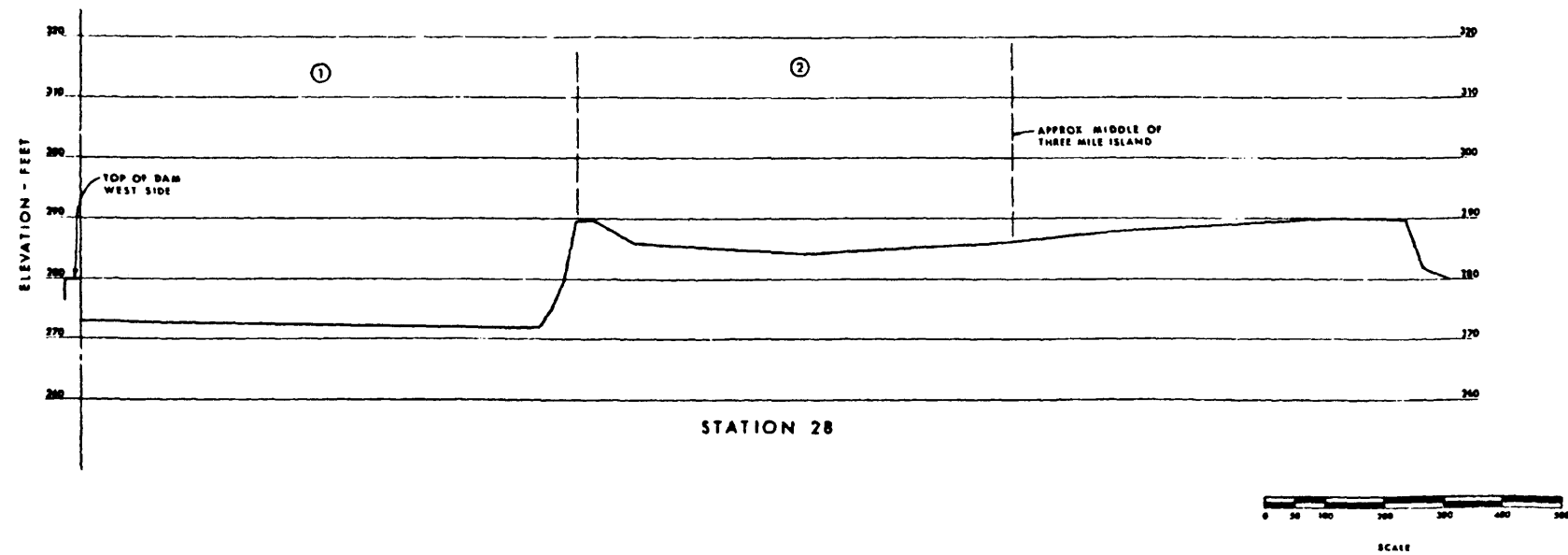
NOTE:
 1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

TMI-1
Updated Final Safety Analysis Report

RIVER CROSS SECTIONS USED IN LOW FLOW STUDIES

Figure 2.6-6F

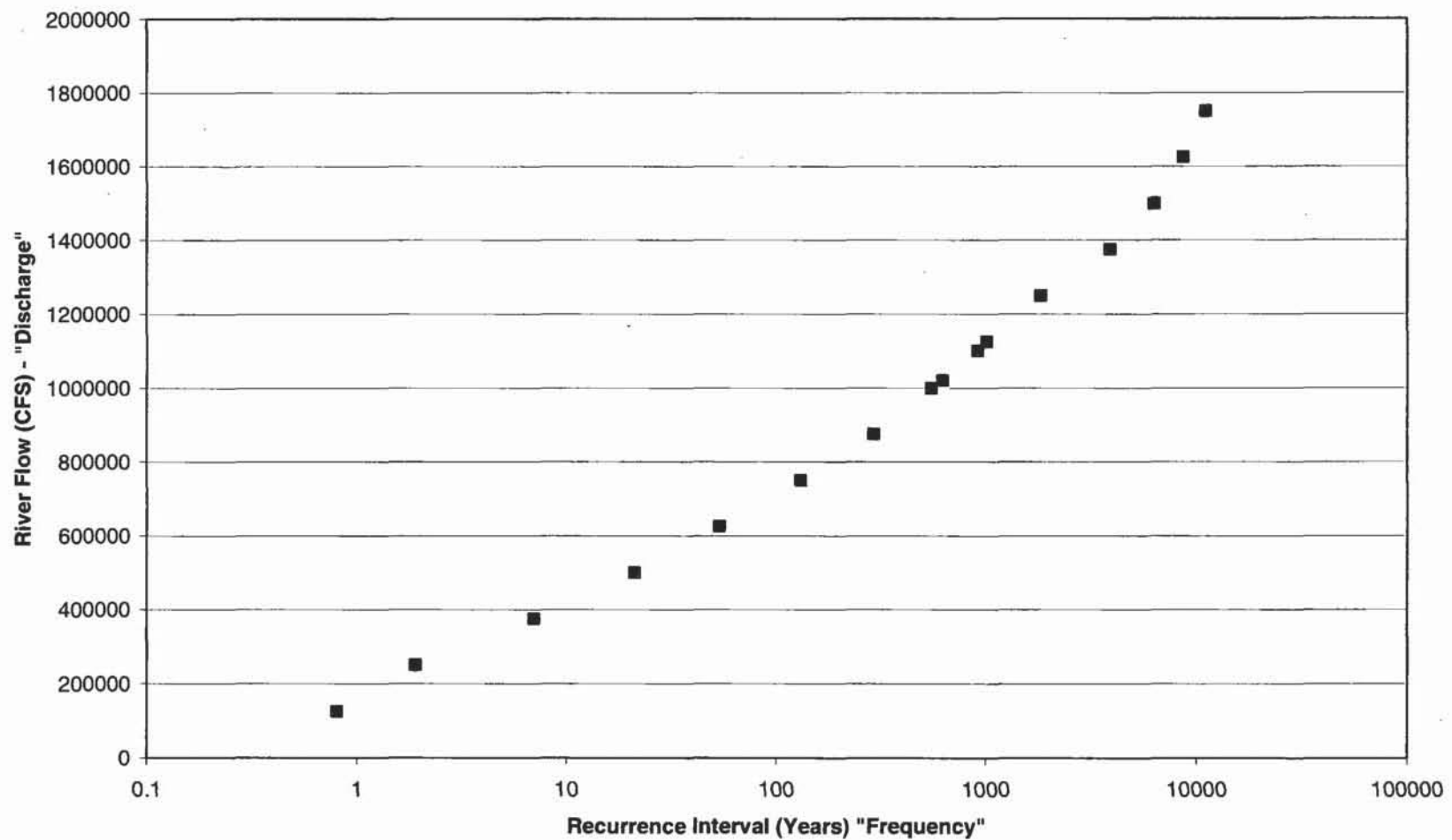
Rev. 22, 04/14



NOTE:
1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

TMI-1
Updated Final Safety Analysis Report
RIVER CROSS SECTIONS USED IN LOW FLOW STUDIES

River Discharge Frequency



DSAR-2.FIG-37 REV. 2

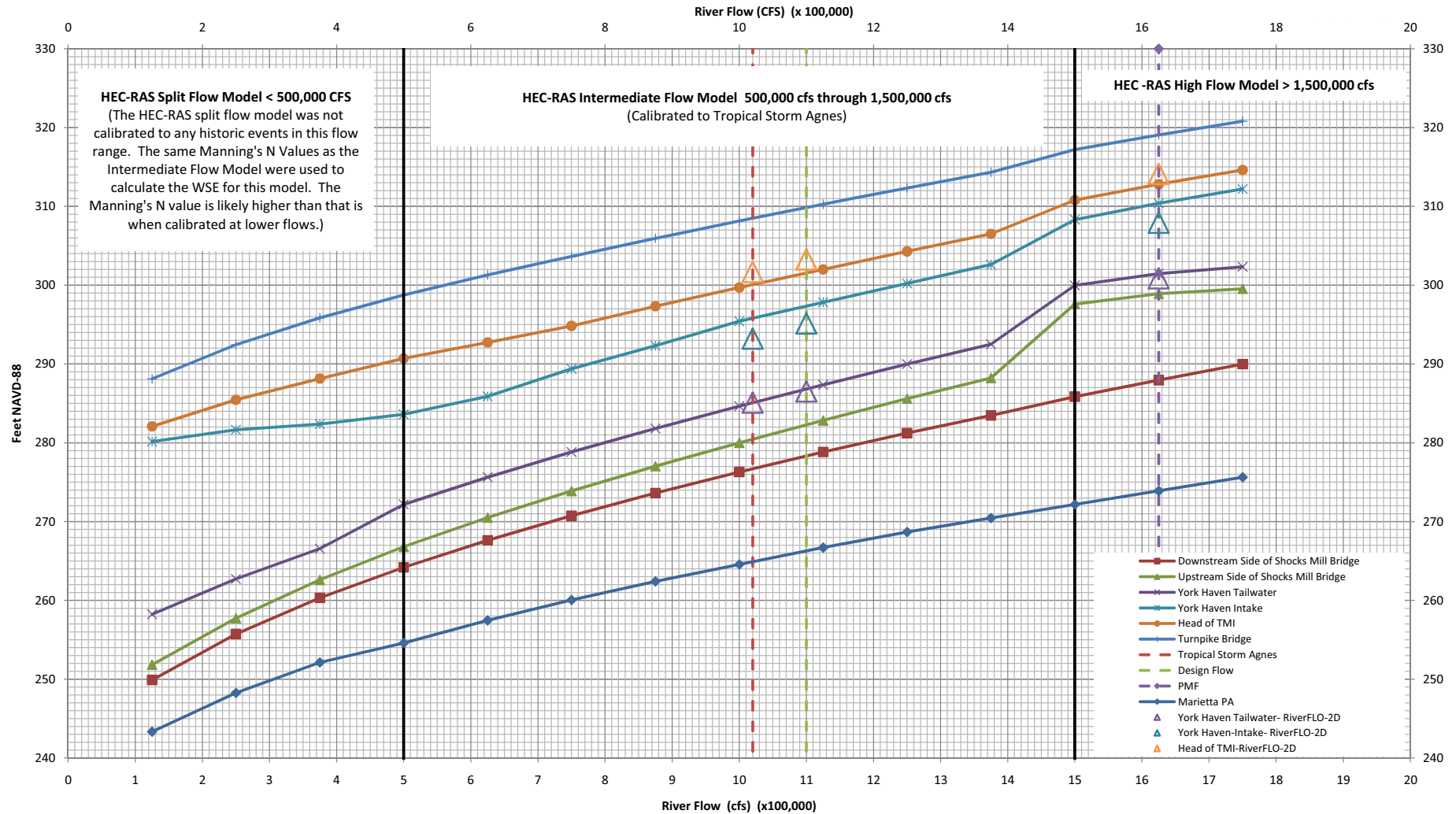
TMI-1
Updated Final Safety Analysis Report

RIVER DISCHARGE FREQUENCY

Figure 2.6-7

Rev. 22, 04/14

Susquehanna River Stage - Discharge Curves PA Turnpike Bridge to Marietta PA



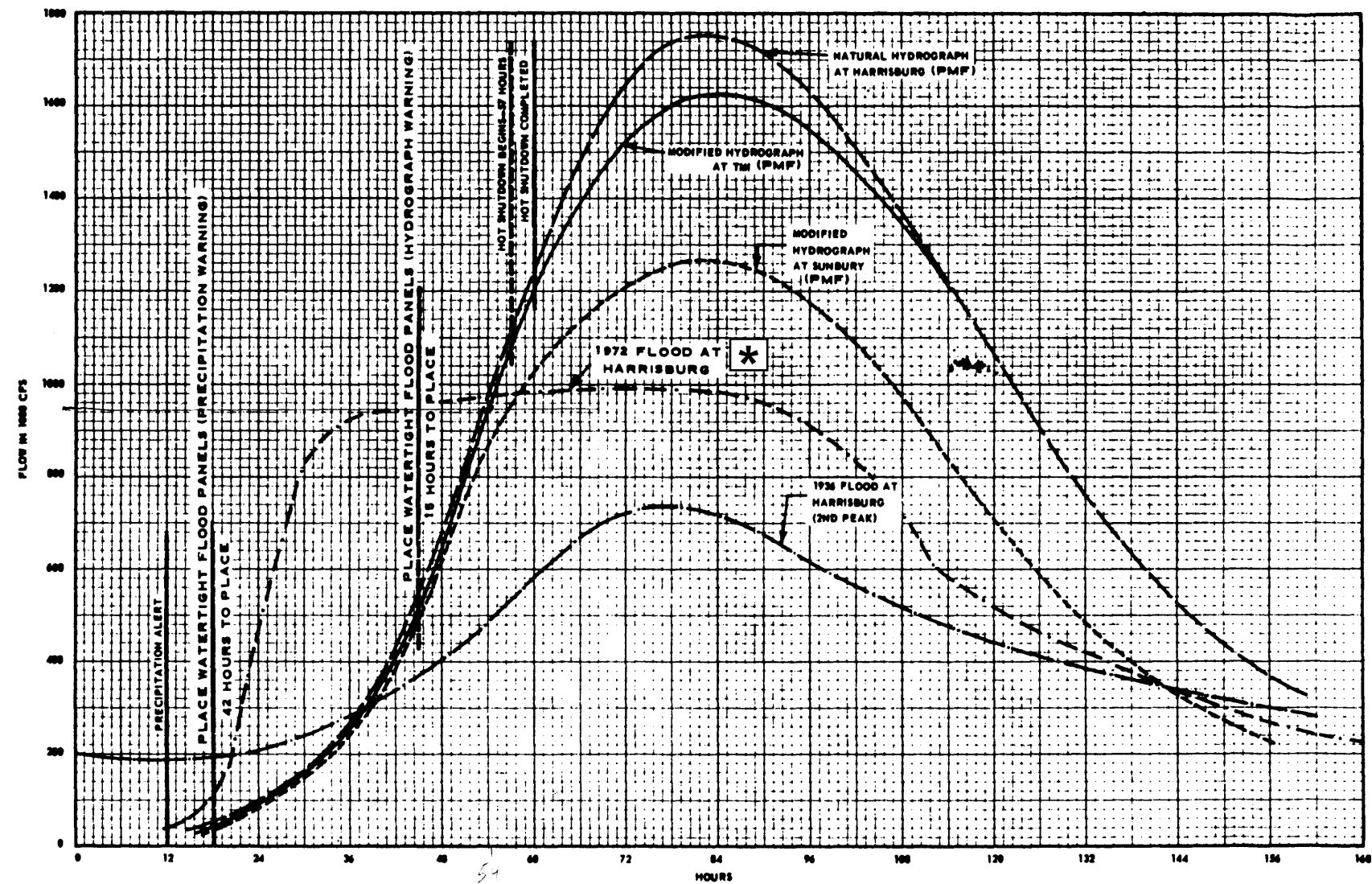
DSAR-2.FIG-38 REV. 2

TMI-1
Updated Final Safety Analysis Report

STAGE DISCHARGE CURVES PA
TURNPIKE BRIDGE TO MARIETTA

Figure 2.6-8

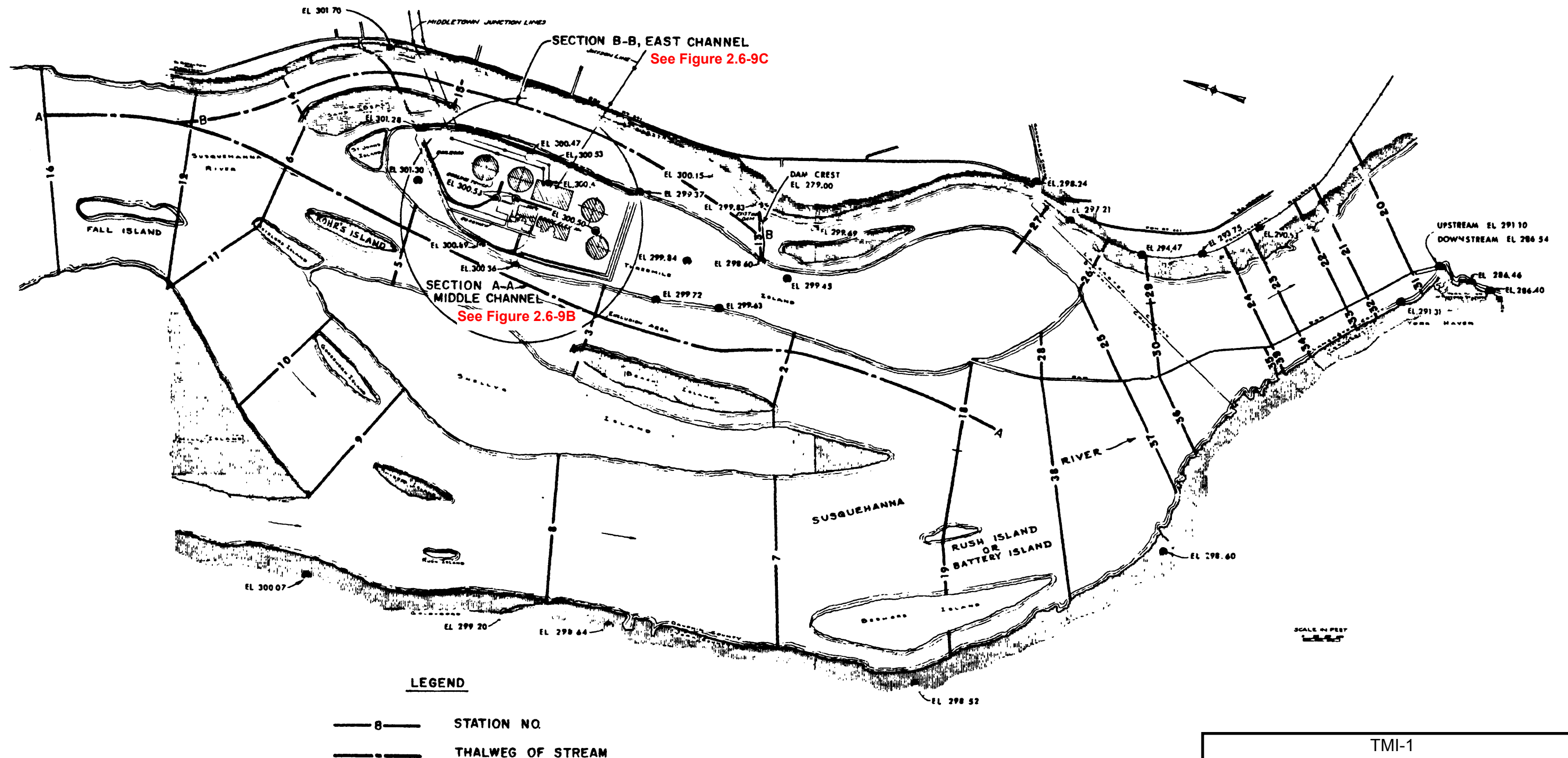
Rev. 22, 04/14

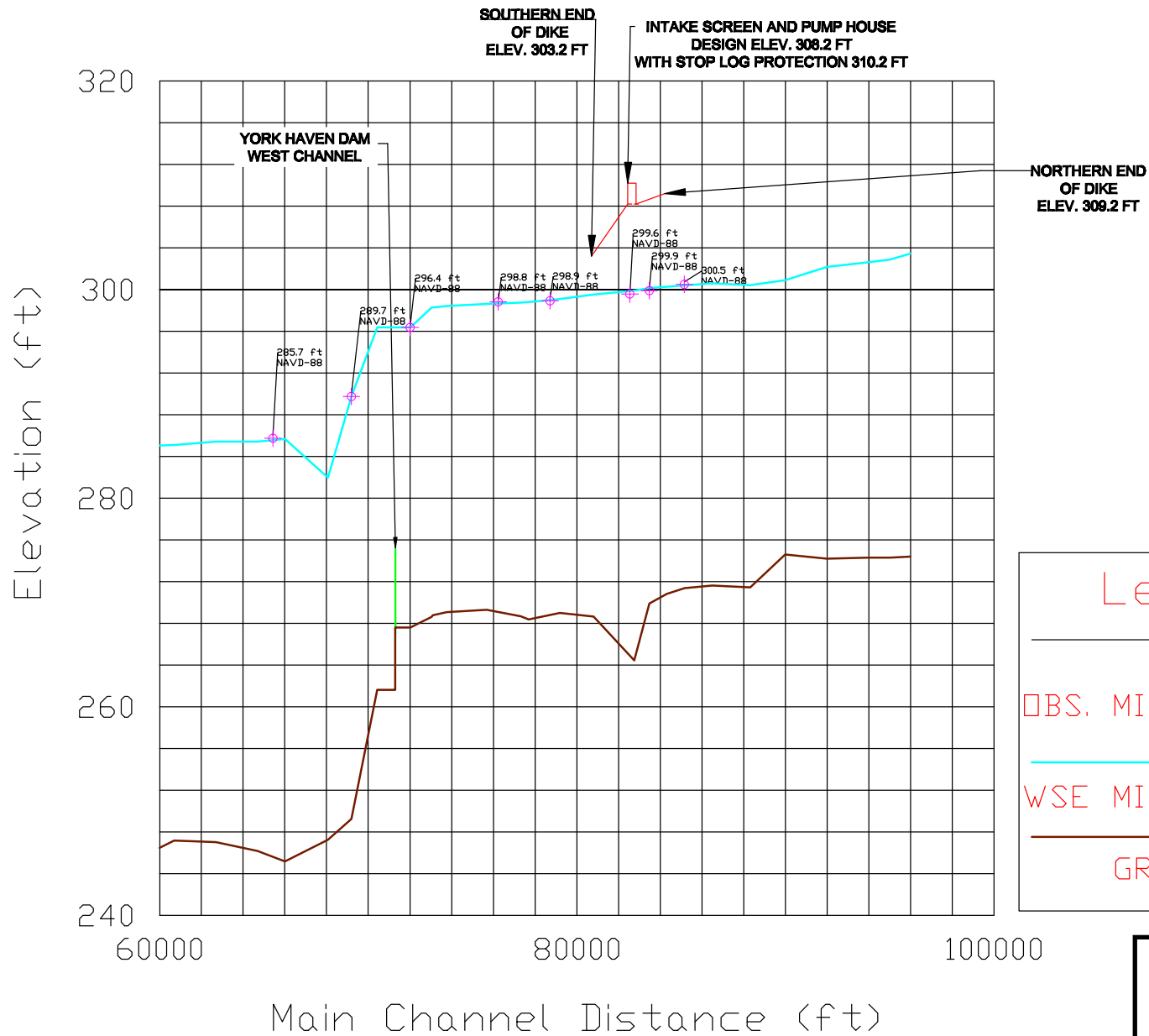


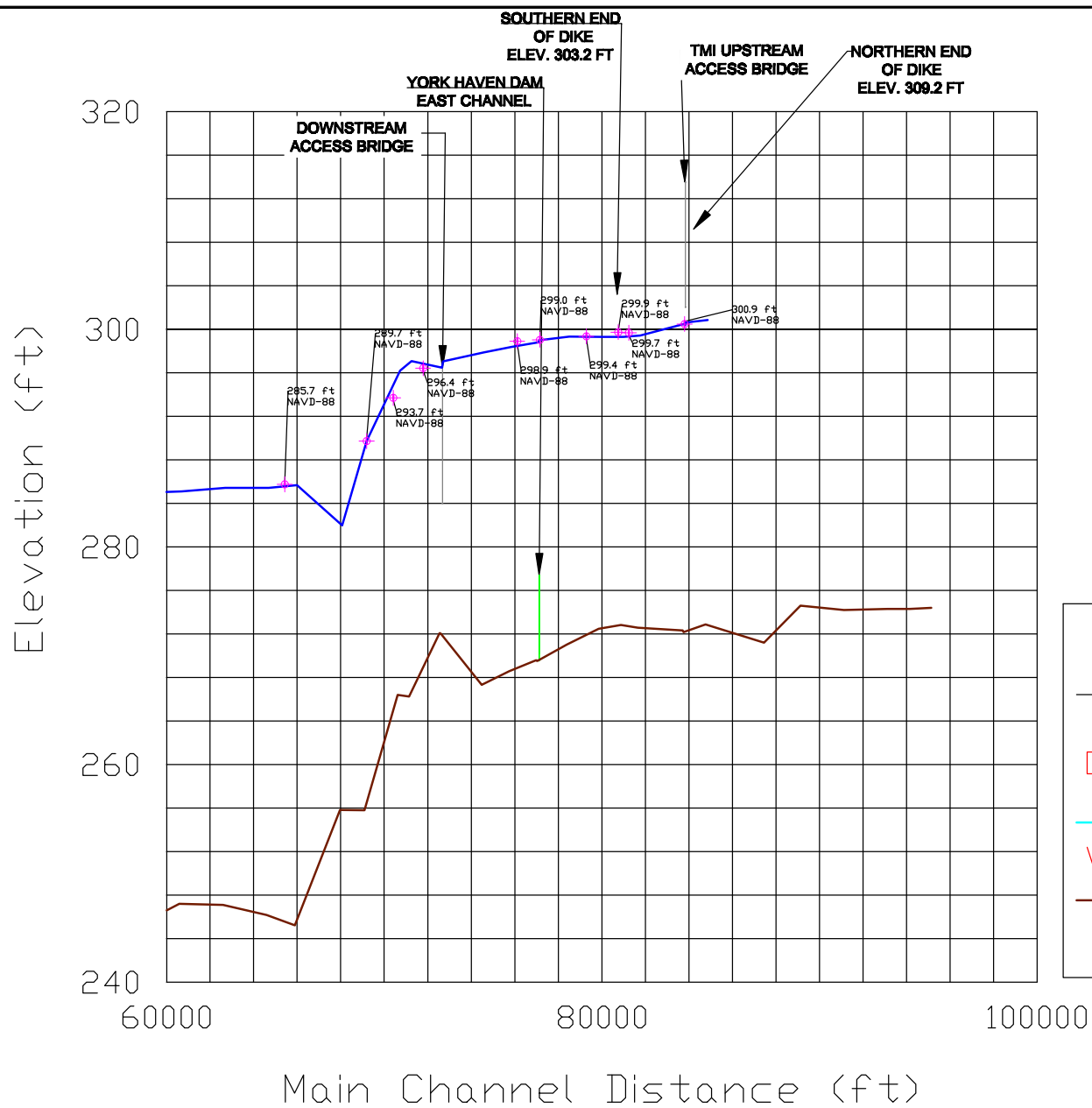
* NOTE: The curve titled "1972 FLOOD AT HARRISBURG" has no known basis. Hydrographs for that flood show a curve similar in shape / duration to the other curves (see Figure 2.6-9a).

TMI - 1
Updated Final Safety Analysis Report

Flood Hydrographs







DSAR-2.FIG-42 REV. 2

TMI-1
Updated Final Safety Analysis Report

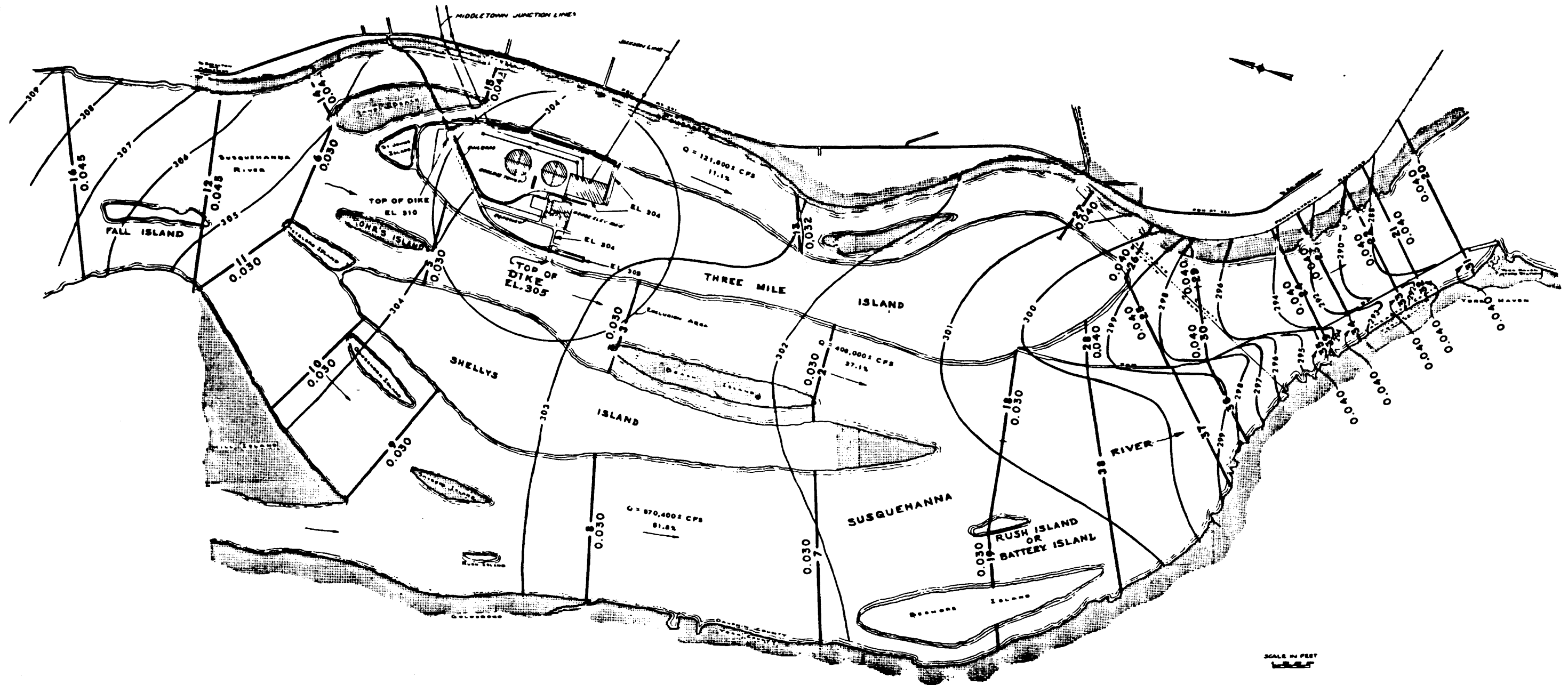
WATER SURFACE CALIBRATION
HEC RAS MODEL - EAST CHANNEL
TROPICAL STORM AGNES 1020000 CFS

Figure 2.6-10

Deleted

Figure 2.6-11

Deleted



LEGEND

— 309 —	WATER SURFACE CONTOURS
— 8 — 0.030	STATION NO. AND MANNINGS ROUGHNESS COEFFICIENT

NOTE:

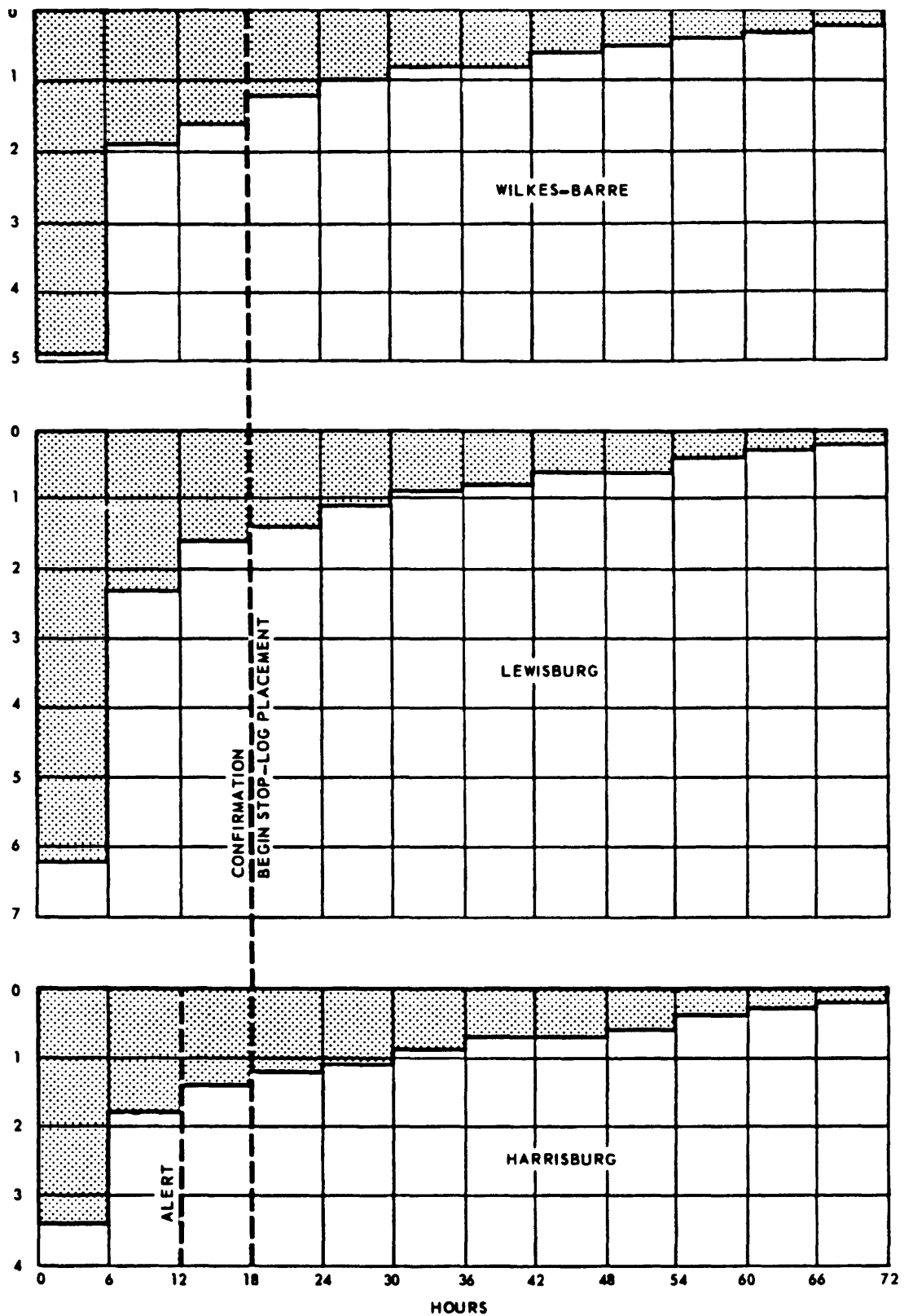
These water surface elevations and velocities for a flow for a flow of 1,100,000 cfs were the basis for the design of the site inundation projection. The predicted conditions for a flow of 1,100,000 cfs have been superseded by those in Reference 38.

TMI-1
Updated Final Safety Analysis Report

DESIGN FLOOD LEVELS

Figure 2.6-12

Rev. 22, 04/14

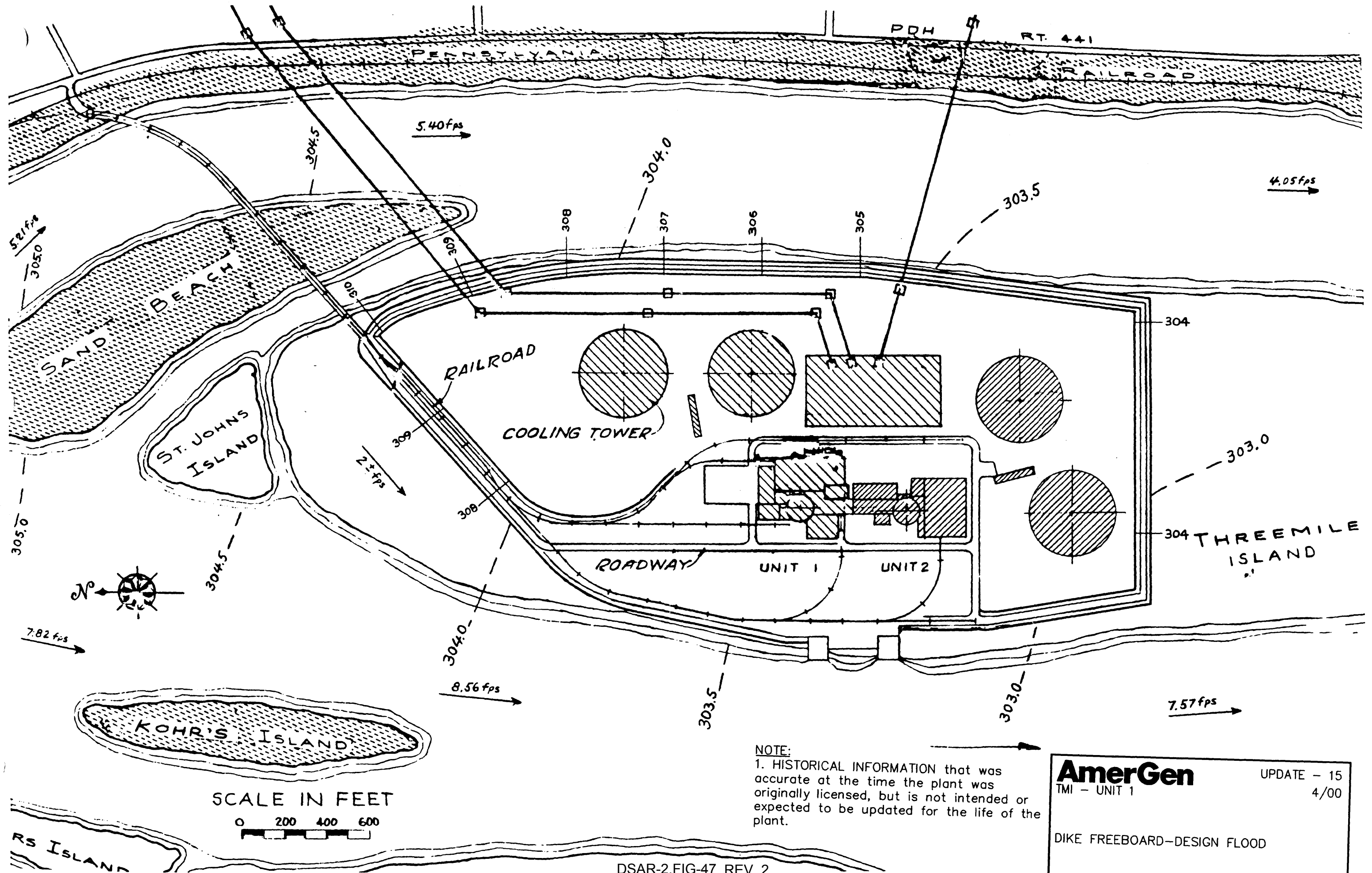


TMI-1
Updated Final Safety Analysis Report
PRECIPITATIONS FROM HYDROMET NO. 40

DSAR-2.FIG-46 REV. 2

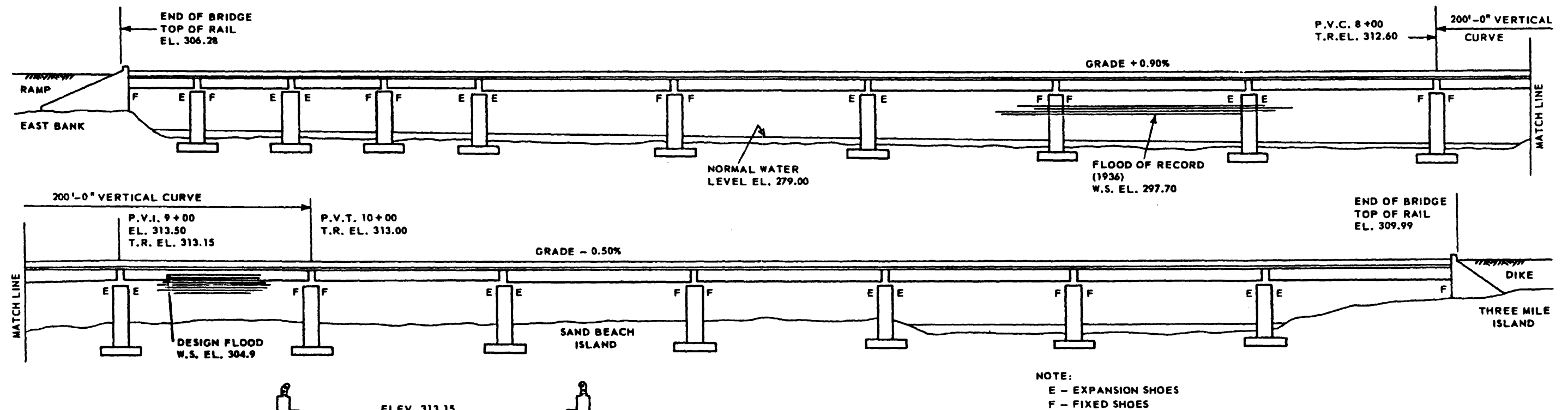
Figure 2.6-14

Rev. 22, 04/14

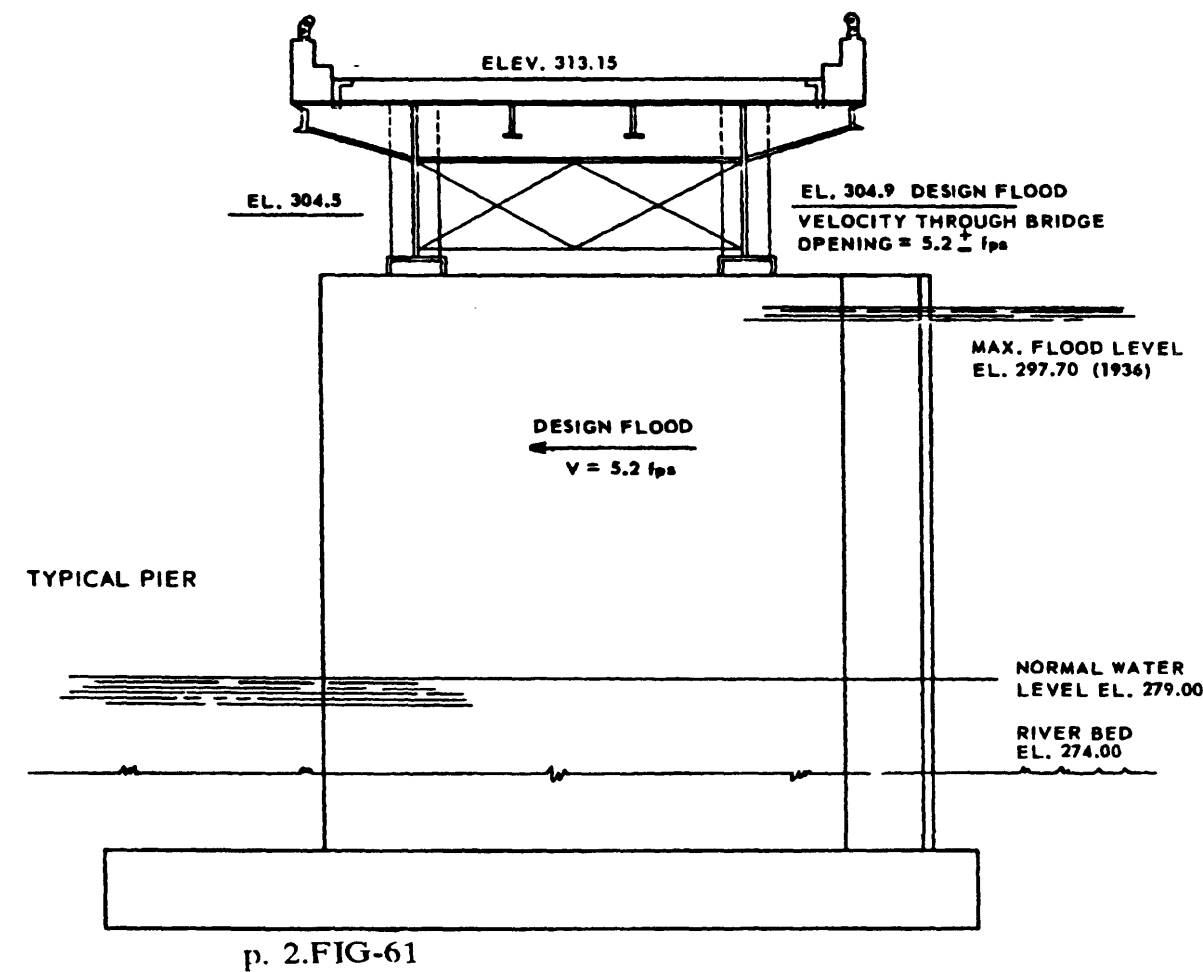


NOTE:
 1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

AmerGen TMI - UNIT 1 DIKE FREEBOARD-DESIGN FLOOD CAD FILE: 6698R15.DWG	UPDATE - 15 4/00
	FIG 2.6-15



ELEVATION



NOTE:
1. HISTORICAL INFORMATION that was accurate at the time the plant was originally licensed, but is not intended or expected to be updated for the life of the plant.

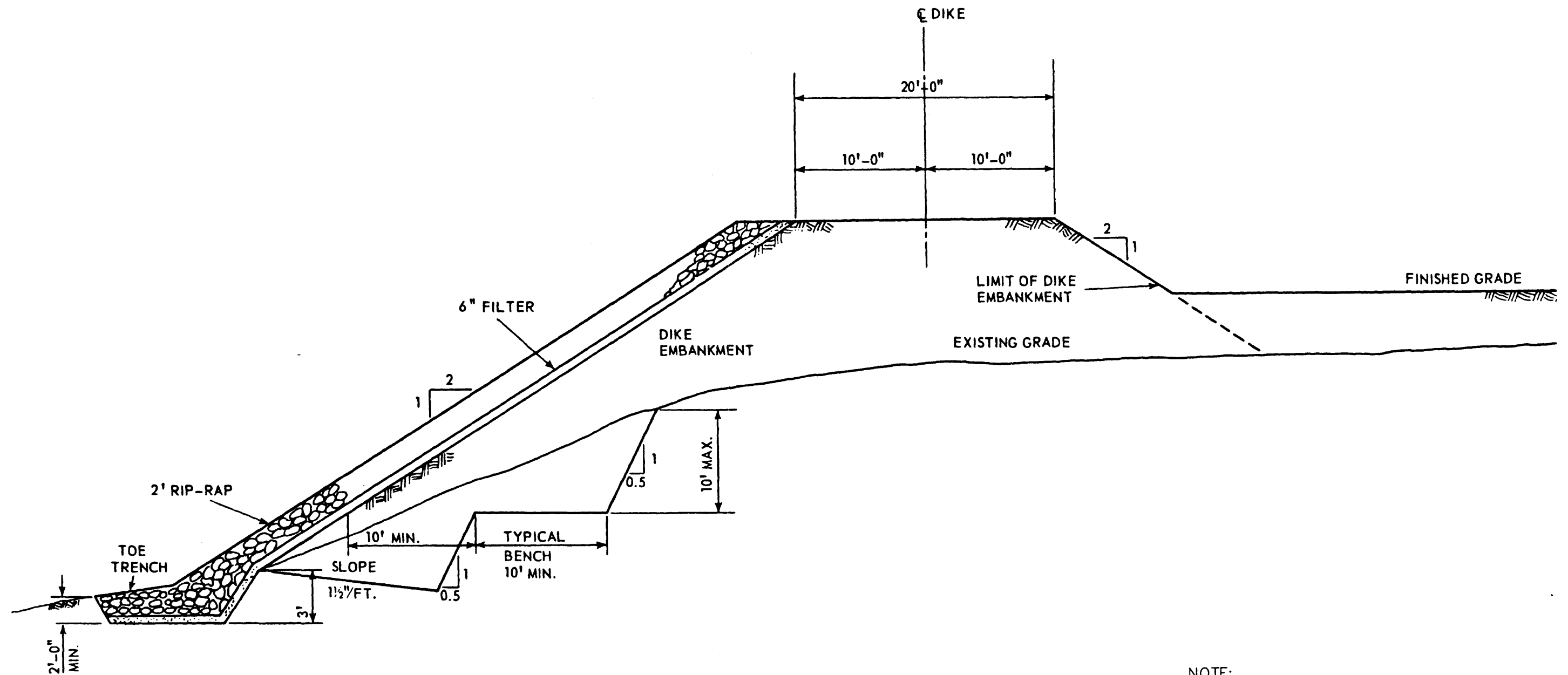
AmerGen
TMI - UNIT 1

UPDATE - 15
4/00

HYDRAULIC CONDITIONS AT NORTH
BRIDGE-DESIGN FLOOD

CAD FILE: 6699R15.DWG

FIG 2.6-16



DIKE RIP-RAP

90% OF PIECES SHALL WEIGH MORE THAN 60#
 50% OF PIECES SHALL WEIGH MORE THAN 100#
 MAXIMUM WEIGHT SHALL BE 180#

NOTE

BENCHING IS REQUIRED WHEN EMBANKMENT
 IS PLACED ON SLOPES STEEPER THAN
 4 (HORZ.) TO 1 (VERT.)

NOTE:

1. HISTORICAL INFORMATION that was
 accurate at the time the plant was
 originally licensed, but is not intended or
 expected to be updated for the life of the
 plant.

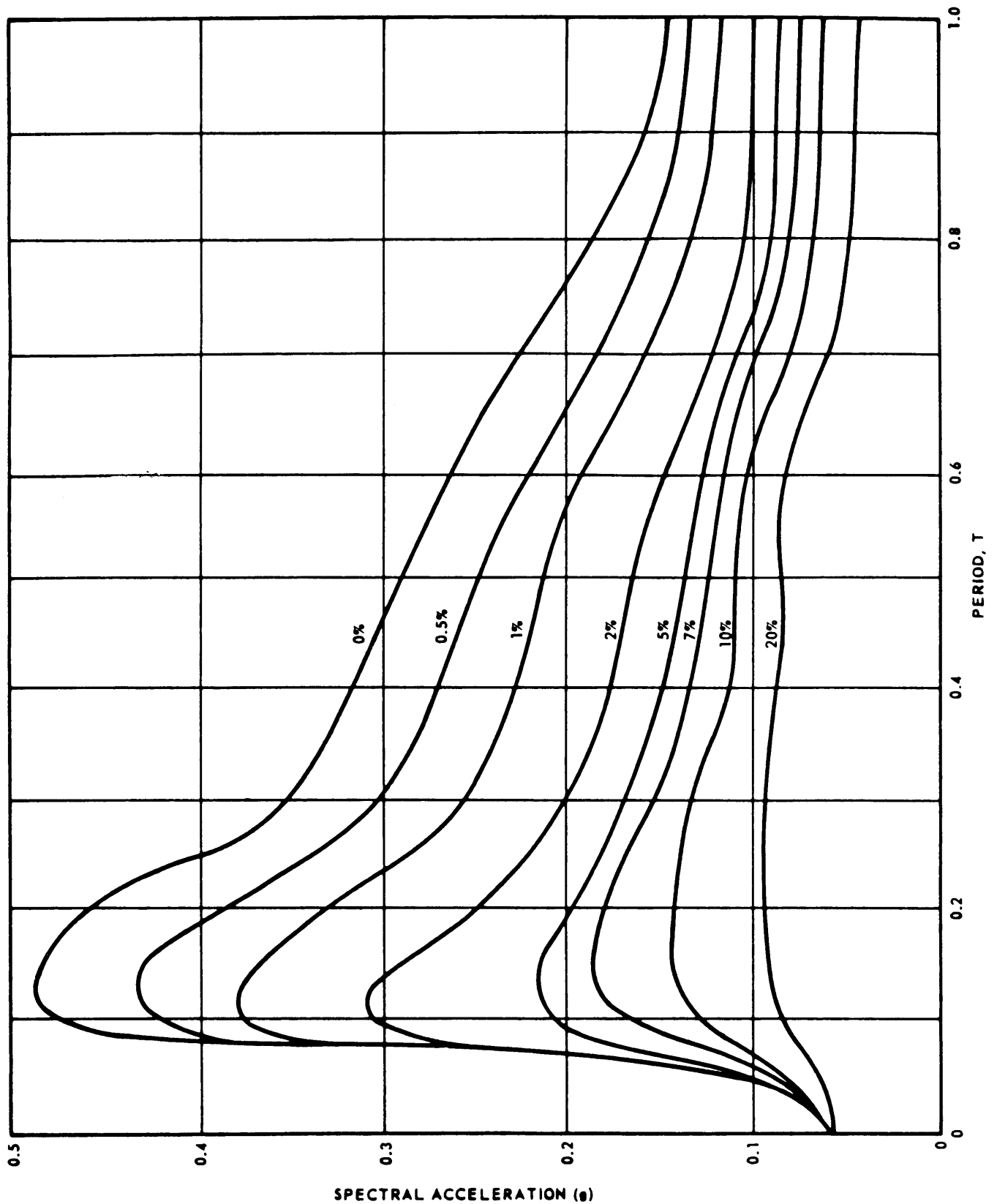
AmerGen
 TMI - UNIT 1

UPDATE - 15
 4/00

TYPICAL DIKE SECTION

CAD FILE: 6700R15.DWG

FIG 2.6-17



SPECTRAL ACCELERATION (g)

GPU Nuclear

TMI Unit-1

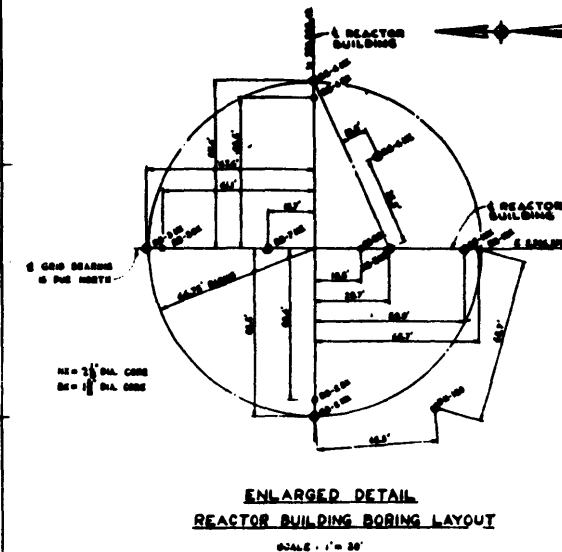
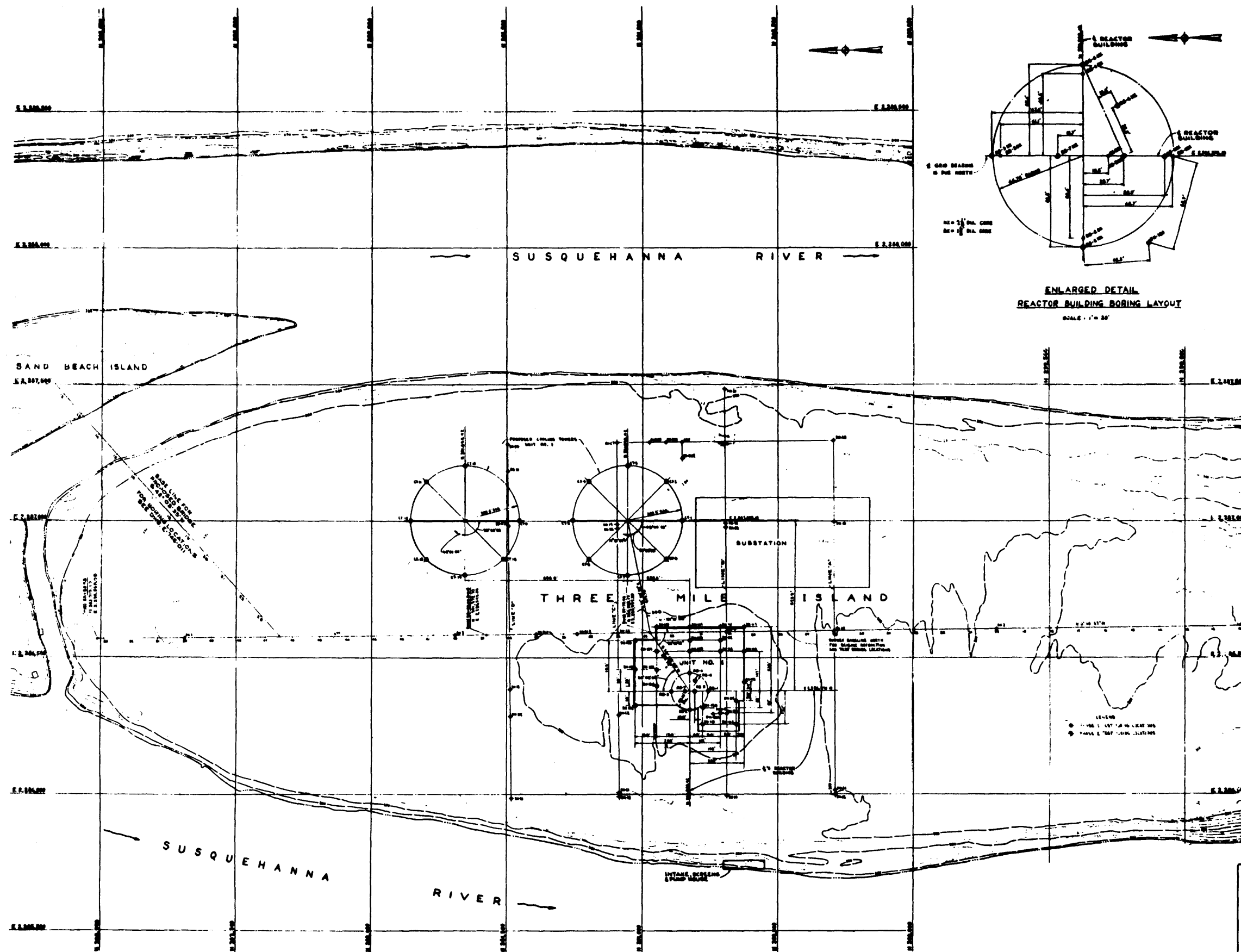
Acceleration Response Spectra

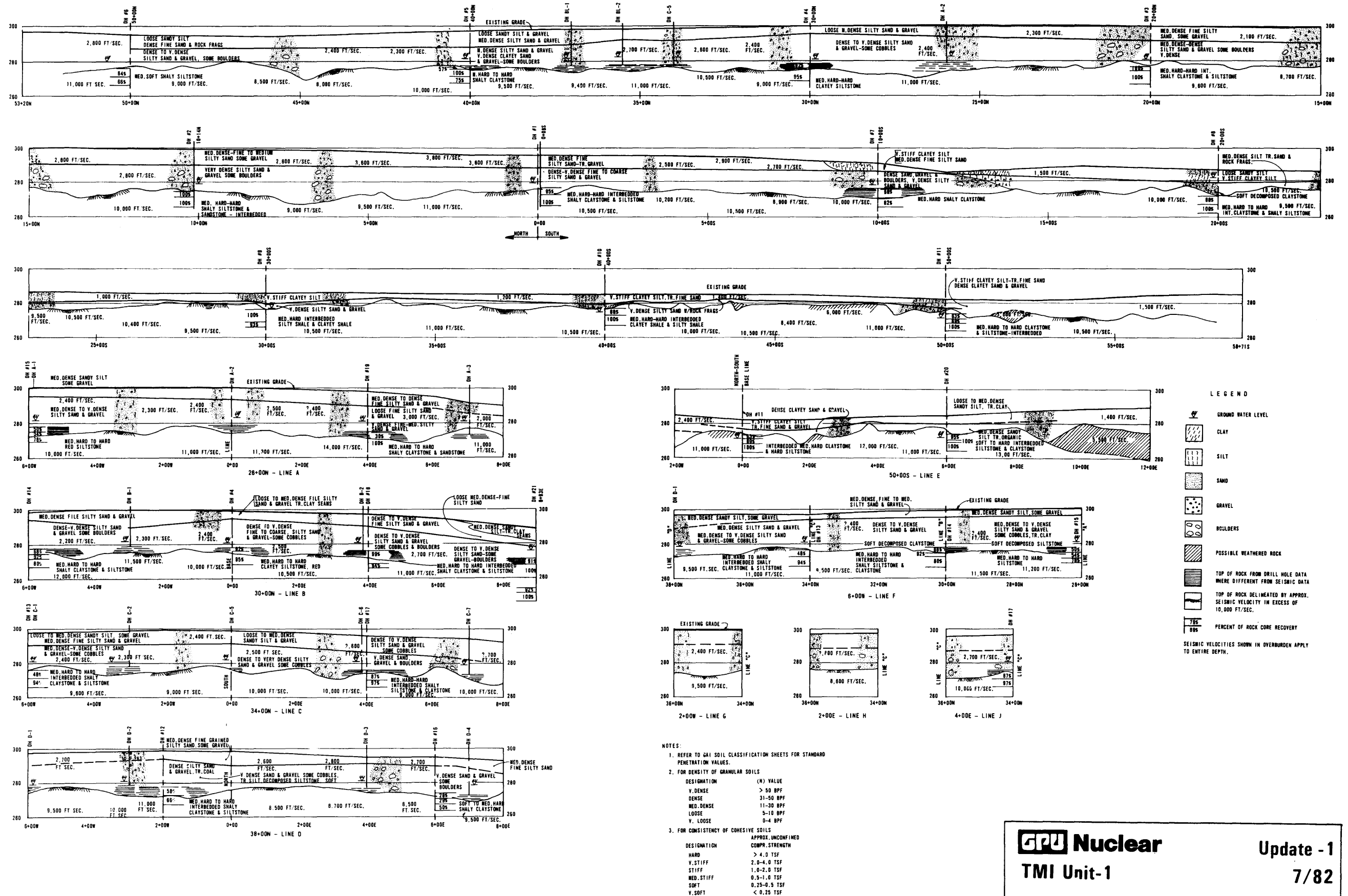
Update -1

7/82

DSAR-2.FIG-50 REV. 2

Fig. 2.7-1





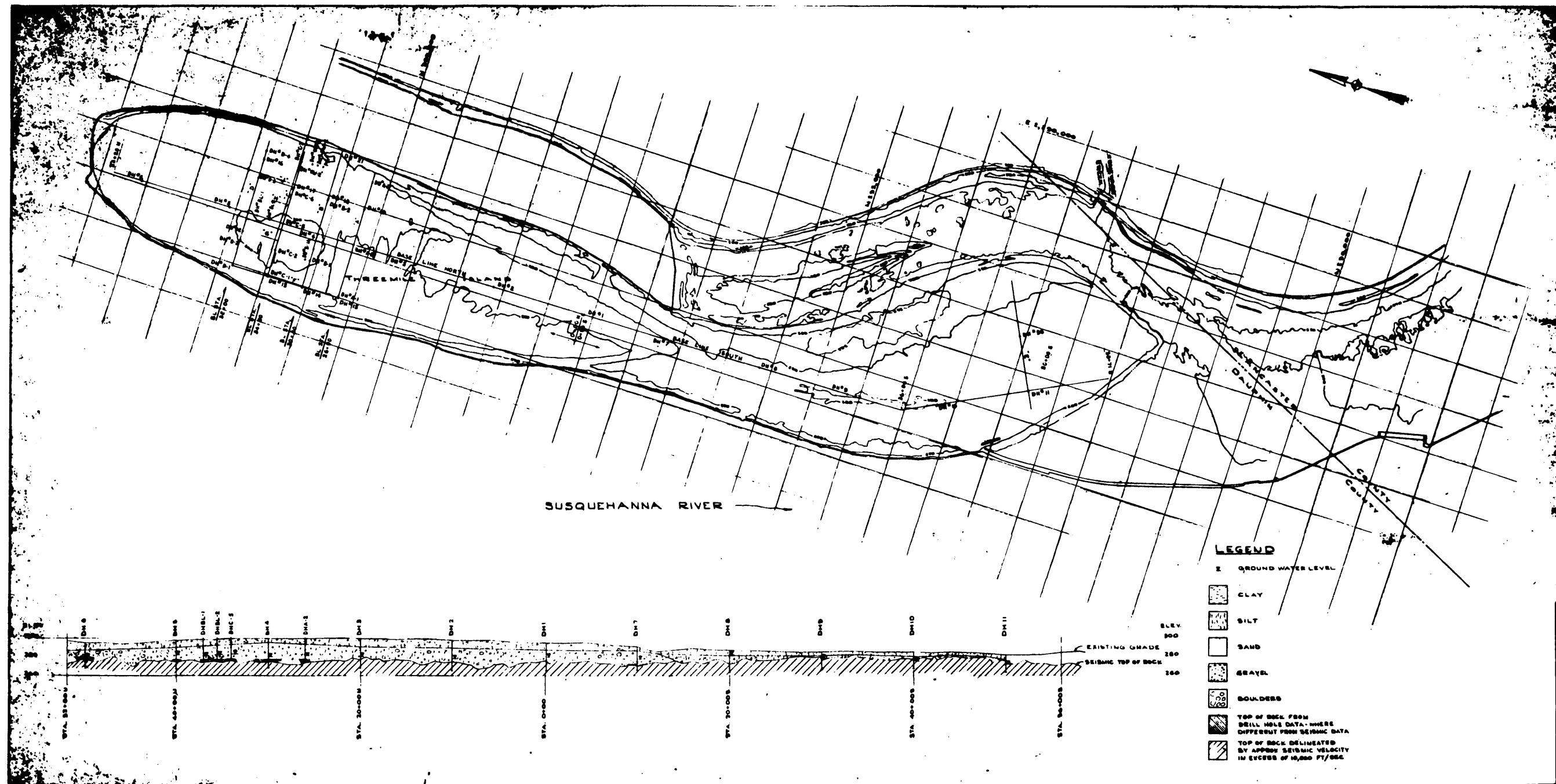
DSAR-2.FIG-52 REV. 2

GPU Nuclear
TMI Unit-1

Update -1
7/82

Subsurface Profiles

Fig. 2.7-3



DSAR-2.FIG-53 REV. 2

GPU Nuclear
TMI Unit-1

Update - 1
7/82

Seismic Refraction Test Boring Plan

Fig. 2.7-4

TMI-1 DSAR

CHAPTER 03 – FACILITY DESIGN

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3.1.6	PIPING SYSTEMS
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3.1.6.2	Deleted
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3.3	<u>SPENT FUEL HANDLING AND STORAGE</u>
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3.4	<u>ELECTRICAL SYSTEMS</u>
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3.6.2.3	<u>Methods of Operation</u>
3.6.2.4	<u>Evaluation</u>
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3.7	<u>PLANT FIRE PROTECTION PROGRAM</u>
3.7.1	PROGRAM FUNCTIONS
3.7.2	PROGRAM DESCRIPTION
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3.9.1.2	<u>System Description</u>
3.9.2	Deleted
3.10	<u>REFERENCES</u>

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LIST OF TABLES

<u>TABLE</u>	<u>TITLE</u>
3.1-1	LIST OF CLASS I STRUCTURES, SYSTEMS AND COMPONENTS
3.2-1	Deleted
3.3-1	Deleted
3.5-1	Deleted

LIST OF FIGURES

<u>FIGURE</u>	<u>TITLE</u>
3.3-1 (9.7-1)	Deleted

TMI-1 DSAR

3.0 FACILITY DESIGN

The structures, systems and components which directly or indirectly support the containment of radioactive materials are described in Chapters 3 and 4. **Design of the Independent Spent Fuel Storage Installation is as described in the 72.212 Report (Reference 44).**

3.1 GENERAL APPROACH TO DESIGN

3.1.1 CODES and STANDARDS

In general, the following codes and standards are used as applicable in the design, fabrication, and testing of components and structures; the actual codes and standards used for each system are noted in the individual sections.

- a. ASME Boiler & Pressure Vessel Code, Section II, Material Specifications
- b. ASME Boiler & Pressure Vessel Code, Section III, Nuclear Vessels
- c. ASME Boiler & Pressure Vessel Code, Section VIII, Unfired Pressure Vessels and ASME Nuclear Case Interpretations
- d. ASME Boiler & Pressure Vessel Code, Section IX, Welding Qualifications
- e. Standards of the American Society for Testing Materials
- f. USA Standard Code for Pressure Piping, USAS B31.1.0 1967 (Design of Nuclear and Power Piping; Fabrication and Testing of Power Piping)
- g. USA Standard Code for Nuclear Power Piping, USAS B31.7, February 1968 Draft including June 1968 Errata; issued for trial use and comment (fabrication, testing, and inspection of nuclear piping)
- h. Standards of the Institute of Electrical and Electronics Engineers
- i. Standards of the National Electrical Manufacturers Association
- j. Hydraulic Institute standards
- k. Standards of Tubular Exchanger Manufacturers Association
- l. Air Moving and Conditioning Association standards
- m. Standard D-100 of the American Water Works Association

3.1.2 SEISMIC CLASSIFICATION

The plant structures, components, and systems have been classified according to their function and the degree of integrity required to protect the public.

3.1.2.1 Class I

Those structures, components, and systems, including instruments and controls, whose failure might cause or increase the severity of an uncontrolled release of radioactivity, are designated Class I. When a system as a whole is referred to as Class I, certain portions not associated with loss of function of the systems may have been designated under Class II or III as appropriate. A partial listing of Class I structures, components, and systems is presented in TABLE 3.1-1.

Class I equipment and components have not been located within Class II structures.

TMI-1 DSAR

3.1.2.2 Class II

Those structures, components, and systems whose failure **would** not result in the release of substantial amounts of radioactivity are designated Class II. A partial listing of Class II structures, components, and systems follows:

- Fuel Handling Building Ventilation Supply System
- Fuel Handling Building Ventilation Exhaust System
- Auxiliary Building Ventilation Supply System
- Auxiliary Building Ventilation Exhaust System
- Fuel handling bridges
- Waste Disposal System piping and components not included in Class I or Class SIF

3.1.2.3 Class III

Those structures, components, and systems which are not classified Class I or Class II are designated Class III. This includes the following SSC:

- Natural draft cooling towers
- Access bridge
- Electrical Power Supply System
- Nuclear Services Closed Cooling Water and River Water Systems
- Control Building Ventilation System
- Instrument Air System
- Small, local instruments, such as pressure gauges, thermometers, etc. and appurtenances, such as drains and some sampling valves, attached to seismically designed systems have not been procured with seismic analysis, as the failure of these devices due to a seismic disturbance does not impair the proper operation of the related system.

3.1.3 SEISMIC DESIGN CRITERIA

3.1.3.1 Class I Structure Design

Structures classified as Class I have been designed in accordance with the following criteria:

- a. Primary steady state stresses, which include the seismic stress resulting from the design earthquake (i.e. operating basis earthquake) ground acceleration of 0.06g acting horizontally and 0.04g acting vertically and occurring simultaneously, have been maintained within the allowable working stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler & Pressure Vessel Code, Building Code Requirements for Reinforced Concrete, ACI 318 and AISC Specifications for the Design and Erection of Structural Steel for Buildings.
- b. Primary steady state stress, which include the seismic stress resulting from the maximum hypothetical earthquake (i.e. safe shutdown earthquake) ground acceleration of 0.12g acting horizontally and 0.08g acting vertically and occurring simultaneously, has been limited so that the function of the structure is not impaired.

TMI-1 DSAR

3.1.3.2 Class I Systems and Equipment Design

Components and systems classified as Class I have been designed in accordance with the following criteria:

- a. Primary steady state stresses, which included the seismic stress resulting from the design earthquake (i.e. operating basis earthquake) ground acceleration of 0.06g acting horizontally and 0.04g acting vertically and occurring simultaneously, have been maintained within the allowable working stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler & Pressure Vessel Code and USAS B31.1.0, Code for Pressure Piping.
- b. Primary steady state stress and corresponding strains, which include the seismic stress resulting from the maximum hypothetical earthquake (i.e. safe shutdown earthquake) ground acceleration of 0.12g acting horizontally and 0.08g acting vertically and occurring simultaneously, have been limited so that the function of the component, system, or structure is not impaired.
- c. As an alternative to the methodology described in a) and b) above, seismic experience data utilized in accordance with the Seismic Qualification Utilities Group (SQUG) methodology may be used to verify the seismic adequacy of existing, new, modified and replacement items on a case-by-case basis. Such evaluations are performed in a controlled and systematic manner to ensure that the item of equipment is properly represented in the earthquake experience or generic testing classes and that applicable caveats are met. In particular, each new or replacement item must be evaluated for any design changes that could reduce the seismic capacity of the equipment from that reflected in the experience data base, and all such evaluations must be documented in accordance with established procedures. SQUG methodology is applied in accordance with the SQUG Generic Implementation Procedure (Reference 16) and implementation of the SQUG methodology is controlled and documented in accordance with the Reference 17 procedure. All evaluations performed using the SQUG methodology use as input the amplified response spectra contained in EQE Report 50097-R-001 (Reference 18), GPUN Report 990-2362 (Reference 19) and EQE Calculation 42105-C-004 (Reference 20). The use of the SQUG methodology is limited to the scope of equipment covered by the equipment classes described in the SQUG Generic Implementation Procedure (GIP). The methodology is not used to verify the seismic adequacy of equipment not included within the scope of the equipment classes described in the GIP. In addition to the restrictions, inclusion rules and caveats described in the preceding paragraph and those specified by the GIP, the following restrictions as described in the Reference 17 procedure are applied to use of the SQUG methodology for verification of seismic adequacy of equipment at TMI:
 1. Comparisons of seismic capacity to demand are performed using Method B from Table 4-1 of the SQUG GIP unless the use of the Method A is justified as part of the evaluation.
 2. The anchor bolt allowables contained in GIP Appendix C, Section C.2, are used only to verify the adequacy of existing equipment, within the scope of USI A-46.

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3.1.3.3 Class II SSC Design

Structure, components, and systems classified as Class II have been designed for a ground acceleration of 0.06g in accordance with the requirements of the Uniform Building Code or other applicable code as specified in the DSAR. In addition to existing analytical/test methodology, seismic experience data applied in accordance with SQUG methodology may be used to verify seismic adequacy on a case by case basis.

3.1.3.4 Class III Design Basis

Structures, components, and systems classified as Class III have been designed in accordance with applicable building code or other applicable code requirements as specified in the DSAR.

3.1.3.5 Application of Earthquake Load

The vertical component is taken as two thirds of the horizontal component of the response spectrum. The seismic design of structures is based on the response to a ground acceleration as described above. The respective vertical and horizontal seismic components at any point on the structure have been added by summing the absolute values of the response (i.e., stress, shear, moment, or deflection) of each contributing mode due to vertical motion to the corresponding absolute values of the response of each contributing mode due to horizontal motion.

Damping Factors

The following damping factors have been applied to the maximum earthquake in the seismic design of components and structures:

<u>Component or Structure</u>	<u>Percent of Critical Damping</u>
1. Assemblies and structures	
a) Bolted or riveted	2.5
b) Welded	1.0
2. Vital piping systems	5.0 @ < 10 Hz Linearly between 10 Hz to 20 Hz 2.0 @ > 20 Hz
3. Concrete structures above ground	5.0

The modal damping values expressed as a percentage of critical damping values shown in Table 1 of U.S. Atomic Energy Commission Regulatory Guide 1.61 should be used for viscous modal damping for all modes considered in an elastic dynamic seismic analysis of NRS/ITS electrical raceway systems (Reference 21).

3.1.3.6 Methods of Seismic Analysis

The vertical and horizontal seismic components are assumed to occur simultaneously, and for systems designed to the original basis the absolute values of the seismic components were added. Seismic evaluations of existing, new and replacement structures, systems and components may be performed by combining the responses due to each of the three directional components of the earthquake using the square root of the sum of the squares (SRSS) approach as described in USNRC Regulatory Guide 1.92 (Reference 39).

For piping systems which were reanalyzed for USNRC I.E. Bulletin 79-14 the seismic components were combined per USNRC Reg. Guide 1.92 (Reference 22). Table 5.4-1 presents a comprehensive list of Class I structures, systems, and components, their applicable design criteria, and how they were analyzed for seismic loading. Seismic experience data methods (SQUG) may be used on a case-by-case basis as described in Section 3.1.3.2.c.

3.1.3.6.1 Modal Analysis

For Class I structures, components, and systems, the method of analysis used was either a modal analysis wherein modal shapes, frequencies, stresses, and participation factors are determined, or were developed as follows:

- a. The natural period of vibration of the structure, component, or system was determined.
- b. The response acceleration of the component to the seismic motion was taken from the response spectrum curve at the appropriate natural period.
- c. Stresses and deflections resulting from the combined influence of the normal loads and the additional load from the design earthquake (OBE) were calculated and checked against the limits imposed by the design standard or code.
- d. Stresses and deflections resulting from the combined influence of the normal loads and the additional loads from the maximum hypothetical earthquake (SSE) were calculated and checked to verify that deflections do not prevent functioning and that stresses do not produce rupture or excessive distortion.
- e. The dynamic analysis of critical piping systems (i.e., Class I systems) is a modal analysis based upon either a distributed or lumped mass solution depending upon the complexity of the system. Piping configurations of the larger sized pipe lines classified as SI and SII were analyzed using the lumped mass analysis method briefly described in Section 3.1.3.6.2. A general spacing criterion for use by the field to locate seismic restraints for small field run piping was developed using a distributed mass model. In general, this small piping is cold (less than 150°F) and the factor of thermal expansion requires a minimum of consideration.
- f. For the critical piping systems, the effects of differential movement between floors at elevations have been evaluated both for horizontal and vertical movements. The increase in stresses due to differential movement was found to be insignificant. The vertical seismic input at the ground is assumed to be two thirds of horizontal input. For equipment and piping systems anchored to the walls and on floors close to the walls, no amplification due to the floors is assumed. For those anchored at the middle of the floors, the amplification due to the vibrations of the floors would be investigated. Floors with fundamental frequencies 30 cps or up would be considered to be rigid, and no amplification is assumed to apply to equipment on rigid floors. In case the piping is anchored to a piece of flexible equipment, the equipment should be modeled as a part of the piping. In case the mass of the equipment is very large in comparison with that of the piping one may exclude the equipment from the model, but the response

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curve at the point of attachment to the equipment should be used instead of the floor response curve.

3.1.3.6.2 Lumped-Mass Analysis

The system is represented by a series of concentrated masses. First, the space coordinates are established for the system, and coordinates of mass points are determined. Using a static analysis, flexibility matrices corresponding to these mass points are computed. Next, the equations of motion are written in matrix form. Responses of modes with frequencies 10 percent within each other are added absolutely. Then the square root of the sum of the squares approach is taken for the rest of the modes. Since the absolute addition of the close modes method was incorporated into the computer program after certain portions of the pipe system were run, a general check of these systems was made. The effect of absolute addition increases the response less than 12 percent. Hence, for conservative reason, the response of those systems without absolute addition are multiplied by a factor of 1.12 to account for the effect. A more complete description of the analytical methods employed is contained in Reference 23. The application of the amplification curves cited in Reference 23 is explained in Reference 24. In Reference 24, the single degree-of-freedom model was not used to compare with the multi mass system directly. Instead, it was used to compare with one of the normal modes of the multi-mass system. This is the standard procedure in the method of normal mode. Even with the time history approach the same comparison is used, if the normal mode method is employed. An approximate comparison of our floor response curves with these derived from time history method (References 25 and 26) indicated that the method of Reference 24 is no less conservative than the time history method. However, this fact was verified by analyzing the lumped mass model of the Reactor Building interior concrete as shown on UFSAR Figure 5.4-3 (Reference 40) for:

- a. El Centro normalized time history method
- b. Golden Gate normalized time history method

The resulting unsmoothed ground response spectra were compared were comparable with the Three Mile Island ground response spectra (as shown in UFSAR Figure 5.4-4 (Reference 40)).

In addition to the earthquake response for the pipe system, the models described above were used to determine forces and moments with resulting stresses for any transient or permanent displacements which were induced at the support point. Stresses arising from the combination of each horizontal excitation with the vertical excitation are compared, in order to arrive at the maximum values. The interaction of the piping with the structure is taken into account by the floor response curve as described in Reference 27. The model for the buried piping is described as a semi-infinite beam on elastic foundation. The transient and, if applicable, permanent displacements of the structure can be determined. The displacements of the structures are determined independently of the piping system. The displacements are applied to the buried pipe at its point of attachment to the structures, in order to calculate maximum bending moments and shear forces. This technique for analyzing seismic stresses in buried pipes is valid only if the soil surrounding the pipe experiences uniform motion due to the earthquake excitation. This requires the use of a homogeneous backfill material for the complete length of the pipe. This was provided. The concrete pipes have been connected to the structures with a bell and spigot connection. Steel pipes have been rigidly anchored to the structures resulting in stresses up to 8000 psi during maximum hypothetical earthquake.

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3.1.3.6.3 Empirical Test

Static or dynamic analyses of certain equipment, or system components, would be extraordinarily difficult and of questionable value in determining whether the equipment could meet Class I requirements. Therefore, in some cases, testing under simulated seismic conditions was substituted in lieu of an analysis.

For example, a battery charger was mounted on a test machine table. The table was oriented at an angle to the horizontal such that the charger could be subjected to a given horizontal acceleration with two thirds of that acceleration acting vertically. The input acceleration corresponded to the predicted structure acceleration at the equipment location. Structure accelerations were determined by dynamic analysis based on the response spectrum curve with 5 percent damping for concrete structures above ground. The charger was subjected to shocks in the front-to-back and side-to-side directions. equipment functioned satisfactorily during and after the test with no visible damage.

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3.1.5 PROTECTED STRUCTURES

Those structures that contain radioactive materials are designed for the aircraft and associated loadings as specified in Appendix 3A, maximum hypothetical earthquake, tornado winds or missiles and hydrodynamic & hydrostatic flood water loads. Those structures are:

- a. Fuel Handling Building
- b. Designated portions of the Auxiliary Building (see Figure 3.1-1)

3.1.5.1 Structural Design Parameters

The loads used in design of these structures have been determined based on operating and accidental requirements, as specified below, in addition to regular loads as required by applicable codes. At normal operating conditions, the stresses in the concrete and reinforcing steel resulting from combinations of those loads listed below have been maintained in accordance with ACI 318-63 (Reference 31), "Working Stress Design."

- a. Dead load
- b. Live load
- c. Snow and wind load
- d. Equipment support
- e. Operating Basis Earthquake see 3.1.3.1.a

For accident conditions, the stresses in the concrete and reinforcing steel resulting from combinations of those loads listed below have been maintained in accordance with ACI 318-63 (Reference 31), "Ultimate Strength Design."

- a. Maximum hypothetical earthquake (SSE), see 3.1.3.1.b
- b. Tornado loads, see 3.1.5.1.1
- c. Tornado missiles, see 3.1.5.1.1
- d. Hypothetical aircraft incident, see Appendix 3A

3.1.5.1.1 Tornado Load

The tornado design requirements are:

- a. Tangential wind velocity of 300 mph. (A gust factor of 1.3 has been used for those structures designed for tornado loadings.)
- b. An external vacuum of 3 psig.
- c. Missile equivalent to a utility pole 35 ft long, 14 inches in diameter, weighing 50 pcf, and traveling at 150 mph.
- d. Missile equivalent to a 1 ton automobile traveling at grade at 150 mph.

A 300 mph wind has been applied in accordance with standard wind design practice and using applicable pressures, shape factors, and drag coefficients from Reference 34. The vacuum of 3 psig is conservative considering that most measured pressure drops have been in the order of magnitude of 1.5 psig.

3.1.5.2 Codes

These structures have been designed under the following codes:

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- a. Regulations for Protection from Fire and Panic - Commonwealth of Pennsylvania (Reference 5)
- b. Building Code Requirements for Reinforced Concrete, ACI 318-63 (Reference 31)
- c. Specifications for Structural Concrete for Buildings, ACI 301-66 (Reference 28) except as modified in the design and quality control of this building
- d. AISC Manual of Steel Construction (Reference 6)
Inspection of structural weldments that are under the purview of American Welding Society Standard D.1.1 or other non-ASME class structures shall be conducted in accordance with the provisions of Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants (VWAC), Revision 2. VWAC is not utilized for inservice inspections required by ASME Section XI.

3.1.5.3 Materials and Specifications

This design is based upon specifications giving acceptable limitations of physical and chemical properties for the structural materials used.

3.1.5.3.1 Concrete

Structural concrete work has been performed in accordance with ACI 301-66 (Reference 28). The prestressed concrete has a minimum compressive strength of 5000 psi in 28 days. The base mat also consists of 5000 psi strength concrete. Portland Cement conforms to ASTM C-150, Type II (Reference 29), modified for low heat of hydration. Concrete aggregates conform to the Pennsylvania Department of Highways Specifications (Reference 30). The type and size of aggregate, slump, and additives have been established to minimize shrinkage and creep. Neither calcium chloride nor any admixture containing calcium chloride or other chlorides, sulphides, or nitrates were used. Mixing water was controlled so as not to contain more than 100 ppm of each of the above chemical constituents.

3.1.5.3.2 Mild Reinforcing Steel

The mild steel reinforcing conforms to ASTM A-615-68 grade 60 for 14s and 18s rebars. All the smaller size rebars conform to ASTM A-615-68 grade 40.

The allowable bond and anchorage stresses used for the mild steel reinforcing were specified by the ACI 318-63 Code (Reference 31). Bond for deformed bars relies chiefly on the bearing of the deformations on the concrete and not on the adhesion or friction between the concrete and steel. Because crack-controlling reinforcement is provided, the width of the cracks occurring parallel to the reinforcing will be relatively small and should not affect the bearing capacity of the deformations or the bond characteristics of the deformed bars. It is also to be noted that the anchorage and bond stresses allowed by the ACI 318-63 Code (Reference 31) for slabs in which cracks do occur parallel with the reinforcing are equal to the values allowed by the ACI 318-63 Code (Reference 31) for beams in which cracks normally do not occur in a direction parallel to the reinforcing.

Splices at points of maximum tensile stress were avoided insofar as possible. Alternate splices for concrete reinforcement were staggered a minimum of 6 ft when the center to center spacing of bars was less than 12 inches.

3.1.5.4 **DELETED**

3.1.6 PIPING SYSTEMS

Originally, non-nuclear power piping was designed, fabricated, tested and inspected in accordance with USAS B31.1.0 – 1967 and nuclear piping was designed in accordance with USAS B31.1.0 - 1967, but it was fabricated, tested and inspected to USAS B31.7 - February 1968 Draft, including June 1968 Errata. USAS B31.7, February 1968 Draft, including June 1968 Errata (which replaced and improved on the USAS B31.1.0 Nuclear Code Cases), was issued after receipt of the construction permit. USAS B31.1.0 - 1967 was enhanced by specifying fabrication, testing and inspection to USAS B31.7 - February 1968 Draft, including June 1968 Errata for pipe classification N1, N2 and N3. "Nuclear" Piping is piping that normally contains a radioactive substance.

Nuclear valves were specified to be tested and inspected to USAS B31.7 - February 1968 Draft, including June 1968 Errata.

Pertinent sections from USAS B31.1.0 -1967 apply to Class I and II piping described hereafter:

- a. Code Paragraph 101.5.3, Earthquake
"The effect of earthquakes, where applicable, shall be considered in the design of piping, piping supports, and restraints, using data for the site as a guide in assessing the forces involved. However, earthquake need not be considered as acting concurrently with wind."
- b. Code Paragraph 101.5.4, Vibration
"Piping shall be arranged and supported with consideration of vibration."
- c. Code Paragraph 121.2.5, Sway Braces
"Sway braces or vibration dampeners shall be used to control the movement of piping due to vibration."
- d. Code Paragraph 102.3.2 (d), Additive Stresses
"The sum of the longitudinal stresses due to pressure, weight, and other sustained loads shall not exceed the allowable stress in the hot condition S_h . Where the sum of these stresses is less than S_h , the difference between S_h and this sum may be added to the term $0.25 S_h$ in Equation (1) for determining the allowable stresses range S_A ."

Eq. (1)

$$S_A = f (1.25 S_c + 0.25 S_h)$$

Where:

S_c = basic material allowable stress at minimum (cold) temperature from the Allowable Stress Table.

S_h = basic material allowable stress at maximum (hot) temperature from the Allowable Stress Tables.

f = stress range reduction factor for cyclic conditions for total number N of full temperature cycles over total number of years during which system is expected to be in operation.

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The longitudinal pressure stress S_{ip} shall be determined by dividing the end force due to internal pressure,

$$F = \frac{p \pi d^2}{4}$$

by the cross-sectional area of the pipe wall,

$$A = \frac{\pi}{4} (D_o^2 - d^2)$$

or:

$$S_{ip} = \frac{F}{A} = pd^2 / (D_o^2 - d^2)$$

Where:

S_{ip} = longitudinal pressure stress, psi

p = internal design pressure, psi

d = nominal inside diameter of the pipe, inches.

D_o = nominal outside diameter of the pipe, inches.

e. Code Paragraph 102.3.3, Limits of Calculated Stresses Due to Occasional Loads

During Operation

The sum of the longitudinal stresses produced by internal pressure, live and dead loads and those produced by occasional loads such as the temporary supporting of extra weight may exceed the allowable stress values given in the Allowable Stress Tables by the amounts and durations of time given in Paragraph 102.2.4 (as follows):

Ratings: Allowance for Variations from Normal Operation

It is recognized that variations in pressure and temperature inevitably occur, and therefore the piping system shall be considered safe for occasional operation for short periods at higher than the design pressure or temperature.

Either pressure or temperature, or both, may exceed the design values if the stress in the pipe wall calculated by the formulas using the maximum expected pressure during the variation does not exceed the S-value allowable for the maximum expected temperature during the variation by more than the following allowances for the periods of duration indicated:

- 1) Up to 15 percent increase above the S-value during 10 percent of the operating period.
- 2) Up to 20 percent increase above the S-value during 1 percent of the operating period.

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Major portions of Classes I and II piping systems as herein described have been checked by thermal, dynamic, (seismic), dead load and pressure stress analysis. For those pipe lines operating at elevated temperatures thermal analyses have been made. For those pipe lines which fall within the seismic criteria, dynamic analyses have been made as herein described giving due attention to designed supports and restraints. In addition, due recognition has been taken of stresses caused by internal pressure and the dead weight load of piping.

For the critical piping systems, the effect of differential movement between floors at different elevations, the in phase and out of phase displacements at different elevations between equipment, piping, and buildings has been evaluated, and the increase in stresses was found to be insignificant.

The results of these various analyses have been compared with allowable stress values for the various limiting conditions as follows:

- a. Piping stress for plant operation during the Operating Basis Earthquake (OBE) shall not exceed $1.2 \times S_h$ (USAS B31.1.0 Code Tables A-1 and A-2 and Code Cases N7, N9 and N10).
- b. The allowable piping stress for plant operation during the maximum hypothetical earthquake shall not exceed $1.8 \times S_h$ (USAS B31.1.0 Code Tables A-1 and A-2 and Code Cases N7, N9 and N10); but not more than $1.5 S_y$ where S_y is the material yield strength at operating temperature.
- c. Thermal stresses shall not exceed S_A as defined by Code Paragraph 102.3.2.
- d. Pressure stresses shall be calculated in accordance with Code Paragraph 102.3.2.
- e. Maximum estimated dead weight load stresses shall be calculated in accordance with conventional beam formulas.

The stresses for each section of piping have been evaluated and combined to ascertain the total maximum stress value. Giving due consideration to the pipe line conditions, this totalized stress does not exceed the Code allowable stress. This evaluation checks the piping design for the condition of operating load plus earthquake load.

The net energy developed from pipe travel and collision loss has been absorbed by the pipe whip restraints and designed to yield and form plastic hinges.

Piping not falling within Classes I and II has been considered conventional (Class III) and has been designed and checked by adherence to the Code for Pressure Piping, USAS B31.1.0 1967, without consideration of seismic effects.

The piping, support, and restraint systems have been designed in accordance with good design practice which has been previously proven on similar type systems. Piping systems will be observed during the initial operation of the unit to determine if any unusual operating characteristics are present such as water hammer or excessive vibration. Particular emphasis will be placed on high energy dissipation processes such as pump recirculation, heater drains, main steam dump to the condenser, high pressure sampling systems, etc. Conditions considered to be detrimental to reliable unit operation will be corrected as required.

3.1.6.1 DELETED

3.1.6.2 **DELETED**

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Table 3.1-1
(Sheet 1 of 2)

LIST OF CLASS I STRUCTURES, SYSTEMS AND COMPONENTS

<u>SSC</u>	<u>Design Code</u>	<u>Seismic Design</u>
*Reactor Building	UFSAR 5.2.2, 5.2.3.1	Dynamic
*Auxiliary Building	DSAR 3.1.5.2	Dynamic DSAR 3.1.3.6
*Fuel Handling Building	DSAR 3.1.5.2	Dynamic DSAR 3.1.3.6
*Control Building	UFSAR 5.4.3.1	Dynamic UFSAR 5.4.5

*Protect Class 1 waste disposal systems

Waste Disposal System

Reactor coolant bleed tanks	ASME III-C	Static

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Table 3.1-1
(Sheet 2 of 2)

LIST OF CLASS I

STRUCTURES, SYSTEMS AND COMPONENTS

SSC

Design Code

Seismic Design

3.2 **DELETED**

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**Table 3.2-1
Deleted**

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3.3 SPENT FUEL HANDLING AND STORAGE

MAGNASTOR cask operations, is addressed by the MAGNASTOR Certificate of Conformance 1031, Amendment 9, **Revision 1** (CoC 1031-9 **R1**), MAGNASTOR FSAR and TMI-1 72.212 Evaluation Report as required by 10 CFR 72 (Reference 44).

3.3.1 **DELETED**

3.3.2 **DELETED**

TMI-1 DSAR

**Table 3.3-1
Deleted**

3.4 ELECTRICAL SYSTEMS

3.4.1 DESIGN BASIS

The design of the Electrical Systems for Three Mile Island Unit 1 provides required power sources and equipment to ensure operation of plant equipment required for **SAFSTOR**. **Protection** of public health and safety is not dependent upon active safety features.

In the **SAFSTOR** condition, all Spent Nuclear Fuel is transferred to the Dry Cask System. Electrical Systems are not required to provide fuel cooling. There are no DSAR Chapter 6 accidents. The Class 1E design requirements do not apply.

Electrical power required to support monitoring and water management functions is provided by the 13.2 kV offsite power supply.

3.4.2 DELETED

3.4.3 DELETED

3.4.4 DELETED

3.4.5 DELETED

3.5 **DELETED**

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**Table 3.5-1
Deleted**

3.6 VENTILATION SYSTEMS

Plant ventilation systems provide outside air to facilitate personnel access, to maintain temperatures within acceptable limits for personnel and equipment, and to maintain positive control of air flow in areas with radioactive materials. These systems will maintain air temperature above 40F to prevent freezing (or above 60F in areas as necessary to prevent boron precipitation). These systems will introduce outside air at a rate of at least 35 cfm per person in all inhabited areas.

3.6.1 DELETED

3.6.2 AUXILIARY AND FUEL HANDLING BUILDING VENTILATION SYSTEM

3.6.2.1 Design Basis

The Auxiliary and Fuel Handling Building systems maintain suitable ambient conditions for equipment and personnel access for decommissioning activities. This includes control of the air flow to minimize the potential for personnel exposure to radioactive gases within the building. The air discharged to the environment from this system is filtered and monitored for radioactivity before it is discharged through the station vent. This equipment could provide mitigation of the radiological release from a **spill of radioactive material** but is not necessary for public safety. This system does not have a safety function.

3.6.2.2 Description of System

- A. The Auxiliary and Fuel Handling Building Ventilation System (AFHBVS) is shown on Drawings 302-831 and 302-841. Supply fan (AH-E-11) delivers filtered and tempered air from the air intake tunnel to elevations 281 ft. and 305 ft. of the Auxiliary and Fuel Handling Buildings. Supply fan (AH-E-10) delivers filtered and tempered air from the air intake tunnel to elevation 348 ft. of the Fuel Handling Building (including both Unit 1 and Unit 2 operating floors and the FHB truck bay). The electric heating coils associated with these supply fans can maintain Auxiliary and Fuel Handling Building temperature above 60F. Ventilation exhaust flow from Auxiliary and Fuel Handling Building areas below 331 ft. area in front of RB personnel hatch (TB 305) and Control Building Controlled Access Area (AH-E-20 discharge) is monitored for radioactivity (RM-A6). Fuel Handling Building operating floor (348' elevation) ventilation exhaust flow is monitored for radioactivity (RM-A-4). These air streams are combined and filtered through roughing and, HEPA filters, before being discharged to the environment by exhaust fans (AH-E-14A/D). The discharge is through an elevated stack which is monitored for radioactive particulate and gaseous activity by monitor RM-A-8.
- B. Ventilation of the Auxiliary Bldg. area above 331 ft elevation is independent from the AFBVS. Outside air and recirculated air are used for ventilation and temperature control. This system is shown on Drawing 302-831. The Hot Instrument Repair Shop, located in this area, has a separate HEPA-filtered exhaust system within the boundaries of the main system for this area.

3.6.2.3 Methods of Operation

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During normal operation, the supply fans and exhaust fans operate continuously. Where standby fans have been provided, the inactive fan is isolated by automatic control dampers. The systems is operated from the Control Room. Heating coils in the supply system are automatically controlled.

The AB 331 ventilation systems are operated continuously under local control.

3.6.2.4 Evaluation

A. The systems Includes the following instruments and controls:

1. High temperature detection devices in the discharges of the fans to stop fans and alarm the Control Room Personnel on detection of high temperature.
2. Flow switches in the exhaust fans to alarm the Control Room Personnel upon loss of air flow.
3. Flow switches in the supply fans to de-energize the electric heaters upon loss of air flow.
4. If high radiation level is detected at RM-A-4, then the supply fan (AH-E-10) is shutdown and the Fuel Handling Operating floor (Elevation 348 ft.) is isolated (AH-D-122 closed). If high radiation level is detected at RM-A-6, then the supply fan (AH-E-11) is shutdown. In the event of a high radiation signal from the vent stack monitor, RM-A8, the supply fans AH-E-10 and AH-E-11 (for Fuel Handling and Auxiliary Buildings) will stop and the waste gas discharge valve will close. The exhaust system continues to operate to exhaust air from the Auxiliary and Fuel Handling Buildings.
Flow switches provided at the discharge ducts will alarm in the Control Room upon detection of loss of air flow. Operator action is required to start the redundant standby unit.

B. The exhaust air from the Hot Instrument Repair shop is continuously sampled for potential airborne activity using a portable air sampler with a built-in high radiation alarm.

C. Fuel Handling Building Environmental Barrier

The ventilation system design was modified to prevent air flow paths between the lower elevations of the Auxiliary and Fuel Handling Building and the Fuel Handling operating floor areas. The ventilation system modifications and barriers installed after the TMI-2 accident maintain separation between TMI-1 and TMI-2 Fuel Handling and Auxiliary Buildings and isolate the Unit 1 refueling floor from the Unit 1 Auxiliary Building and from the Control Access Building. This relies on the following design features:

- 1) damper in the discharge of the FHB supply fan
- 2) isolating the supply duct and a branch duct to the FHB general area
- 3) providing an equivalent opening in the FHB supply duct to discharge air to the south side of the elevator shaft that is exhausted at the refueling floor.
- 4) isolating a branch duct in the spent fuel pool cooler area with the required air supplied from the Auxiliary Building through a wall opening at elevation 305 ft 0 inch.

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- 5) damper is provided in the exhaust duct main as it leaves the FHB upstream of the connection with Auxiliary Building main duct
- 6) dampers added to the FHB supply and exhaust ducts
- 7) enclosure of the personnel passage between Unit 1 Control Access Building and Auxiliary Building at elevation 305 ft 0 inch
- 8) enclosure of the east end of the truck bay at elevation 305 ft 0 inch

3.6.3 DELETED

3.6.4 DELETED

3.6.5 DELETED

3.6.6 DELETED

3.7 PLANT FIRE PROTECTION PROGRAM

3.7.1 PROGRAM FUNCTIONS

The functions of the plant fire protection program are as follows:

- a. Reduce the likelihood of fire occurrences;
- b. Promptly detect and extinguish fires if they occur; and
- c. Minimize the risk to the public, the environment and plant personnel resulting from fires that could result in a release of radioactive material

3.7.2 PROGRAM DESCRIPTION

The Fire Protection Program for TMI Unit 1 is maintained to ensure compliance with 10CFR50.48(f). The program is authorized by the Chairman of the Management Committee and CNO, and the Plant Manager TMI Unit 1 to assure that the appropriate levels of management are directly involved in the program. The fire protection program is defined in Administrative Procedure 1038 (Reference 1) and the TMI-1 Fire Hazards Analysis Report (Reference 2). Administrative Procedure 1038 describes the responsibilities, quality assurance, maintenance, inspection, testing and training associated with the TMI Fire Protection Program. The Fire Hazards Analysis Report (FHAR) describes fire protection systems, the fire protection configuration for confinement, detection and extinguishment of fires and the evaluation for each fire area. The program covers all fire protection concerns at TMI. 10CFR50.48 (f) applies to TMI-1 fire areas described in FHAR section 5.2 and maintaining Spent Fuel integrity as described in FHAR Section 4. TMI has implemented an incipient only fire fighting brigade (Reference 45).

3.7.3 SYSTEM DESIGN REQUIREMENTS

The Fire Protection systems and their design requirements are described in the TMI-1 Fire Hazards Analysis Report.

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3.7.3.1 Codes and Standards

The plant fire protection systems comply with the following codes and regulating bodies, where pertinent:

- a. American Nuclear Insurers (ANI).
- b. National Fire Protection Association (NFPA). NFPA codes are considered in the design, installation, maintenance and testing of fire protection systems.
- c. Commonwealth of Pennsylvania Code.
- d. American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code.
- e. American National Standards Institute (ANSI) Piping Codes.
- f. Nuclear Electric Insurance Limited (NEIL).

3.7.3.2 DELETED

3.7.3.3 Interface Relationships

The plant fire protection program includes the following significant interface with systems, other than fire protection systems:

- a. Plant drains must accommodate all water discharged automatically or by manual control of fire suppression systems.
- b. Water to fill the altitude tank) is provided by filtered water from pumps WT-P-5A and B. The tank is arranged to permit a limited amount of water in the upper portion of the tank to flow by gravity to other uses of filtered water (i.e. other than Fire Protection). All water in the tank is available for fire suppression.
- c. Actuation of smoke detectors in the Control Building ventilating ducts will close selected dampers in that system to isolate potential fires and will sound an alarm.

3.7.4 TESTING OF FIRE PROTECTION SYSTEMS

The testing of fire protection systems is governed by the TMI Administrative Procedure 1038.

3.8 OPERATING CONTROL STATIONS

In a permanently defueled condition, public safety does not depend upon a central control facility. In any design basis event, there is significant time to mitigate the event and emergency capabilities to maintain the integrity of the spent fuel cladding can be reliably managed from local and diverse locations.

3.8.1 COMMUNICATION

Radios and Paging handsets and speakers have been provided for complete unit coverage.

Non-emergency communication outside the plant is through **commercial phone line, Voice over Internet Protocol, and mobile device.**

Emergency communications systems including the Emergency Notification System (ENS) are described in DSAR section 5.7 and references described there.

3.8.2 **DELETED**

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3.9 AUXILIARY SYSTEMS

3.9.1 INSTRUMENT AIR SYSTEM

3.9.1.1 Design Basis

The function of the Instrument Air System is to deliver sufficient quantity of compressed air for operation of equipment for the (1) Ventilation Systems (DSAR section 3.6), (2) Radwaste Disposal systems (DSAR section 4.2) and other equipment not described in the DSAR. The compressed air supply does not have a safety function.

3.9.1.2 System Description

The instrument air system supply equipment is shown on drawing 302-270.
The distribution system is shown on associated drawings.

Instrument air can be supplied to the main distribution header by the following:

- a. Air compressors IA-P-5 and IA-P-6 and associated equipment located in the Turbine bldg.
- b. All pressure vessels are designed in accordance with the ASME codes for Unfired Pressure Vessels.

3.9.2 **DELETED**

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3.10 REFERENCES

1. 1038, TMI-1 Administrative Controls - Fire Protection Program
2. 990-1745, TMI-1 Fire Hazards Analysis Report.
3. Holtec International Report No. HI-2053400, Rev. 0, "Bulk Local Thermal Hydraulic Analysis for TMI – Unit 1 SFP," 6/28/2005.
4. **DELETED**
5. Regulations for Protection from Fire and Panic-Commonwealth of Pennsylvania.
6. AISC Manual of Steel Construction
7. **DELETED**
8. **DELETED**
9. **DELETED**
10. **DELETED**
11. **DELETED**
12. Holtec International Report HI-20773814, Rev. 0, "Confirmation That All TMI-1 Spent Fuel Storage Racks Are Bounded by Licensing Report," 11/14/2007.
13. **DELETED**
14. **DELETED**
15. **DELETED**
16. "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Seismic Qualification Utility Group (SQUG), Revision 3A, December 2001.
17. CC-AA-320-002, "Use of SQUG Methodology for Seismic Qualification of New and Replacement items."
18. EQE Report No. 50097-R-001, "Conservative Design In-Structure Response Spectra for Resolution of Unresolved Safety Issue A-46 for the Three Mile Island Nuclear Generating Station," Revision 1, July 1993.
19. GPUN Report No. 990-2362, "TMI Intake Screenhouse and Pumphouse SSE Spectra," Revision 0, May 1994.
20. EQE Calculation 42105-C-004, "TMI Diesel Generator Building SSE Spectra," Revision 0, May 1995.

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21. URS/John A. Blume & Associates, Report on "Analytical Techniques, Models and Seismic Evaluation of Electrical Raceway Systems".
22. GAI Report 79-14, Design Input Record: (DIR) TMI Unit 1 Piping Analysis, Rev. 3, dated 2/12/82 (Task NM-35, W.O. 04-4692-665).
23. Associates, Inc., Topical Report No. M29, "Dynamic Analysis of Vital Piping Systems Subjected to Seismic Motion."
24. Professors J.M. Biggs, and J.M. Roesset's "Seismic Analysis of Equipment Mounted on a Massive Structure."
25. "Summary of Current Seismic Design Practice for Nuclear Reactor Facilities", USAEC TID-25021, pp. 87-98.
26. FSAR of Oconee Nuclear Power Station, Duke Power Co.
27. GAI Report No. 1729
28. ACI-301-66 "Specifications for Structural Concrete for Buildings," American Concrete Institute.
29. ASTM C-150, "Specifications for Portland Cement."
30. "Specifications for Concrete Aggregates," Pennsylvania Department of Highways Specifications.
31. ACI 318-63, Building Code Requirements for Reinforced Concrete.
32. EPRI NP-6041SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Electric Power Research Institute, Revision 1, July 1991.
33. Report 240046-R-001, "Seismic Verification Review of Auxiliary Steam System Piping," EQE International, Revision 1, June 5, 1998.
34. ASCE Paper No. 3269, Wind Forces on Structures.
35. IPCEA(ICEA) Standard P-46-426, 1962, Power Cable Ampacities Vol 1 Copper Conductors, reprinted as IEEE S-135, 1984.
36. **DELETED**
37. C-1101-770-E420-018, Derating of Cable Ampacity Due to Raceway Fire Barriers.
38. **DELETED**
39. USNRC Regulatory Guide 1.92 Rev 1, "COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS"
40. TMI-1 UFSAR Rev 24

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- 41. **DELETED**
- 42. **DELETED**
- 43. **DELETED**
- 44. TMI-1 10CFR72.212, Rev **2**, "TMI-1 10 CFR 72.212 Evaluation Report"
- 45. Technical Evaluation 4214971-15, "TMI Incipient Fire Brigade Evaluation"

TMI-1 DSAR

FIGURE 3.3-1
Deleted

TMI-1 DSAR

APPENDIX 3A

GILBERT ASSOCIATES, INC., REPORT

TO

METROPOLITAN EDISON COMPANY

ON

SUMMARY OF AIRCRAFT IMPACT DESIGN

FOR

THREE MILE ISLAND NUCLEAR STATION
UNIT 1

This Report Contains:

21 pages of text
7 tables
42 figures

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1.0 INTRODUCTION

The vital structures of the Three Mile Island Nuclear Station Unit No. 1 were designed to withstand the following hypothetical aircraft impact loadings (Reference 1):

<u>Case</u>	<u>Item</u>	<u>Weight</u>	<u>Velocity</u>	<u>Effective Area</u>
A	Object	6,000 lbs.	200 knots	5 ft. diameter
B	Object	4,000 lbs.	200 knots	3 ft. diameter
C	Total Aircraft	300,000 lbs.*	200 knots	16 ft. diameter
D	Total Aircraft	200,000 lbs.	200 knots	19 ft. diameter

This report presents results of the aircraft impact study, which includes the analysis of the Reactor Building shell for various locations of the above loadings; and, the plate analysis for the Case D loading, which is the basis for wall and roof slab designs for vital structures other than the Reactor Building shell. The paper entitled "On the Stress Analysis of Structures Subjected to Aircraft Impact Forces" is applicable to this work except for the static analysis of flat plates and a minor deviation in the dynamic load factors.

Also presented are additional studies to determine if the aircraft impact loading will produce a loss of prestress force in the Reactor Building shell and, if the loss of prestress does occur, what effect this loss would have on the structure.

The final study presented is concerned with the possibility of the spalling of the anchors of the liner due to an aircraft impact on the Reactor Building shell.

2.0 DYNAMIC LOAD FACTORS

The technique used to analyze these structures is based upon establishing a dynamic load factor and applying this factor to a static solution. In determining the DLF curve the response of an undamped, linear elastic one-degree-of freedom system is used.

* The analytical check on the basis of this loading considered a uniform collapse resistance of the fuselage which indicated that the integrity of the Reactor Building would not be jeopardized. Further investigation indicated that the assumption regarding uniform deceleration is not conservative. When a revised description of deceleration for Case C was assumed, based on a variable distribution similar in principle to that shown in Figure 3A-35 for Case D, the analytical methods employed did not demonstrate that the Reactor Building would remain stable. As described in Section 3.1.1.2, the check was made for the Case C aircraft based on a constant loading and was found to be less severe than the Case D aircraft analyzed is described in Section 3.1.1.3. Because of the extremely remote probability of the Case C aircraft impacting at the most unfavorable location and attitude, the Case D aircraft for impingement was finally adopted as the design basis.

The idealized total reaction vs time curve is as shown in Figure 3A-1. This total reaction vs time curve describes quite well the results obtained from Tables 3A-6 and 3A-7 and Figures 3A-35 and 3A-36. The time variable t_n and the ratio factor for a Boeing 720 airplane (Figure 3A-1) are shown in Table 3A-1.

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The equations for dynamic load factors (DLF) for an undamped linear elastic one-degree-of-freedom system (Reference 2) are shown in Table 3A-2 in terms of the fundamental period T. The dynamic load factor is defined as the ratio between the dynamic response at any time (t) and the static response to the peak load P.

The maximum response as a function of the period T for Boeing 720 impact is calculated by the equations of Table 3A-2 and shown graphically in Figure 3A-3. Since this maximum response curve is obtained, the analysis of plate and shell structures can now be analyzed statically once the dynamic load factor is chosen from Figure 3A-3 with reference to appropriate period T.

Figure 3A-3 represents a revised curve obtained from plotting additional points at

$\frac{5}{1000} - \frac{2}{100}$ cps frequency steps.

Basically this curve is similar to the previous one but does pick up some oscillation which has been considered in the design.

3.0 ANALYSIS

The analysis of the vital structures as defined in Section 5.1.3 of the Final Safety Analysis Report is divided into two concepts: Section 3.1, Shell Analysis; and Section 3.2, Plate Analysis of this appendix.

3.1 SHELL ANALYSIS

This analysis is used for the Reactor Building. The areas of impact that are considered to be the most critical are; Apex of the Dome, Dome to Girder Transition, Girder to Cylinder Transition, Impact at Grade.

3.1.1 APEX OF THE DOME

This analysis is divided into three parts.

3.1.1.1 Analysis For Case A & B Impact Loadings

A study of the protection against missiles resulting from a hypothetical aircraft impact was made. The hypothetical missiles are defined as follows:

<u>Case</u>	<u>Weight</u>	<u>Velocity</u>	<u>Impact Surface</u>
A	6,000 lb	200 knots	5 ft diameter
B	4,000 lb	200 knots	3 ft diameter

This study consisted of an investigation of the overall structural response due to central impact of the missile on a spherical dome as well as the resistance to penetration due to a local material failure.

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3.1.1.1.1 Structural Response

a. Introduction

An upper bound of permanent displacements was determined resulting from direct central aircraft impingement on a spherical dome. The basic tool used was the displacement bound theorem for rigid-plastic continua (Reference 3). The initial velocity distribution is determined on the basis of an inelastic collision between the missile and the structure.

b. Limit Analysis for Ring Loads

First we considered a simply supported spherical cap under a ring load (See Figure 3A-4). The intensity of the load is "P" per unit length (i.e. the total load = $2\pi Pa$). A lower bound on the limiting value of "P" is found by determining a stress field which satisfies equilibrium condition and which nowhere violates the yield condition.

To obtain a lower bound, we assumed that for $r \geq a$

$$N_\phi = 0, M_\theta = M_o$$

where " M_o " is the fully plastic moment per unit length. On this basis it can be then determined that:

$$\frac{2\pi Pa}{M_o} = 1.81\pi \quad 2\pi Pa = \frac{4\pi M_o \sin \phi}{\cos \phi} \frac{1}{\ln \left[\frac{(1 + \sin \alpha)(1 - \sin \phi)}{(1 + \sin \phi)(1 - \sin \alpha)} \right]}$$

where " $2\pi Pa$ " is the total ring load.

For this condition where $\frac{a}{R}$ approaches zero.

c. Determination of the Initial Velocity Field

From the elastic solution for a concentrated load at the apex of a shell investigators have determined that " r_e ," the length over which the initial velocity distribution is felt is approximately $2\sqrt{hR}$ where "h" is the shell thickness (References 4 and 5). Therefore the initial velocity is sensibly zero for $r = r_e$.

The initial velocity distribution is considered proportional to the elastic static deflection due to a concentrated load at the apex. Because it is difficult to determine these deflections in closed form for a spherical shell it was necessary to approximate the deflection by several functions each one of which were used to determine λM velocity of the missile immediately after contact. These functions included the following:

1. Linear variation
2. Simple trigonometric variation
3. Variation suggested by a simply supported circular plate under a central concentrated load.

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4. Variation suggested by a clamped circular plate under a central concentrated load.

For these cases the numerical values of " λ ", v_o and T_o are as shown on Table 3A-3.

where: λ = fraction of responding dome mass as previously described

v_o = velocity of λM immediately after contact

T_o = initial kinetic energy of the dome

- d. Application of the Displacement Bound Theorem

Using the displacement bound theorem it can be shown that where $W_o^{U.B.}$ is the upper bound of the deflection at $r = 0$.

$$W_o^{U.B.} = \frac{T_o}{1.81\pi M_o}$$

" M_o ," the fully plastic moment per unit length, is conservatively developed considering that at plastic collapse the tendons are not carrying any load and that only the 3/8 in. steel liner acts as reinforcement with a yield strength of 30,000 psi. Therefore, $M_o = 393,000$ lb ft/ft which results in a conservative lower bound. Considering the previous cases for distribution of initial velocity, the upper bound of displacements are therefore as shown on Table 3A-4.

The average value of 0.97 inches for $W_o^{U.B.}$ is considered to be a reasonable and representative number for an upper bound deflection. It should be noted that this analysis provides only an order of magnitude determination of the upper bound of displacement and based upon comparison with actual displacement of a flat circular plate with " $a/R = 0$," that is a concentrated in lieu of a ring load, the upper bound errs on the high side.

The conclusion can be drawn on the basis of this analysis that the structural response of the dome does not produce a condition of collapse. This solution does not consider the problem of local material failures which could lead to a more serious problem than the overall structural response.

3.1.1.1.2 Local Material Failure

A study was made of the problem of local penetration making use of the modified Petry formula, wherein:

$$D = k A_p V'$$

where: D = depth (in feet) of penetration

k = experimentally obtained material's coefficient for penetration

A_p = sectional pressure obtained by dividing the weight of the missile by its maximum cross-sectional area (expressed as pounds per square foot)

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$$V' = \text{velocity factor expressed as } \log_{10} \left(1 + \frac{V^2}{215,000} \right)$$

where "V" represents the terminal or striking velocity in feet per second.

On the basis of "k" being equal to 0.0023 the penetrations are as follows:

Case A D = 0.128 ft = 1.54 in.

Case B D = 0.237 ft = 2.85 in.

both of which are less than the limit established for valid use of this equation. The material coefficient "k" has been determined by experimental results for reinforced concrete with different compressive strengths. Variation in material properties will affect the k-value used and thereby the depth of penetration. However, the thickness of the reinforced concrete used for aircraft protected structures exceeds the lower limit for use of the modified Petry formula. A study of missile penetration was made using the Ballistic Research Laboratories formula (Reference 22) which resulted in deeper penetration but well within acceptable limits.

The impact of detached aircraft elements such as engines etc., will hit at a distance away from the center of impact. The top and bottom surface of the shell (See Figure 3A-15) in this location will be in compression, and the combined flexural stress less than at the center of impact. The engines have not been considered to hit at center of impact, however, the effect of the whole Class D aircraft remaining intact has been evaluated as discussed in Section 3.1.1.3.

3.1.1.2 Analysis for Case C Impact Loading

A study was made of the protection against a hypothetical aircraft impingement by an aircraft weighing 300,000 pounds, traveling at a velocity of 200 knots and impacting over an area with an effective diameter of 16 feet. The stability of the reactor containment structure was verified by means of dynamic elastic analyses for the impingement of the total aircraft.

The effect of a large aircraft impingement against the apex of the reinforced concrete dome was as studied by calculating the dynamic response of an elastic solid of revolution to time-dependent forces acting on the area of impact, as shown in Figure 3A-5. The magnitude and duration of the impact forces were determined according to the mass, structural characteristics, and vertical component of the velocity of the aircraft. The grid for the finite-element idealization of the dome is given in Figure 3A-6. Based on virtual work, the equilibrium equations for the entire structure are formulated as follows:

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$$\ddot{\vec{d}} + [\alpha(k) + 2\beta(m)] \dot{\vec{d}} + (k) \vec{d} = \vec{f}$$

where:

(m) = mass matrix

$\alpha(k) + 2\beta(m)$ = damping matrix

(k) = stiffness matrix

$\vec{d}$ = displacement vector

Dots indicate time derivatives. These equations are integrated by means of a Predictor-Corrector method with a Runge-Kutta-Gill starting procedure using a computer program developed at Franklin Institute Research Laboratories. FIRL is acting as Consultant to GAI in connection with this problem.

Evaluation of the results, in conjunction with a procedure proposed by Dr. Steven Batterman and utilized in the study described herein to calculate limit (final) displacements in a rigid-perfect plastic shell will lead to safe estimates of the size and velocity of the largest aircraft that may impinge upon the containment building without jeopardizing its structural integrity.

An analysis was performed considering the following loading condition (Refer to Figure 3A-5 for nomenclature):

$$P_n = 200 \text{ psi}$$

$$t_1 = 0$$

$$t_2 = t_3 = 0.16 \text{ sec.}$$

The diameter of the impact area was considered to be 16 feet. In order to obtain a preliminary indication of displacements, this analysis was performed on the basis of the conservative assumption of no internal damping. Also to simplify the solution, the steel liner was not considered.

The equivalent diameter of the fuselage of the B707 type aircraft is approximately 13.3 feet. The assumed impact area is considered to be reasonably indicative of the impact area of such an aircraft considering the significant distortion which will occur to the fuselage as well as the load distribution afforded by the concrete to the middle surface of the dome.

The loading pressure of the 300,000 lb aircraft without impact would be 10.4 psi. Therefore the loading considered represents a constant deceleration of the impacting aircraft of 20g. That means that the entire aircraft remains intact and all elements decelerate at 20g. This represents an equivalent load on the fuselage of the aircraft of 5,800,000 lb, which it is estimated would result in gross collapse of the aircraft.

The analysis indicates that the maximum displacements and stresses all of which occur at the center of impact (i.e. the apex of the dome) are as follows:

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	<u>Displacement (in.)</u>	<u>Stress (psi)</u>
Maximum	-0.98	-2264 + 354
Static	-0.66	-1832 + 346

The displacement at the apex of the dome as a function of time is depicted in Figure 3A-7. This graphical representation of displacements indicates that the most severe duration of the loading is equal to or greater than 0.16 seconds. The static displacement is that produced by the 200 psi loading applied for an infinite period. Figure 3A-8 depicts the displacements and stresses which occur at time 0.16 seconds after impact which is the instant of maximum displacement at the dome apex.

The loading considered in this analysis represents the case where the aircraft with all its engines, fuel tanks, and wings remains intact and the total resulting load is applied on the nose of the aircraft. It has been concluded that the resultant load due to one or both wings shearing off the fuselage and impacting against the dome will result in a less critical condition than that previously considered. The static displacement of the dome at the point of impact of one engine is approximately 0.1 inches. The displacement results from a loading equivalent to a 20g deceleration applied for an infinite period. The physical separation of engines is sufficient to produce only a minimal increase in displacements due to the fact that the wings or engines upon separation from the remainder of the aircraft would be traveling at significantly reduced speed.

The results obtained from the Analysis for Case C Impact Loading were not used for the final design of the Reactor Building Dome and Shell.

3.1.1.3 Analysis for Case D Impact Loading

A study was made of the protection against a hypothetical aircraft impingement by an aircraft weighing 200,000 pounds, traveling at a velocity of 200 knots and impacting over an area with an effective diameter of 19 feet using the same analytical techniques described in Section 3.1.1.2 of this appendix. The final design of the Reactor Building was based upon the case D Impact Loading Analysis.

The load-time curves for the 200,000 lb. aircraft as shown on Figure 3A-2 were used as the design basis and was derived from the geometry, structure characteristics, and mass distribution of the B720 type aircraft.

Two loading cases were considered; one in which the outboard engines, wing structure, and outboard fuel are separated from the aircraft at a prescribed time following impact (Table 3A-6 and Figure 3A-2), and one in which the aircraft is assumed to remain intact after the initial impact (Table 3A-7 and Figure 3A-2).

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The reaction load can be derived as follows:

Linear Impulse = Linear Momentum

$$\text{and: } -R \Delta t = \mu L \frac{[V - \Delta V]}{LM_2} - \frac{[\mu LV + \mu XV]}{LM_1}$$

$$R \Delta t = \mu L (\Delta V) + \mu X (V)$$

$$R = \mu L \frac{(\Delta V)}{\Delta t} + \frac{\mu \Delta X}{\Delta t} (V)$$

In the limit: $R = \mu La + \mu V^2$

and $\mu L = M =$ mass of uncrushed aircraft
 thus $\mu La = Ma = P_B$ (the unbalanced force
 on the uncrushed
 portion of aircraft)

therefore: $R = P_B + \mu V^2$

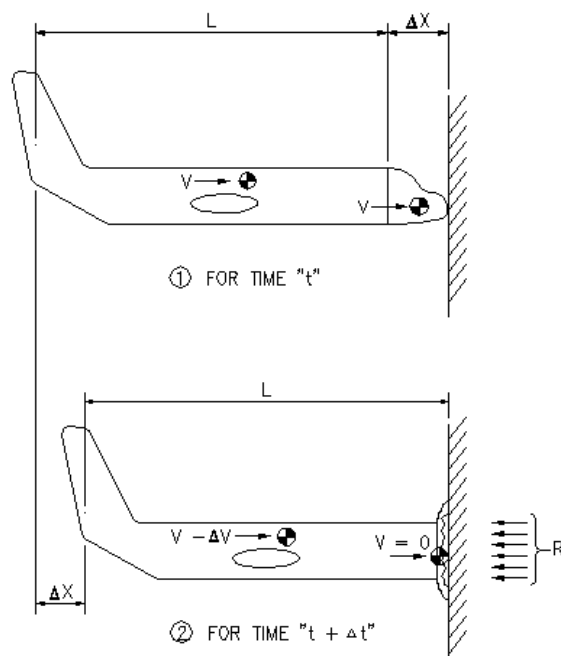
where:

$R =$ Total reaction load on rigid surface in pounds

$P_B =$ Load in pounds required to crush or deform fuselage longitudinally

$\mu =$ mass of aircraft per unit length (slugs/ft)

$V =$ velocity in ft/sec of uncrushed portion of aircraft
 at any time or distance during the impact



Instrumented data from a full-scale C-119 aircraft impact into a vertical wall indicated that the results given by the above momentum exchange principle for a B720 aircraft were of the right order of magnitude; however, the actual reaction load (P_B) to the wall by the C-119 aircraft was not recorded. The rate of change of the aircraft velocity was determined, however, by high speed film analysis and compared with the rate of velocity change with the B720 aircraft as shown in Figures 3A-9 & 3A-10. This comparison shows that both aircraft decelerate at approximately the same rate; however, the B720 required more than twice as much crush distance because of its higher initial impact velocity.

* Structural properties of the aircraft considered in Reference 1 were obtained from The Boeing Company.

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The reaction load as a function of time is presented in Figure 3A-2 and 3A-35 for the B720. Note that the peak reaction occurs as the wing and fuselage are crushed between the front and rear spars. This is caused by the fact that the " μ^{V2} " is largest at this location (" μ " is very high here as shown by the mass distribution in Figure 3A-11). Also note that the fuselage deceleration is highest when the reaction load is rather low. This phenomena is caused by the reduced mass of the uncrushed portion of the aircraft being decelerated by the relatively constant buckling load (P_B) acting on the uncrushed portion. See Tables 3A-6 and 3A-7, and Figure 3A-2. The buckling load (P_B) of the fuselage is shown in Figure 3A-12*. The average diameter of the fuselage for the B720 aircraft is 13.3 feet. As can be seen from the load-time curve the peak load occurs after the wings have impacted against the dome. Considering that the wings constitute a large proportion of the total mass, it is considered justifiable to consider a portion of the wings that is in contact with the dome at the peak load as additional impact area. Considering this additional area and the load distribution afforded by the concrete to the middle surface of the dome, the effective diameter of the impact area is 19 feet.

The analysis indicates that maximum induced extreme compressive fiber stress of 7742 psi and displacement of 1.81 inches occurs at the center of the impact area.

For the two loading cases, the displacements of the apex of the dome as a function of time are shown in Figures 3A-13 and 3A-14. For comparison, the displacement of the points at a radius of 115.2 and 268.8 inches are also shown. Stresses due to the aircraft impingement and prestress near the apex and at a radius of 85 inches are shown in Figures 3A-15 and 3A-16.

The maximum combined extreme compressive fiber stress is approximately 9372 psi. This includes stresses due to aircraft related loads and the prestress loads. It has been recognized that biaxial stress conditions as produced in the Reactor Building due to prestress, increases the ultimate strength of concrete. Considering that the minimum cylinder strength of the concrete for the Reactor Building is based upon a 28 day curing time, an increase of 20 percent in strength can be justified (Reference 12) considering that the concrete will have cured more than two years when the plant is in operation. The present records of the 90 day compressive strength of the same concrete used in the shell is in agreement with this strength increase. The strength of concrete under biaxial stress when $\sigma_\phi \cong \sigma_\theta$ was determined by Rosenthal and Glucklich (Reference 26) to be $2.2f'_c = 2.2 \times 6000 = 13,200$ psi.

The local radial tensile stresses around the dome tendon conduit was determined utilizing the Finite Element Method including in the model in the effect of the 5" f, Schedule 40 steel tendon conduit. The model shown in Figure 3A-40 consist of an axisymmetric solid of revolution. The loads applied to the model are:

- a. Radial prestress from each of the three tendon layers
- b. Meridional prestress
- c. Meridional stresses due to aircraft impact

The resulting radial tensile or compressive stresses are shown on Figure 3A-41. The radial stresses shown in parenthesis are the results of the loads described above excluding the external dome pressure due to the crushed fuselage. The radial stresses have been adjusted for this external pressure. The stress concentration around the conduit was greatly absorbed by the steel conduit and thereby reduced the expected larger tensile and compressive stresses in the adjacent concrete. Two local areas of concrete tensile stresses exist on an axis $\pm 45^\circ$ from the axis parallel with the dome surface. Any tensile cracking in this region would be limited by

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the confining compressive stresses. A local tensile crack in these regions of the concrete close to the conduit will increase the biaxial concrete compressive stress to approximately 9.150 psi < 13,200 psi considering an average stress over the uncracked portion of concrete. This is the worst condition which exists approximately 8 inches from the dome surface, and is a conservative method for evaluating the load carrying capability of the concrete. In fact, essentially all concrete is subjected to triaxial compressive stresses which further increases the concrete capacity. A schematic showing the stress state of the dome in relationship to the conduit is shown in Figure 3A-42.

The maximum compressive stress at the center of the first tendon conduit is $f_c = 5000$ psi, with no radial tensile stresses.

The loading considered in this analysis of the dome were:

- a. Prestress load
- b. Radial tensile stresses due to tendon curvature
- c. Stresses introduced due to aircraft impact.

Stresses in the dome at the edge of the loaded area of the aircraft are for wings and engines attached:

	<u>Meridional</u>		<u>Hoop</u>
f_c top =	- 6268 psi	f_c top =	-6419 psi
f_c bottom =	- 4931 psi	f_c bottom =	+128 psi
Shear v_c =	470 psi		

The concrete compressive stresses are acceptable; and the shear stress below that permitted by ACI 318-63, Chapter 26.

Stresses in the dome 36 inches out from the loaded area of the crushed aircraft for wings and engines attached (Figure 3A-37) are:

	<u>Meridional</u>		<u>Hoop</u>
f_c top =	-3600 psi	f_c top =	-5268 psi
f_c bottom =	-3440 psi	f_c bottom =	-342 psi
Shear v_c =	356 psi		

The concrete compressive stresses are acceptable; and the shear stress below that presented by ACI 318-63, Chapter 26. (Figures 3A-38 and 3A-39)

It is of interest to note that the high fiber stresses at the apex of the dome are reduced significantly to the location at the periphery of the loaded area and at a distance "d" from the periphery of the loaded area.

No special investigation has been made for the case of simultaneous impact of the partially disabled aircraft and the detached elements (outboard engines, wing structure, and outboard fuel) on separate locations. The impact of the aircraft with a 19 ft-0 in. diameter impact area is the most critical load case. It will be seen by inspection of Figure 3A-15 that detached elements will impact in areas at significantly lower stress level. It is considered improbable that the

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partially disabled aircraft can impact on the structures at a location that has been damaged by the previous impact of the detached elements. The impact of the entire aircraft on the dome has been analyzed and the results are shown in Figure 3A-16.

Therefore, the conclusion can be safely drawn that the dome will not collapse due to the established loading condition.

3.1.2 DOME TO GIRDER TRANSITION

This analysis is in accordance with the methods described in Section 5.2.4.1 and was made for Case D loading by a hypothetical aircraft described in Section 3.1.1.3. The non-axisymmetrical load is represented by a Fourier Series and has the general dimensions and shape as shown in Figure 3A-17. The stress resultants are shown in Figure 3A-18.

3.1.3 GIRDER TO CYLINDER TRANSITION (SPRING LINE)

Analyzed the same as 3.1.2 above. The stress resultants are as shown in Figures 3A-19 and 3A-20.

3.1.4 IMPACT AT GRADE

Analyzed the same as 3.1.2. The stress resultants are as shown in Figure 3A-21.

3.2 PLATE ANALYSIS

This analysis was made for the Case D loading by a hypothetical aircraft as described in Section 3.1.1.3.

3.2.1 FUNDAMENTAL FREQUENCY

It is readily seen in Figure 3A-3 that as long as the fundamental frequency of the plate is greater than 10 cps, or less than 6 cps, the dynamic load factor will be less than 1.32. In the present plate analysis, all of the slabs have the fundamental frequency greater than 10 cps. The fundamental frequency calculated for each slab (except two slabs which will be explained subsequently) is based on the assumption of simply-supported boundary conditions. This assumption will lead to a lower value of fundamental frequency for the current plates because their boundaries are actually restrained rather than simply-supported. The value of E employed in the dynamic analyses was equal to the static modulus, and no account was taken of the increase in E resulting from the dynamic load effect in order to compensate for any reduction of E due to high stress levels. This approach is felt to be conservative because (1) the structure fundamental frequencies fall to the left of the peak in the maximum DLF vs. frequency curve and (2) the area of impact, where the highest stress levels occur, is relatively small compared with the total structure. These low values of fundamental frequency will give a conservative dynamic load factor as can be seen in Figure 3A-3. Consequently, variation of the elastic properties and edge conditions from the assumed parameters would lead to a reduction of the magnitude of the dynamic response. When the dynamic load factor value falls below unity, a minimum factor of 1.0 is used. The theoretical background of calculating fundamental frequency of simply supported plate is straightforward and well documented. (References 2, 13, and 14). The well known formula of natural frequency is:

$$a_{mn} = X^2 \sqrt{\frac{D}{\rho h}} \sqrt{(\frac{m}{a})^2 + (\frac{n}{b})^2}$$

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$$\frac{p h^3}{12 D} \left(\frac{a^3}{b} + \frac{b^3}{a} \right)$$

where m and n denote the mode number; D is the flexural rigidity; p is the density; and a , b , and h , are length, width, and thickness of plate, respectively.

For the two exceptional plates where the boundary conditions are more likely to be fixed, the fundamental frequencies are calculated based on fixed boundaries. There is no exact solution of fundamental frequency for such cases; however, numerical approximations which are based on energy principle are available. (References 2, 15, and 16). The present calculations of fundamental frequency of fixed plate are obtained from the tables and suggested formulae in Chapter 5 of "Introduction to Structural Dynamics" (Reference 2).

After obtaining the fundamental frequency of each slab and the dynamic load factor, the remaining work is a static slab analysis. This approximate dynamic analysis technique is equivalent to the assumption that the DLF's of all modes are equal to that of the fundamental mode. For the computation of maximum response for design purposes, this approach is conservative for the following reasons:

- a. The fundamental periods of all the slabs fall to the left of the peak in the diagram of maximum DLF vs. period, thus indicating that the higher modes have smaller maximum DLF's.
- b. The maximum DLF's of the various modes do not occur simultaneously.
- c. The maximum bending moment results when the impact occurs at the center of the slab. For this case, the fundamental mode dominates the response.
- d. The maximum transverse shear is obtained when the impact takes place near a supporting edge, in which case most of the load is transferred directly to the support with little dynamic participation of the slab.

In order to assess the validity of these assertions, the study reported in Reference 21 was recently extended to evaluate the dynamic response of a typical slab subject to impact at various locations (center, edge, and intermediate points). In this study, mode superposition, using up to 900 modes (30 harmonics in each coordinate direction), was employed with the expressions for the DLF given in Table 3A-2 to calculate the time history of the displacements and stress resultants at critical positions for the Boeing 720 impact. The maximum values of these quantities were compared with the corresponding static values, and it was found that for every one the ratio of maximum dynamic value to static value was less than the maximum DLF for the fundamental mode. Therefore, the design values obtained by factoring the static responses by the maximum fundamental DLF would in every case be larger than the actual maximum, indicating that this procedure will yield a conservative design.

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3.2.2 FINITE-ELEMENT ANALYSIS FOR SLABS

The present finite-element analysis is based upon a rectangular plate element (Figure 3A-22) as developed in References 3A-16 and 3A-17. Each nodal point has 6 degrees-of-freedom. A comprehensive explanation of the satisfaction of the "completeness" and "compatibility" of the chosen displacement function is given in References 3A-18 and 3A-19. The convergence of the solution accuracy vs grid refinement is monotonic and rapid as evidenced in various References 3A-16, 3A-17, 3A-18, and 3A-19. For the problem of plate in bending, 16 elements discretization can lead the solution of deflections to a small error of less than five percent as compared with classical solutions. The element number used in the present calculation ranges from 25 to 64 for various slabs.

Since the finite-element method is much less restricted to the geometry and boundary conditions than the classical solution, the actual geometry and boundary conditions are represented in the present calculation without modifications.

As an illustration of the present slab analysis and design, an example of roof slab at the heat exchanger vault of the auxiliary building at Elevation 305 ft is chosen. The dimensions and boundary conditions are shown in Figure 3A-23(a). The dynamic load factor, based on the elastic undamped one degree-of-freedom assumptions, was found to be 1.188.

As shown in Figure 3A-23(a), nine critical impact positions, which produce the critical moments and shears at various sections, were examined. As a simplified demonstration, the moment diagrams along line AA are shown in Figure 3A-23(b). The top and bottom reinforcements corresponding to the moments in Figure 3A-23(b) were designed and shown on GAI Drawing 422030. The shear reinforcement designed for this slab is shown on GAI Drawing 422031. The shear reinforcements were designed on the basis of the aforementioned finite-element computer program output of shears.

The slab is a rectangle 121'-0 by 55'-0 by 6'-0 as shown on GAI Drawings 422030 and 422031. The slab is supported along all four edges and along the East-West centerline. For the purpose of this analysis, only the southern half has been considered. Moment curves have been plotted for the different impact positions. Five impact positions have been considered. An envelope was then constructed in order to obtain maximum moments at any point along section A-A, Figure 3A-23. This envelope was used to calculate the required reinforcement. A similar procedure was used to calculate the moments for another 4 strips of the slab in the North-South direction and 5 strips in the East-West direction.

3.2.3 DESIGN CRITERIA FOR REINFORCING

The wall and roof slabs exposed to aircraft impact have been designed according to the following criteria.

- a. The main reinforcing is designed on the basis of bending moment diagrams obtained from computer printouts using Case D loading and according to the ACI 318-63, Ultimate Strength Design Method, Chapter 15.
- b. The shear reinforcing is designed on the basis of the shear force diagram obtained from computer print-outs using the Case D loading and according to the ACI 318-63, Ultimate Strength Design Method, Chapter 17.

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- c. Anchorage for the reinforcing bars is provided by carrying the reinforcing bars past the theoretical cut-off point a distance sufficient to develop the ultimate strength of the bar.
- d. The allowable bond stresses are calculated on the basis of ACI 318-63, Chapter 18.
- e. No welding has been done on reinforcing bars.
- f. In the design no special construction methods were specified.

3.2.4 DESIGN CHECK

It is recognized that the elastic analysis employed for the flat slabs is conservative and that the resulting design does have a capacity in excess of that implied by the load considered for slab design.

As a conclusion of this design, each wall and roof slab exposed to aircraft impact was checked utilizing the Yield Line Theory (Reference 25).

These aircraft-protected structures can withstand at least a load equal to $1.52 \times (17,500,000) = 26,400,000$ lbs. This load represents the maximum load as determined in the previous section, with the peak DLF = 1.52 as shown in Figure 3A-3.

4.0 ADDITIONAL DETAIL STUDIES

Some additional studies on the detail structural analysis of the aircraft impact on the containment vessel have been made:

- a. Bearing failure of concrete in the neighborhood of anchors of tendons under direct impact.
- b. Shear-off of the anchors of the dome tendons, vertical tendons, and hoop tendons.
- c. Spalling due to aircraft impact on the outside wall.
- d. Impact effects on equipment and components.

4.1 BEARING FAILURE OF CONCRETE UNDER DIRECT IMPACT

The bearing capacity of concrete, according to ACI-318-63, is $1.9 \times 0.375 f'_c = 3560$ psi. The aircraft impact force of 21,100 kips (assuming a dynamic load factor of 1.2), according to the 19 foot diameter (283.5 ft²) circular normal impact area, produces an impact pressure of $21.1 \times 10^6 / 283.5 \times 144 = 517$ psi. It is readily seen that the aircraft impact does not cause bearing failure as long as the total impact force is distributed on an area greater than 41.1 ft².

(Reference 2) (corresponding to a bearing stress of 3560 psi). Thus, it may be concluded that bearing failure can be prevented as long as the impact force is distributed on an area greater

than $\frac{41.1}{283.5} = 14.5$ percent of the normal impact area.

Intuitively, it is believed that no matter where the aircraft hits, the impact area should be at least 14.5 percent of the area of the case when the aircraft impacts on a flat wall.

The anchors of the dome and vertical tendons are embedded in concrete at least one foot in depth. No damage of the anchors is possible when the aircraft hits normal to the anchorages.

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Although the hoop tendon anchorages are not embedded in concrete, the aircraft would have to hit in a tangential direction to the containment vessel to produce a bearing failure of the anchorage.

This being the case, it is unlikely that the total impact force will be concentrated over less than 14.5 percent of the normal impact area; therefore, a bearing failure of the concrete at the hoop tendon anchorages will not occur.

4.2 SHEAR-OFF THE ANCHORS

Three cases have to be investigated when considering the possibility of shearing-off the anchors of the tendons due to aircraft impact, particularly the impact of engines and sharp object.

4.2.1 CASE A: SHEAR-OFF THE ANCHORS OF VERTICAL TENDONS

The aircraft may travel in the direction shown in Figure 3A-26 such that the anchors of vertical tendons may be sheared-off. Based on the investigation conducted in Reference 1, the information is available that the maximum response of statically equivalent impact load is 21.1×10^6 lb (assuming conservatively a dynamic load factor of 1.2) and the maximum impact area is a 19 ft diameter circle (assuming the aircraft strikes normal to a flat wall). The shear strength at Section A-A shown in Figure 3A-27 governs whether or not the aircraft will shear the concrete and impinge at the anchors. The shear stress at Section A-A must be calculated for the worst condition of impact loading and compared with the shear capacity.

Considering the maximum aircraft impact area with a diameter of 19 ft, the maximum number of covered tendons are:

$$N = \frac{19 \text{ ft}}{\text{C. to C. dist. between anchors}} = \frac{19 \times 12}{30.5} = 7.47$$

As a very conservative estimation, the impact force on the area between two anchor centers is:

$$P = \frac{\text{Max. response}}{N} = \frac{21.1 \times 10^6}{7.47} = 2.83 \times 10^6 \text{ lb/tendon}$$

The area available for resisting the shear can be estimated from Figures 3A-27 and 3A-28.

$$A = 28" \times 30.5" = 853 \text{ sq in.}$$

It is reasonable to assume that the load applied above Section A-A is

$$\text{proportional to the area of impact. } \frac{[d \times 25]}{15 \times 25} \times 21.1 \times 10^6 \text{ lbs.}$$

where $d = 2'-2"$.

The applied load = 3.1×10^6 lbs which results in a shearing

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stress across Section A-A equal to $\frac{3.1 \times 10^6 \text{ lbs.}}{25 (12) 28} = 369 \text{ psi} < 600 \text{ psi}$

Ultimate shear strength of concrete $f_{ult} = 600 \text{ psi}$. Although unlikely, it is, however, assumed that nine anchors are sheared-off by the aircraft impact. As illustrated in Figure 3A-29(a) it is seen that Section AA is of primary concern if vertical tendons fail under the impact. The moment m_1 , caused by the aircraft impact, has to be resisted by the moment m_2 . The moment caused by the aircraft impact is:

$$m_1 = 21.1 \times 10^3 \times 166.5 \times 12 = 4.2 \times 10^7 \text{ in-kips}$$

The moment due to the undestroyed tendon prestressed forces resisting, m_1 is:

$$m_2 = T(d_1 + d_2)$$

where T is the total prestressed forces on tension side of N.A., for the case when there is no tensile force in section AA. The tendons do not exceed their prestressed forces. Then,

$$T = 37 \times 2 \times 1090 = 8.06 \times 10^4 \text{ kips}$$

d_1 and d_2 are the distances from the neutral axis to the centers of gravity of the active tendons on the tension and compression sides of the neutral axis respectively as shown in Figure 3A-29(b).

Therefore:

$$m_2 = T(d_1 + d_2) = 8.06 \times 10^4 \times 987 = 7.95 \times 10^7 \text{ (k-in)}$$

It is now concluded that even with a dynamic load factor 1.2 and a very conservative assumption that nine tendons are covered by the airplane impact and are destroyed, the safety factor for not causing tension at section AA is:

$$S.F. = \frac{m_2}{m_1} = \frac{7.95 \times 10^7}{4.2 \times 10^7} = 1.90$$

In order to shear-off nine vertical tendon anchorages, the aircraft must impact as shown in Figure 3A-26. A reasonable assumption is that the impact load will be concentrated above the ring girder. As a conservative estimate of the resulting forces, the forces shown on Figure 3A-18 due to the aircraft impact at the girder to dome transition were used. Figure 3A-18 shows that the maximum moments and shears occur in the dome. Therefore, the resisting prestress forces in the critical dome area are not affected by a loss of vertical prestress.

The shear stresses in the wall are of concern because the allowable shear stress reduces when the meridional axial force is lost due to the failure of nine vertical tendon anchorages. The shear stresses at three locations were determined and compared with an ultimate shear stress of $2f'_c$ (ACI 318-63).

The three locations are:

- a. Cylinder wall to ring girder transition

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- b. Base of wall
- c. Ten feet above the base of the wall
(Haunch to typical wall transition)

Although the shear stress at location a. exceeds $2.0 \phi f_c$, the shear reinforcement required is less than that provided for the normal loading cases under no loss of prestress. The shear stresses at locations b. and c. are less than the ultimate shear stress.

4.2.2 CASE B: SHEAR-OFF THE ANCHORS OF DOME TENDONS

If the aircraft impact occurs as shown in Direction 2 of Figure 3A-30, the shear resisting capacity of the concrete at Section AA has to be greater than the impact force so that no force will be transferred to the anchors of the roof tendons. The shear resisting area, assuming a 19 ft width of impact area is:

$$A = 19 \text{ ft} \times 6.416 \text{ ft} \times 144 = 17,554 \text{ in}^2.$$

The total shear capacity of concrete against the vertical impact is:

$$F = 600 \times 17,554 = 10.53 \times 10^6 \text{ lb}$$

which is smaller than the aircraft impact load $21.1 \times 10^6 \text{ lb}$ (with a conservative assumed dynamic load factor 1.2).

It is reasonable to assume that the applied load is proportional to the area of impact. $\frac{1.25 \times 19}{15 \times 19} 21.1 \times 10^6 \text{ lbs.}$

The applied load = $1.76 \times 10^6 \text{ lbs}$ which results in a shearing stress across section A-A of Figure 3A-30 = $\frac{1.76 \times 10^6 \text{ lbs}}{19 (6.416) (144)} = 100 \text{ psi} < 600 \text{ psi.}$

Under the above analysis, the dome tendon anchorages will not fail in shear.

The dome tendons are composed of three layers lying on top of each other. Each layer is composed of parallel and equally spaced tendons (Figure 3A-31). As can be seen from Figure 3A-32, the orientation of the three layers is such that the tendons cross each other with a constant angle of 60 degrees. Since the dome tendons are so close to each other, crossing each other, and overlying each other is intuitively believed that no damage will occur to the dome even if a few tendons are assumed to be broken under the aircraft impact.

In addition to the previous logic, providing the failure of the dome tendon anchorages is caused by an aircraft impact as shown in Direction 2 of Figure 3A-30, the major portion of the impact load will be resisted by the cylindrical wall and not by the dome which has lost the prestress forces due to 8 broken dome tendons.

If the aircraft travels in the Direction 3 as shown in Figure 3A-30, it is seen that the concrete bearing failure is relatively small. Previously, a calculation has shown that as long as the actual impact area is at least 14.5 percent of the maximum theoretical impact area, no bearing failure will occur. Therefore, impact Direction 3 is of no critical concern.

4.2.3 CASE C: SHEARING-OFF THE HOOP TENDONS

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The aircraft may travel in the direction shown in Figure 3A-33. Since there is no concrete cover to protect the anchor, direct impact on the anchors may shear-off several tendons. As shown in Figure 3A-33 the minimum vertical spacing between anchors on one side of the buttress is 33 inches. Each hoop tendon is anchored in one buttress, then passes by the adjacent buttress, and is finally anchored in the next adjacent buttress. The aircraft impact area has a minimum depth of 19 ft which can cover seven (i.e., $\frac{19 \times 12}{33} = 6.9$) anchors of hoop tendons.

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If the most conservative assumption is made that all of the seven anchors are sheared-off due to aircraft impact, one-half of the hoop prestress force in a cylindrical panel with a depth of 19 ft and a curve length of one third of the cylinder periphery would be eliminated. An analysis is made considering the containment vessel with normal prestress conditions with the exception that a cylindrical segment with a depth of 15 ft has only one half of the hoop prestress. The resultant axial forces and shears with half of the hoop tendon forces lost in the range between 800 inches and 965 inches above the base of the cylindrical wall are listed in Table 3A-5. It is seen that the out of plane shear and hoop force have changed due to the failure of the hoop tendons. The shear has increased considerably, but, is in the opposite direction of the shear due to aircraft impact, and therefore aids in resisting the aircraft impact. When the aircraft impact diminishes and the hoop tendons are still broken, the remaining shear is much less than the ultimate shear stress according to ACI 318-63.

The above analysis does not consider the moments caused by the loss of 8 hoop tendons. Referring to Figure 3A-34, which is a comparison of (1) the loading due to aircraft impact plus total prestress and (2) the loading due to aircraft impact plus total prestress minus the hoop prestress due to the loss of hoop tendons, it can be deduced that the resulting moments will tend to counteract the moments caused by the aircraft impact. When the aircraft impact diminishes, the axial compression due to the remaining prestress forces (see Table 3A-5) should be sufficient to overcome the tension due to the moments caused by the loss of hoop prestress.

Based on the above logic and conservative numerical calculation, it is believed that the aircraft impact in the direction shown on Figure 3A-33 does not jeopardize the stability of the structure.

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4.3 SPALLING DUE TO AIRCRAFT IMPACT ON THE OUTSIDE WALL

Spalling is the kind of fracture which results from the interference between the incident compressional wave and its reflected counter-part. The mechanics of spalling can be described graphically as follows:

Compressive wave before
reaching the free surface

Part of the compressive wave
reflected back from the free
surface

The spalling will occur at a distance Y from the free surface, if the net tensile stress AB exceeds the tensile fracture strength of the material (References 23 and 24). From these we can see that the spalling is influenced by two factors, (a) the duration and shape of the stress pulse, and (b) property of the material acted on.

The plane dilatational wave will travel through the plate at the speed of $\sqrt{\frac{S + 2T}{\rho}}$, where ρ is the density of the material, S and T are Lamé's constants and Y is the

Transform $(S + 2T)$ to a form in terms of Young's Modulus E, and Poisson's ratio U, which engineers are more familiar with. We have:

$$\frac{S + 2T}{(1 + U)(1 - 2U)} = \frac{E(1 - U)}{(1 + U)(1 - 2U)}$$

For concrete, with E = 4,000 ksi, U = 0.15, $\rho = 145 \text{ lb/ft}^3$, the wave velocity is:
 $c = 1.4 \times 10^5 \text{ in./sec} = 11,650 \text{ ft/sec}$.

Hence, for the structure we are concerned with, the spalling may occur only for the stress pulse duration of the order of microseconds. The first unloading of our impact force occurs at 0.19 seconds. In other words, for a wall of 5 ft thickness, the wave has to travel back and forth almost 220 times before the impact force reaches the unloading point. Due to the internal friction, the stress wave is long dispersed before it can build up tensile stress in the structure. So the spalling effect is almost impossible to occur.

Nevertheless, the anchors on the containment vessel liner above grade will be deeply anchored into the concrete wall with one inch diameter bolts (form ties). These bolts have a capacity to resist 1.7 kips per foot of anchor. This measure further protects the liner anchors against failure due to spalling; even though such behavior is not anticipated, as previously described.

The design criteria for the liner anchors is described in Section 5.2.3.2.5 "Liner Anchor". The liner angle welds have been tested by 20 percent liquid penetrant test 100 visual inspection in accordance with the requirements of Section VIII of ASME Boiler and Pressure Vessel Code. Three specimens of the liner angle welds were tested by Pittsburgh Testing Laboratory and the factor of safety was found to be 2.7 against the worst possible load on the liner anchor.

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4.4 IMPACT EFFECTS ON EQUIPMENT AND COMPONENTS

To eliminate the shock effect caused by an aircraft impact, the concrete floor slabs in the Control Building have been separated by a 2 inch wide joint from the exterior walls exposed to an aircraft impact. The concrete slabs are supported by steel beams which in turn are supported on elastomeric pads which act as vibration dampeners.

4.5 REFERENCES

1. Haley, Jr., J. and Turnbow, J., "Total Reaction Force due to an Aircraft Impact into a Rigid Barrier," AvSER Report prepared for Gilbert Associates, Inc. by Dynamic Science, Phoenix, Arizona, April 1968.
2. Biggs, J. M., "Introduction to Structural Dynamics," McGraw-Hill Book Co., N.Y. 1964, Section 2.3.
3. J. Martin, "Impulsive Loading Theorem For Rigid-Plastic Continua," J. Engineering Mechs. Div., ASCE, EM5, October 1964, pp. 27-42.
4. K. Forsberg and W. Flugge, "Point Load On a Shallow Elliptic Parabaloid," To Appear in Journal of Applied Mechanics.
5. A. Kalnins and P.M. Naghdi, "Propagation of Axisymmetric Waves in An Unlimited Elastic Shell," Journal of Applied Mechanics, 27, 1960, pp 690-695.
6. C. H. Norris 'et al': "Structural Design for Dynamic Loads," McGraw Hill, 1959.
7. J. N. Cernica and M. J. Charignon: "Ultimate Static and Impulse Loading of Reinforced Concrete Beams." ACI Journal, V. 50, No. 9, 1963, pp. 1219-1228
8. Walter Cowell: "Dynamic Properties of Plain Portland Cement Concrete." Naval Civil Engin. Lab., Port Huenema, Calif. Report NCEL 447, 1966.
9. H. Weigler and G. Becker: "Der Bauingenieur," Berlin, Vol. 36, No. 10, 1961, pp. 390-396.
10. J. Peter: "Zur Bewehrung von Scheiben und Schalen fur Hauptspannungen Schiefwinklig zur Bewehrungstrichtung," Doctoral Thesis, Technische Hochschule Stuttgart, 1964.
11. G. W. D. Vile: "Strength of Concrete Under Short-Time Static Biaxial Stress." Proceedings of International Conference on Structural Concrete, London, 1965.
12. Fritz Leonhardt: "Prestressed Concrete Design and Construction" Wilhelm Ernst & Sohn, 1964, pp. 59.
13. Volterra, E. and Zachmanoglou, E., "Dynamics of Vibrations," Charles Merrill Books, Inc., Columbus, Ohio, 1965.

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14. Timoshenko, S. and Young, D., "Vibration Problems in Engineering," 3rd Ed., D. Van Nostrand Co., Inc., 1955.
15. Harris, C. and Crede, C., "Shock and Vibration Handbook," McGraw-Hill Book Co., N.Y., 1961.
16. Bogner, F., Fox, R., and Schmit, L., "The Generation of Interelement-Compatible Stiffness and Mass Matrices by the Use of Interpolation Formulae," Proc. Conf on Matrix Methods in Structural Mech., Dayton, Ohio, 1965.
17. Gallagher, R., "The Development and Evaluation of Matrix Methods for Thin Shell Structural Analysis," Ph.D. Dissertation, State University of New York at Buffalo, 1966.
18. Gallagher, R. and Yang H., "Elastic Instability Prediction of Doubly Curved Shell Structures," 2nd conference on Matrix Methods in Structural Mechanics, Wright-Patterson Air Force Base, Dayton, Ohio, Oct. 1968.
19. Yang, H., "A Finite-Element Formulation for the Instability Prediction of Doubly Curved Shell Structures," Ph.D. Dissertation, Cornell University, Ithaca, N.Y., Oct. 1968.
20. Timoshenko, S. and Goodier, J. N., "Theory of Elasticity," McGraw-Hill Book Co., 1951.
21. Riera, J. D., "On the Stress Analysis of Structures Subjected to Aircraft Impact Forces", Nuclear Engineering and Design, August, 1968.
22. Gwaltney, R. C., "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants" U.S. Atomic Energy Commission, DRNL-NSIC-22, September, 1968.
23. Rinehart, John S. and Pearson, John, "Behavior of Metals under Impulsive Loads" Dover, 1965.
24. Kolsky, H. "Stress Waves," Dover, 1963.
25. Wood, R. H. "Plastic and Elastic Design of Slabs and Plates," Ronald, 1961.
26. Rosenthal, I. and Glucklich, J., "Strength of Plain Concrete under Biaxial Stress," ACI Journal, November 1970.

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TABLE 3A-1

TIME VARIABLE t_n

Boeing	t_1	t_2	t_3	t_4	t_5	t_6	I
	0.06	0.14	0.19	0.24	0.26	0.33	0.219

TMI-1 DSAR

TABLE 3A-2

DYNAMIC LOAD FACTORS (DLF)

$$DLF_1 = \left\{ \frac{t}{t_1} - \frac{T}{2X t_1} \sin at \right\} I$$

(0 to t_1)

$$DLF_2 = \left\{ 1 + \frac{T}{2X t_1} [\sin a(t-t_1) - \sin at] \right\} I$$

(t_1 to t_2)

$$DLF_3 = DLF_2 + \left\{ \frac{2(t-t_2)}{t_p} - \frac{T}{X t_p} \sin a(t-t_2) \right\} (1-I)$$

(t_2 to t_3)

$$DLF_4 = DLF_2 + \left\{ 1 - \frac{2t-t_3}{t_p} + \frac{T}{X t_p} [2 \sin a(t-t_3) - \sin a(t-t_2)] \right\} (1-I)$$

(t_3 to t_4)

$$DLF_5 = DLF_2 + \frac{T}{X t_p} \{- \sin a(t-t_4) + 2 \sin a(t-t_3) - \sin a(t-t_2)\} (1-I)$$

(t_4 to t_5)

$$DLF_6 = DLF_5 - \left\{ \frac{t-t_5}{t_6-t_5} - \frac{T}{2X(t_6-t_5)} \sin a(t-t_5) \right\} I$$

(t_5 to t_6)

$$DLF_7 = DLF_5 - \left\{ 1 + \frac{T}{2X(t_6-t_5)} [\sin a(t-t_6) - \sin a(t-t_5)] \right\} I$$

(after t_6)

where:

$$a = \frac{2X}{T} \text{ (radians/sec)}$$

TMI-1 DSAR

TABLE 3A-3

KINETIC ENERGY OF THE DOME

$$r_e = 37.0'$$

<u>S</u>	<u>v₈</u> <u>(fps)</u>	<u>T₈</u> <u>(ft-lbs)</u>	
Case 1	0.1667	5.8	0.182 x 10 ⁶
Case 2	0.172	5.6	0.172 x 10 ⁶
Case 3	0.228	4.3	0.133 x 10 ⁶
Case 4	0.130	7.4	0.23 x 10 ⁶

TMI-1 DSAR

TABLE 3A-4

UPPER BOUND DISPLACEMENTS

$$r_e = 37.0'$$

<u>Case</u>	$W_8^{U.B.}$ <u>(inches)</u>
1	0.995
2	0.925
3	0.715
4	1.245
Average	0.97

TMI-1 DSAR

TABLE 3A-5

COMPARISON OF STRESS RESULTANTS FOR PRESTRESS LOADINGS

E L E V.*	OUT OF PLANE SHEAR		MERIDIONAL FORCE		HOOP FORCE	
	Q(10 ²)		Nf (10 ⁵)		NP (10 ⁵)	
	lbs/in		lbs/in		lbs/in	
	(1)	(2)	(1)	(2)	(1)	(2)
800	-0.33	18.508	-.3730	-.3730	-.7166	-.5628
816	-0.32	14.446	-.3730	-.3730		-.5519
833	-0.309	10.589	-.3730	-.3730	-.7157	-.5429
849		6.892	-.3730	-.3730		-.5362
866		3.308	-.3730	-.3730		-.5320
882	-0.219	-0.22	-.3730	-.3730	-.7151	-.5305
899		-3.74	-.3730	-.3730		-.5317
915		-7.314	-.3730	-.3730		-.5355
932		-10.99	-.3730	-.3730		-.541
948		-14.83	-.3730	-.3730		-.5505
965	-0.010	-18.86	-.3730	-.3730	-.7150	-.5610

* INCHES ABOVE THE BASE CYLINDRICAL WALL

(1) WITHOUT LOSING TENDON

(2) ½ OF HOOP TENDON FORCES LOST

TMI-1 DSAR

TABLE 3A-6
Reaction Load (R) Calculations
Case 1: With Wing and Engines Detached
(C = 200 kts = 338 fps; M = 200,000 lb. = 6200 slugs)

TIME	STRENGTH OF FRAME	MASS /FT	MASS CRUSHED	MASS LOST	MASS INTACT
0.0	0.0	0.	0.	0.	6200.
0.010	0.154	5.	9.	0.	6191.
0.020	0.307	9.	23.	0.	6168.
0.030	0.459	12.	35.	0.	6132.
0.040	0.565	18.	49.	0.	6083.
0.050	0.565	22.	69.	0.	6014.
0.060	0.565	25.	81.	0.	5933.
0.070	0.565	28.	89.	0.	5844.
0.080	0.565	28.	94.	0.	5750.
0.090	0.565	28.	93.	0.	5658.
0.100	0.565	28.	92.	0.	5566.
0.110	0.565	28.	91.	0.	5474.
0.120	0.565	28.	91.	0.	5383.
0.130	0.565	28.	90.	0.	5293.
0.140	0.629	28.	89.	0.	5204.
0.150	0.817	47.	109.	0.	5095.
0.160	1.007	62.	187.	0.	4909.
0.170	1.193	86.	236.	336.*	4336.
0.180	1.378	112.	319.	0.	4017.
0.190	1.550	156.	401.	962.**	2654.
0.200	1.550	167.	501.	0.	2153.
0.210	1.550	157.	485.	0.	1669.
0.220	1.550	89.	405.	0.	1264.
0.230	1.550	33.	128.	0.	1136.
0.240	1.550	32.	88.	0.	1048.
0.250	1.550	31.	79.	0.	969.
0.260	1.550	30.	72.	0.	897.
0.270	1.550	29.	64.	0.	832.
0.280	1.550	28.	57.	0.	776.
0.290	1.495	27.	50.	0.	726.
0.300	1.438	27.	43.	0.	683.
0.310	1.389	26.	36.	0.	646.
0.320	1.347	25.	30.	0.	616.
0.330	1.313	25.	24.	0.	592.
0.340	1.287	25.	18.	0.	574.
0.350	1.269	24.	12.	0.	561.
0.360	1.259	24.	7.	0.	555.
0.367	1.257	24.	2.	0.	553.

MAXIMUM REACTION LOAD = 17.213 LBS*10.0E 6

* Inboard engines stopped without significant deformation at barrier.

** Outboard engines, wing structure, and outboard fuel are separated from wing.

TMI-1 DSAR

TABLE 3A-6
Reaction Load (R) Calculations
Case 1: With Wing and Engines Detached
(C = 200 kts = 338 fps; M = 200,000 lb. = 6200 slugs)

TIME	DECEL	VEL CHANGE	VELOCITY	CRUSHED LENGTH	IMPACT CORCE	DECEL G'S
0.0	0.0	0.0	338.00	0.0	0.0	0.0
0.010	24.9	0.13	337.87	3.41	0.71	0.77
0.020	49.8	0.37	337.50	6.79	1.30	1.55
0.030	74.9	0.62	336.87	10.16	1.84	2.33
0.040	92.9	0.86	336.02	13.49	2.60	2.89
0.050	93.9	0.93	335.08	16.85	3.07	2.92
0.060	95.2	0.95	334.14	20.19	3.38	2.96
0.070	96.7	0.96	333.18	23.53	3.65	3.00
0.080	98.3	0.98	332.19	26.89	3.65	3.05
0.090	99.9	0.99	331.20	30.21	3.62	3.10
0.100	101.5	1.01	330.20	33.51	3.60	3.16
0.110	103.2	1.02	329.17	36.81	3.57	3.21
0.120	105.0	1.04	328.13	40.09	3.54	3.26
0.130	106.7	1.06	327.07	43.37	3.51	3.32
0.140	120.8	1.09	325.99	46.60	3.57	3.76
0.150	160.4	1.41	324.58	49.85	5.72	4.99
0.160	205.1	1.84	322.74	53.12	7.41	6.37
0.170	275.1	2.32	320.42	56.34	10.05	8.55
0.180	343.0	3.08	317.34	59.53	12.63	10.66
0.190	584.0	4.99	312.34	62.68	16.81	18.15
0.200	719.8	6.47	305.87	65.77	17.20	22.37
0.210	928.8	8.10	297.77	68.76	15.46	28.87
0.220	1226.3	10.78	286.99	71.68	8.91	38.11
0.230	1364.3	13.07	273.92	74.49	4.05	42.40
0.240	1479.0	14.36	259.56	77.18	3.71	45.97
0.250	1600.4	15.40	244.17	79.70	3.40	49.74
0.260	1728.7	16.65	227.52	82.06	3.10	53.73
0.270	1862.5	17.96	209.56	84.25	2.82	57.89
0.280	1998.3	19.12	190.44	86.23	2.57	62.11
0.290	2060.6	20.35	170.09	88.03	2.29	64.04
0.300	2107.2	20.84	149.25	89.63	2.03	65.49
0.310	2149.7	21.29	127.96	91.02	1.81	66.82
0.320	2187.4	21.91	106.05	92.20	1.63	67.99
0.330	2218.7	22.04	84.01	93.15	1.49	68.96
0.340	2243.2	22.32	61.69	93.88	1.38	69.72
0.350	2260.4	22.30	39.39	94.38	1.31	70.26
0.360	2270.2	22.66	16.73	94.66	1.27	70.56
0.367	2272.4	16.81	-0.08	94.73	1.26	70.63

MAXIMUM REACTION LOAD = 17.213 LBS*10.0E 6

See Figure 3A-31, Reaction Load and Fuselage Deceleration.

TMI-1 DSAR

3A-7

Reaction Load (R) Calculations

Case 2: With Wing and Engines Attached

(V. = 200 kts = 338 fps; M = 200,000 lb. = 6200 slugs)

TIME	STRENGTH OF FRAME	MASS /FT	MASS CRUSHED	MASS LOST	MASS INTACT
0.0	0.0	0.	0.	0.	6200.
0.010	0.154	5.	9.	0.	6191.
0.020	0.307	9.	23.	0.	6168.
0.030	0.459	12.	35.	0.	6132.
0.040	0.565	18.	49.	0.	6083.
0.050	0.565	22.	69.	0.	6014.
0.060	0.565	25.	81.	0.	5933.
0.070	0.565	28.	89.	0.	5844.
0.080	0.565	28.	94.	0.	5750.
0.090	0.565	28.	93.	0.	5658.
0.100	0.565	28.	92.	0.	5566.
0.110	0.565	28.	91.	0.	5474.
0.120	0.565	28.	91.	0.	5383.
0.130	0.565	28.	90.	0.	5293.
0.140	0.629	28.	89.	0.	5204.
0.150	0.817	48.	110.	0.	5094.
0.160	1.007	84.	217.	0.	4877.
0.170	1.193	111.	318.	0.	4559.
0.180	1.378	134.	300.	0.	4168.
0.190	1.550	156.	458.	0.	3711.
0.200	1.550	167.	505.	0.	3206.
0.210	1.550	157.	492.	0.	2714.
0.220	1.550	141.	450.	0.	2264.
0.230	1.550	123.	391.	0.	1873.
0.240	1.550	102.	314.	0.	1559.
0.250	1.550	94.	273.	0.	1286.
0.260	1.550	82.	232.	0.	1054.
0.270	1.550	50.	161.	0.	893.
0.280	1.502	41.	104.	0.	789.
0.290	1.426	33.	79.	0.	709.
0.300	1.358	27.	58.	0.	652.
0.310	1.296	25.	44.	0.	608.
0.320	1.242	24.	37.	0.	571.
0.330	1.196	24.	31.	0.	540.
0.340	1.159	24.	26.	0.	514.
0.350	1.129	24.	20.	0.	494.
0.360	1.107	24.	15.	0.	480.
0.370	1.094	24.	9.	0.	470.
0.380	1.088	24.	4.	0.	467.
0.381	1.088	24.	0.	0.	467.

MAXIMUM REACTION LOAD = 17.580 LBS*10.0E 6

See Figure 3A-32, Reaction Load and Fuselage Deceleration.

TMI DSAR

3A-7

Reaction Load (R) Calculations

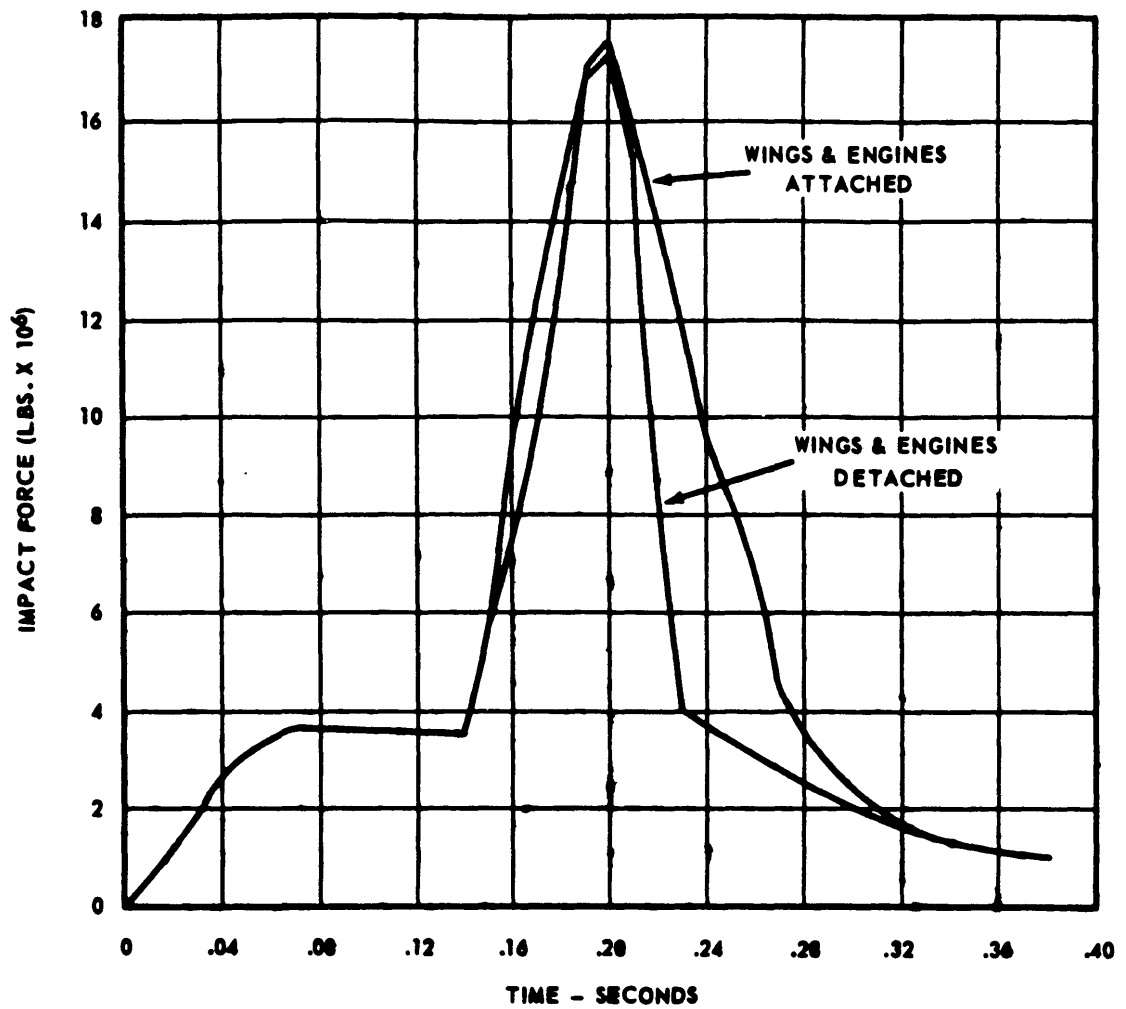
Case 2: With Wing and Engines Attached

(V. = 200 kts = 338 fps; M = 200,000 lb. = 6200 slugs)

TIME	DECEL	VEL CHANGE	VELOCITY	CRUSHED LENGTH	IMPACT CORCE	DECEL G'S
0.0	0.0	0.0	338.00	0.0	0.0	0.0
0.010	24.9	0.13	337.87	3.41	0.71	0.77
0.020	49.8	0.37	337.50	6.79	1.30	1.55
0.030	74.9	0.62	336.87	10.16	1.84	2.33
0.040	92.9	0.86	336.02	13.49	2.60	2.89
0.050	93.9	0.93	335.08	16.85	3.07	2.92
0.060	95.2	0.95	334.14	20.19	3.38	2.96
0.070	96.7	0.96	333.18	23.53	3.65	3.00
0.080	98.3	0.98	332.19	26.89	3.65	3.05
0.090	99.9	0.99	331.20	30.21	3.62	3.10
0.100	101.5	1.01	330.20	33.51	3.60	3.16
0.110	103.2	1.02	329.17	36.81	3.57	3.21
0.120	105.0	1.04	328.13	40.09	3.54	3.26
0.130	106.7	1.06	327.07	43.37	3.51	3.32
0.140	120.8	1.09	325.99	46.60	3.57	3.76
0.150	160.4	1.41	324.58	49.85	5.89	4.99
0.160	206.4	1.85	322.73	53.12	9.79	6.42
0.170	261.7	2.33	320.40	56.34	12.56	8.13
0.180	330.5	2.95	317.45	59.53	14.84	10.27
0.190	417.7	3.74	313.71	62.68	16.93	12.98
0.200	483.5	4.49	309.22	65.80	17.56	15.03
0.210	571.1	5.21	304.01	68.83	16.03	17.75
0.220	684.6	6.26	297.75	71.84	14.05	21.28
0.230	827.5	7.54	290.21	74.78	11.93	25.72
0.240	994.1	9.19	281.03	77.67	9.64	30.90
0.250	1205.2	10.96	270.06	80.43	8.39	37.46
0.260	1470.6	13.34	256.72	83.06	6.98	45.71
0.270	1736.3	16.12	240.59	85.55	4.45	53.96
0.280	1904.9	18.16	222.43	87.84	3.54	59.21
0.290	2010.6	19.61	202.83	89.97	2.80	62.49
0.300	2083.4	20.50	182.33	91.90	2.26	64.76
0.310	2133.4	21.10	161.23	93.62	1.94	66.31
0.320	2176.6	21.77	139.46	95.14	1.71	67.65
0.330	2216.2	21.97	117.49	96.42	1.53	68.88
0.340	2252.6	22.35	95.14	97.49	1.38	70.01
0.350	2283.6	22.46	72.68	98.32	1.26	70.98
0.360	2308.2	22.97	49.71	98.93	1.17	71.74
0.370	2324.4	23.17	26.54	99.31	1.11	72.24
0.380	2330.9	23.28	3.26	99.46	1.09	72.45
0.381	2331.0	3.26	-0.01	99.46	1.09	72.45

MAXIMUM REACTION LOAD = 17.580 LBS*10.0E 6

See Figure 3A-32, Reaction Load and Fuselage Deceleration.



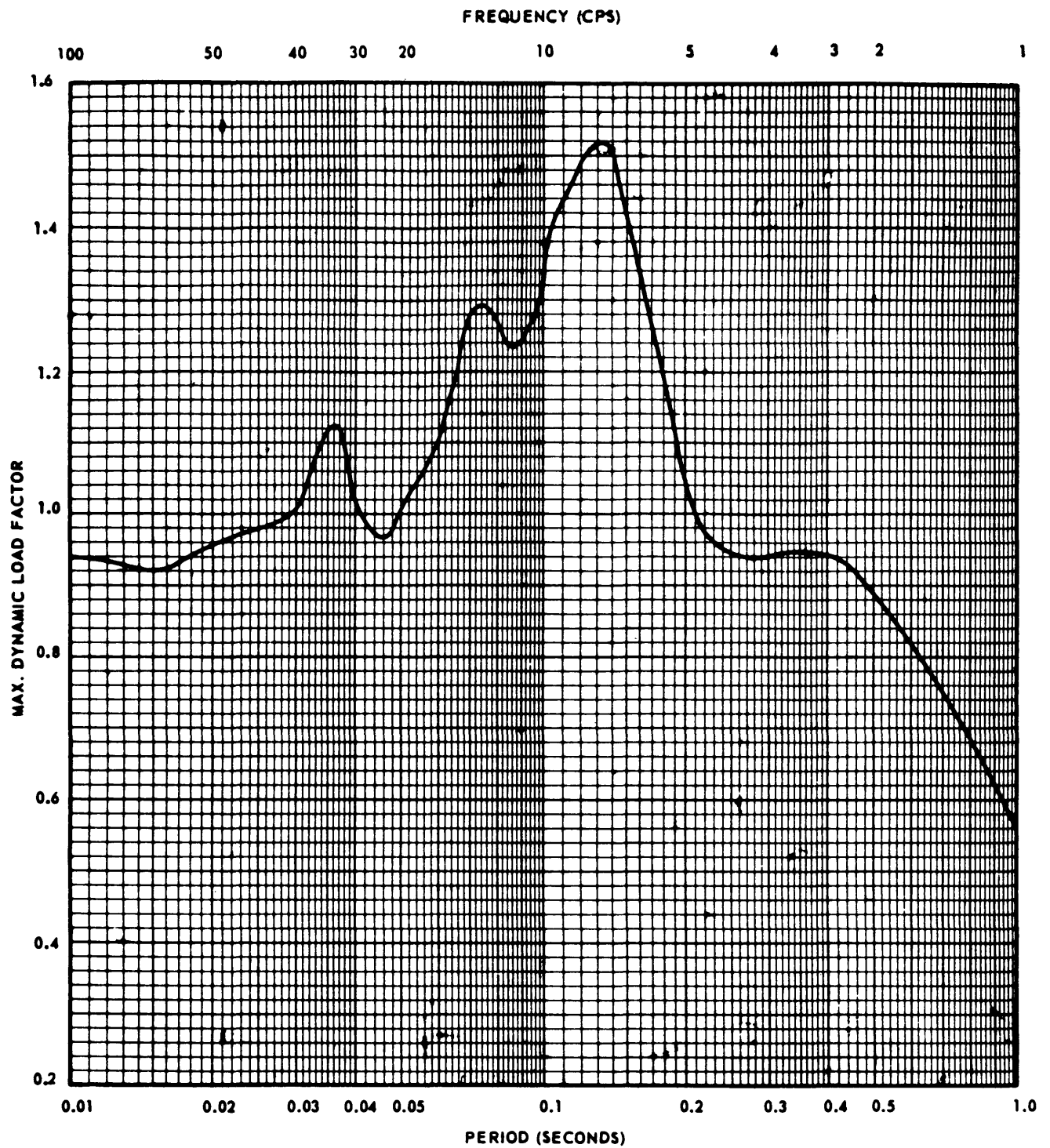
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DSAR-3A.FIG-2 REV. 2

Load Time Curve for 720 Aircraft at 200 Knots

Fig. 3A-2



DSAR-3A.FIG-3 REV. 2

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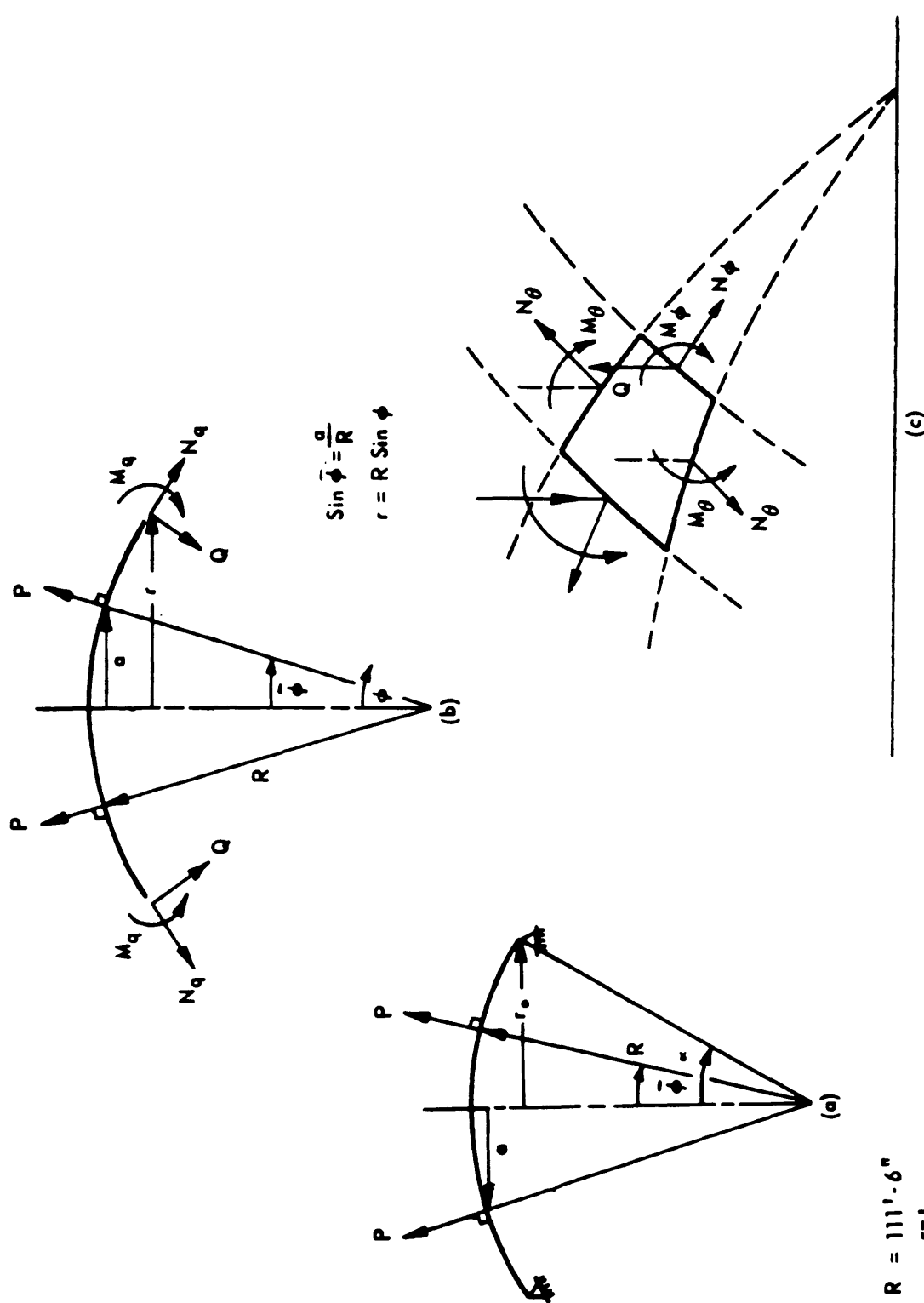
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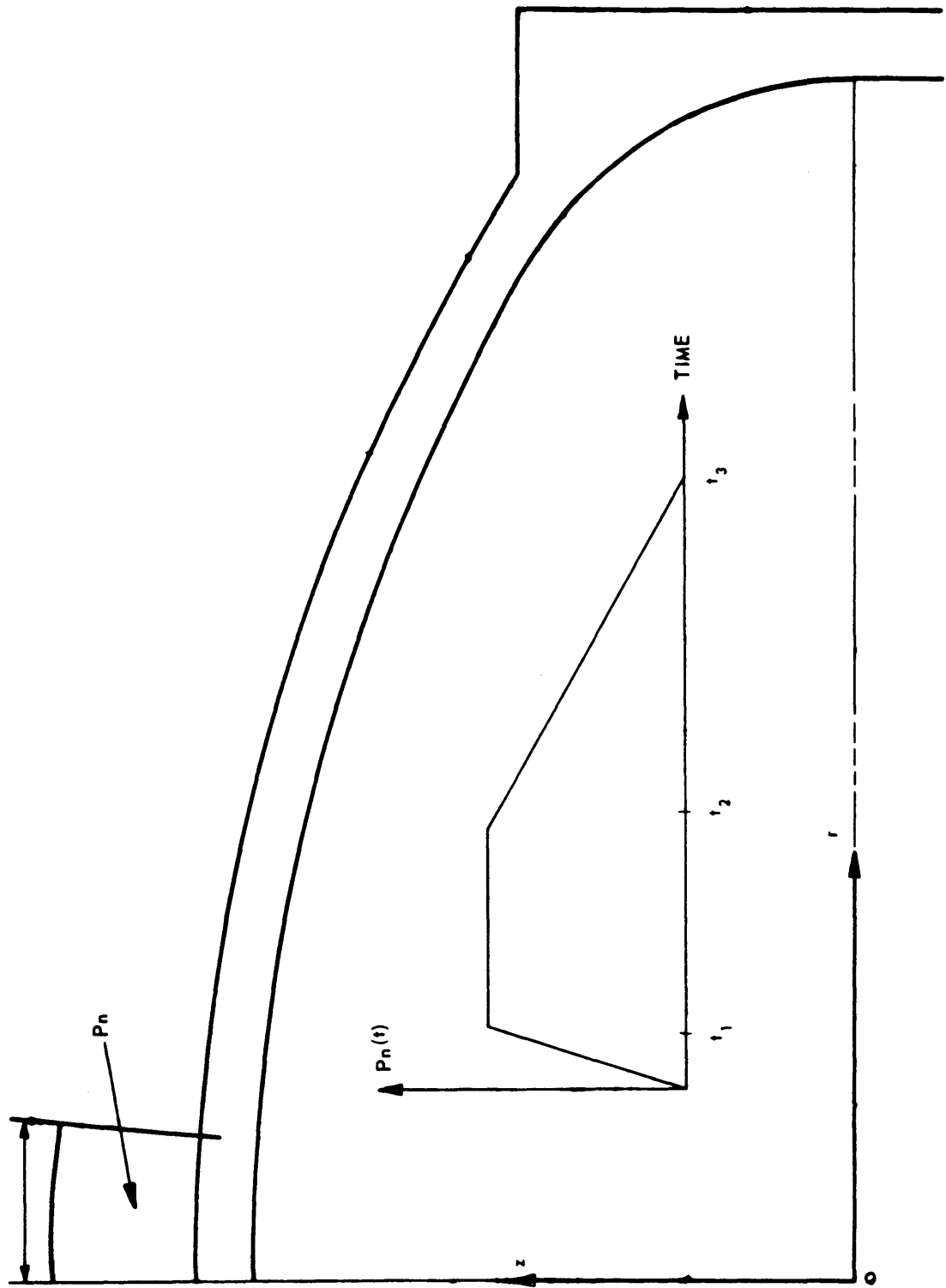
**Maximum Dynamic Load Factor vs Period or
Frequency of a One-Degree-Freedom System under
the Impact of Boeing 720**

Fig. 3A-3



NORMAL PRESSURE DISTRIBUTION

Q AREA OF IMPACT FOR LARGE AIRCRAFT



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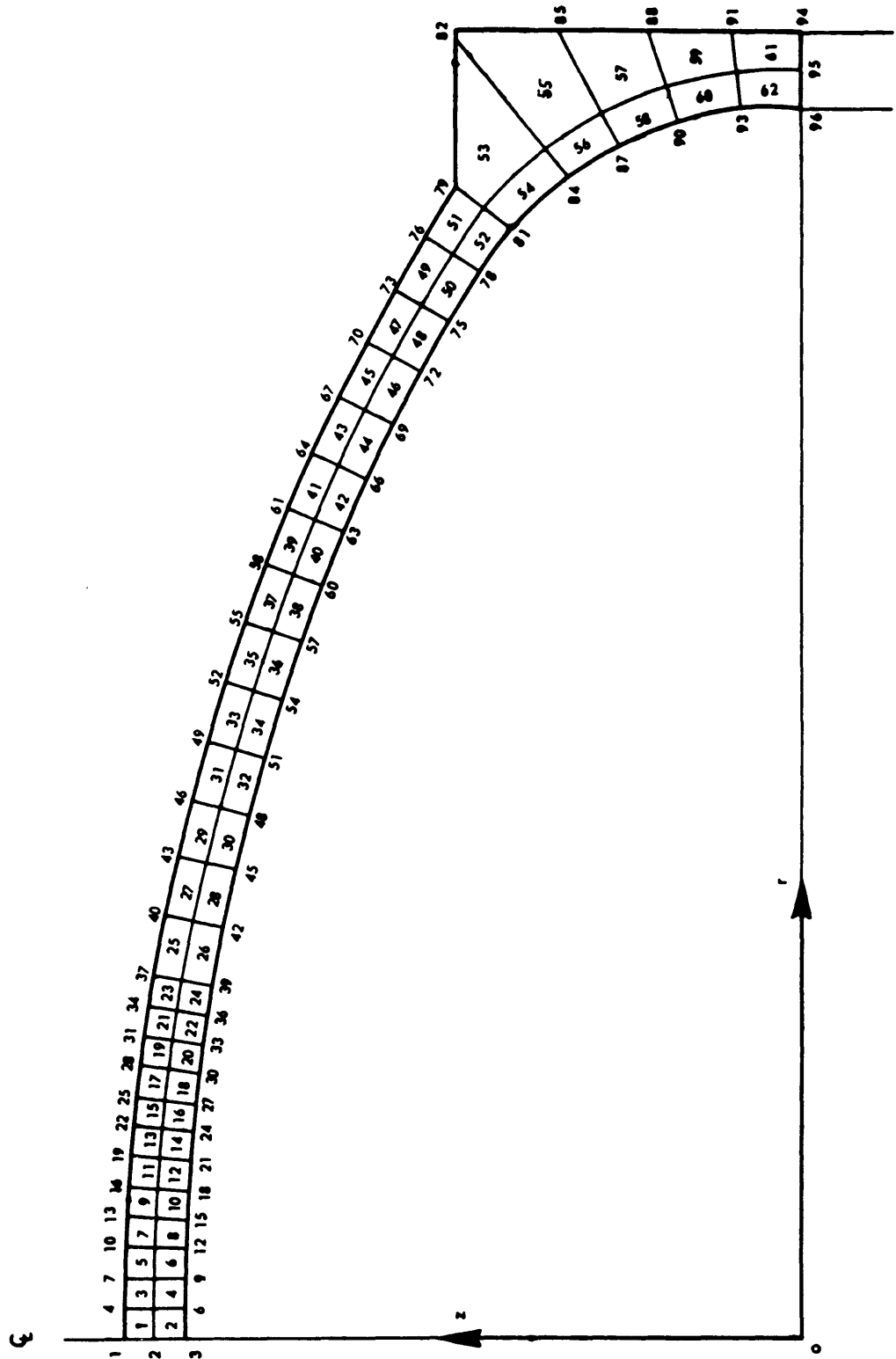
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Spacial and Time Distribution of Load on Shell

Fig. 3A-5

DSAR-3A.FIG-5 REV. 2



DSAR-3A.FIG-6 REV. 2

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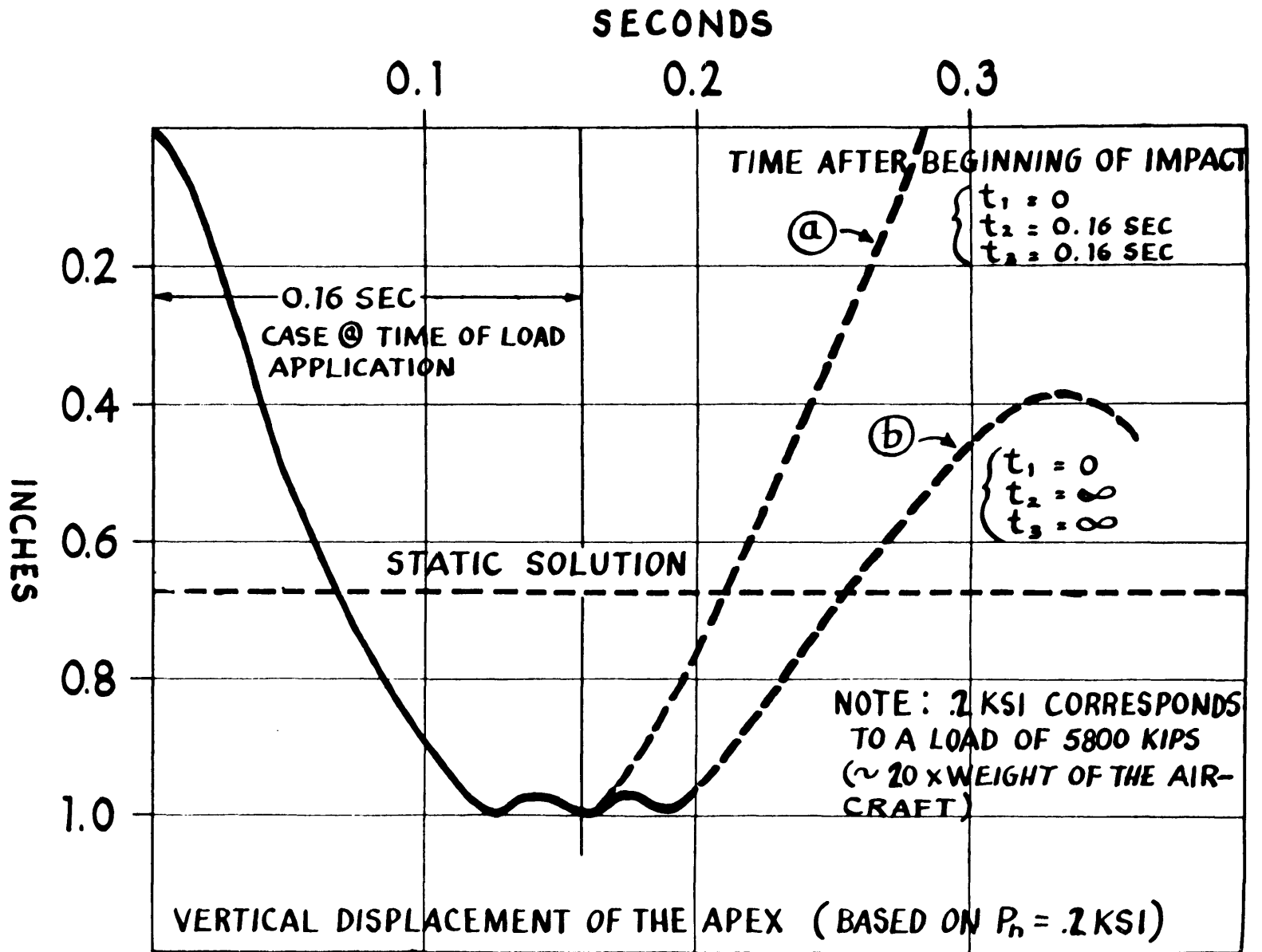
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**Grid for Dynamic Finite Element Analysis of Aircraft
Impingement on Dome**

Fig. 3A-6



DSAR-3A.FIG-7 REV. 2

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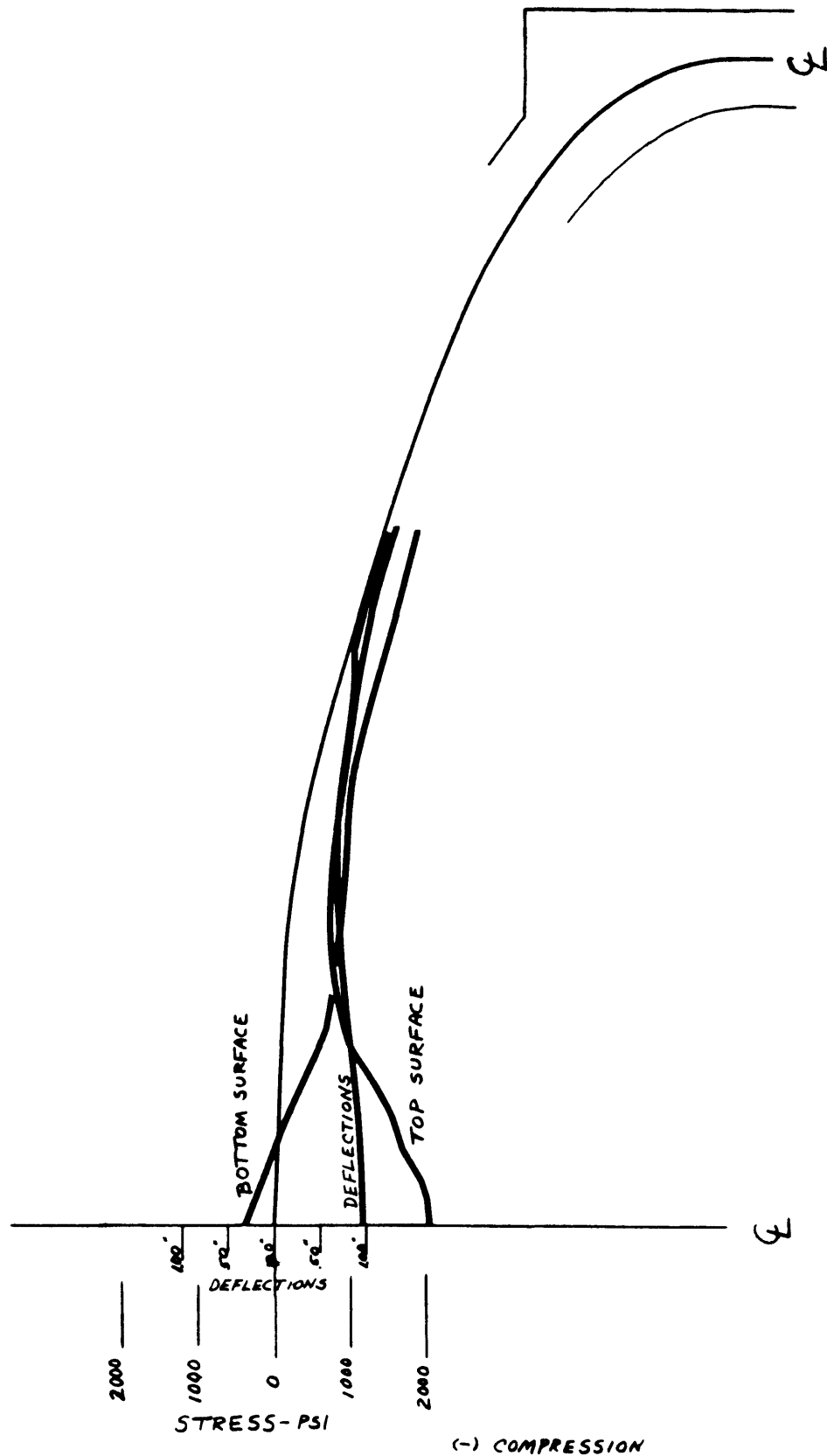
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Effect of Aircraft Impingement on Dome of
Containment Structure-Constant Deceleration

Fig. 3A-7



DSAR-3A.FIG-8 REV. 2

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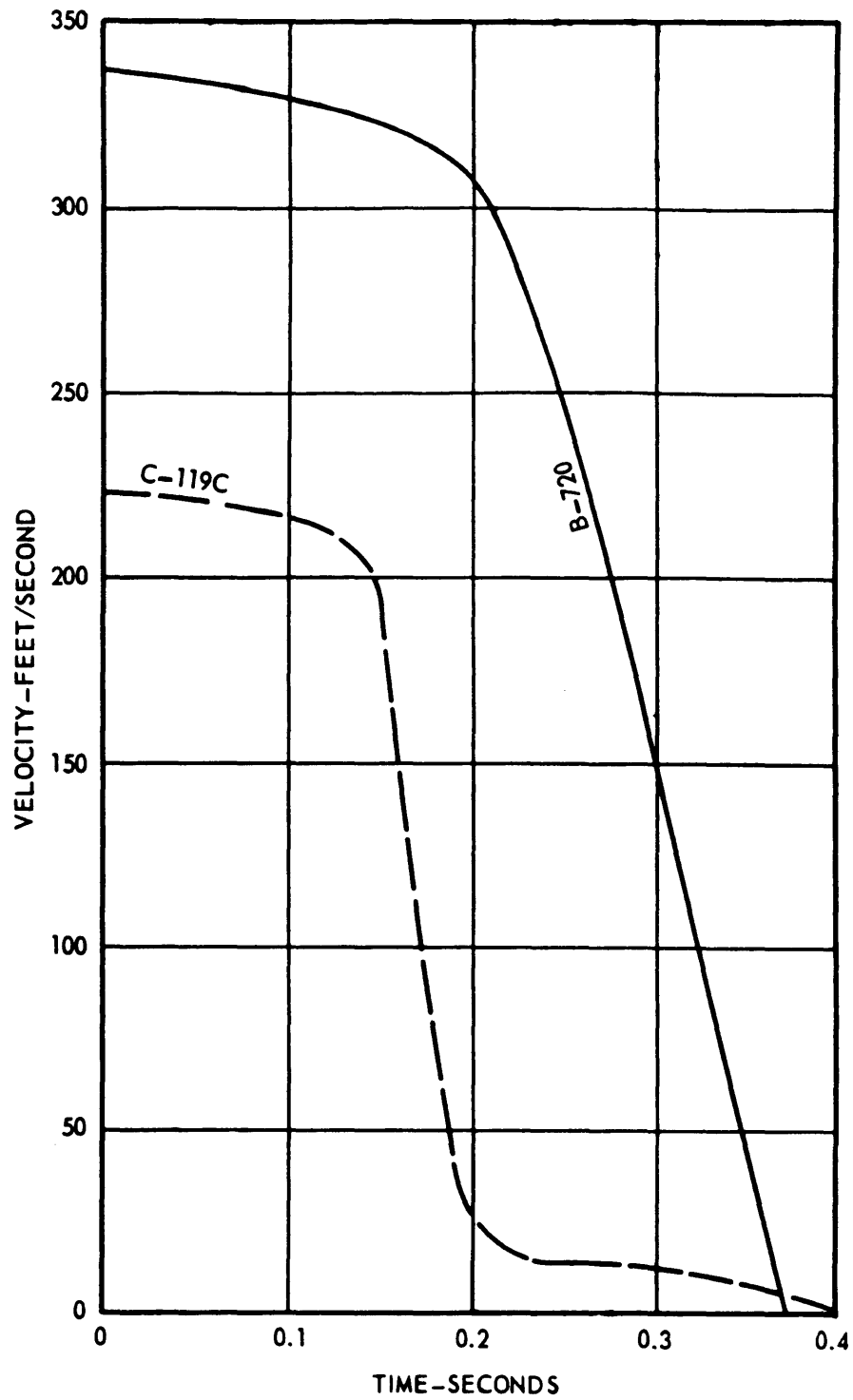
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Deflections and Stresses for Aircraft Impingement
for Time = 0.16 Seconds Constant Deceleration

Fig. 3A-8



DSAR-3A.FIG-9 REV. 2

GPU Nuclear

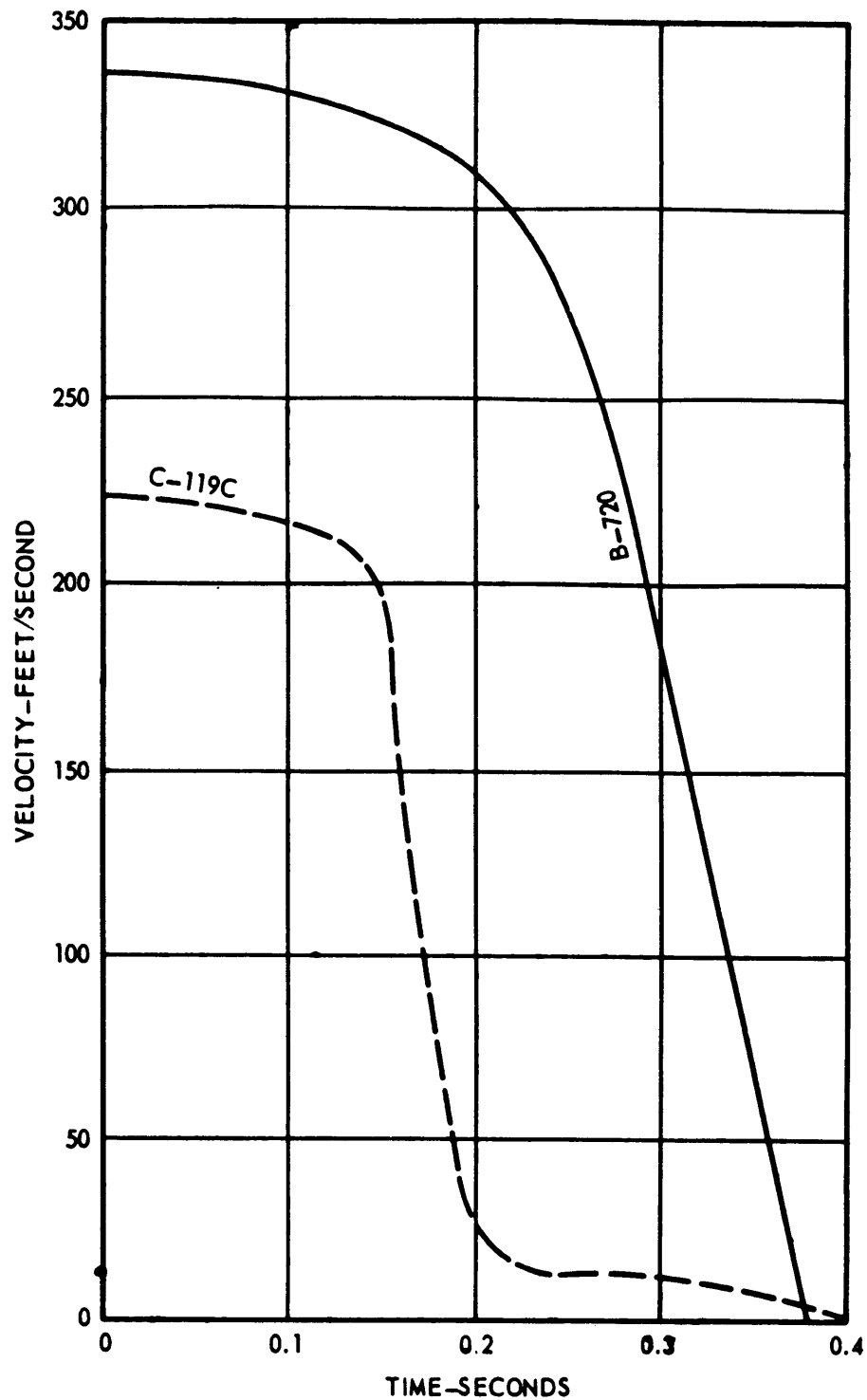
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**Velocity Diagram for 720 Aircraft at 200 Knots
Impact Speed with Wings and Engines Detached**

Fig. 3A-9



DSAR-3A.FIG-10 REV. 2

GPU Nuclear

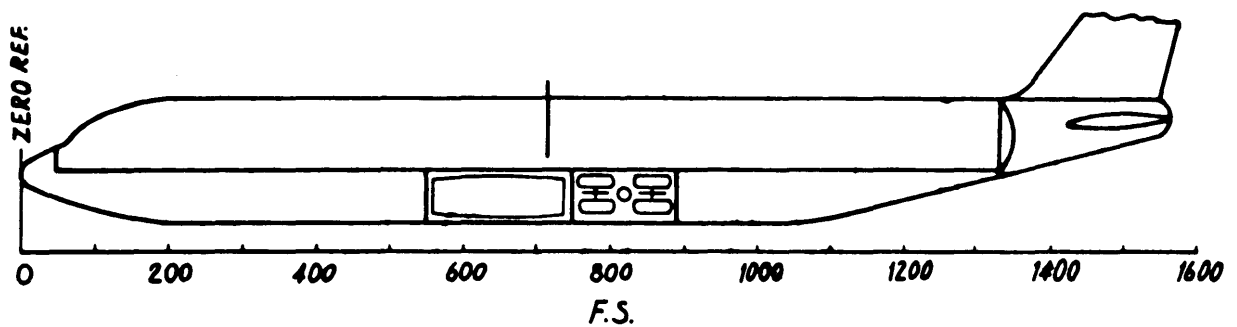
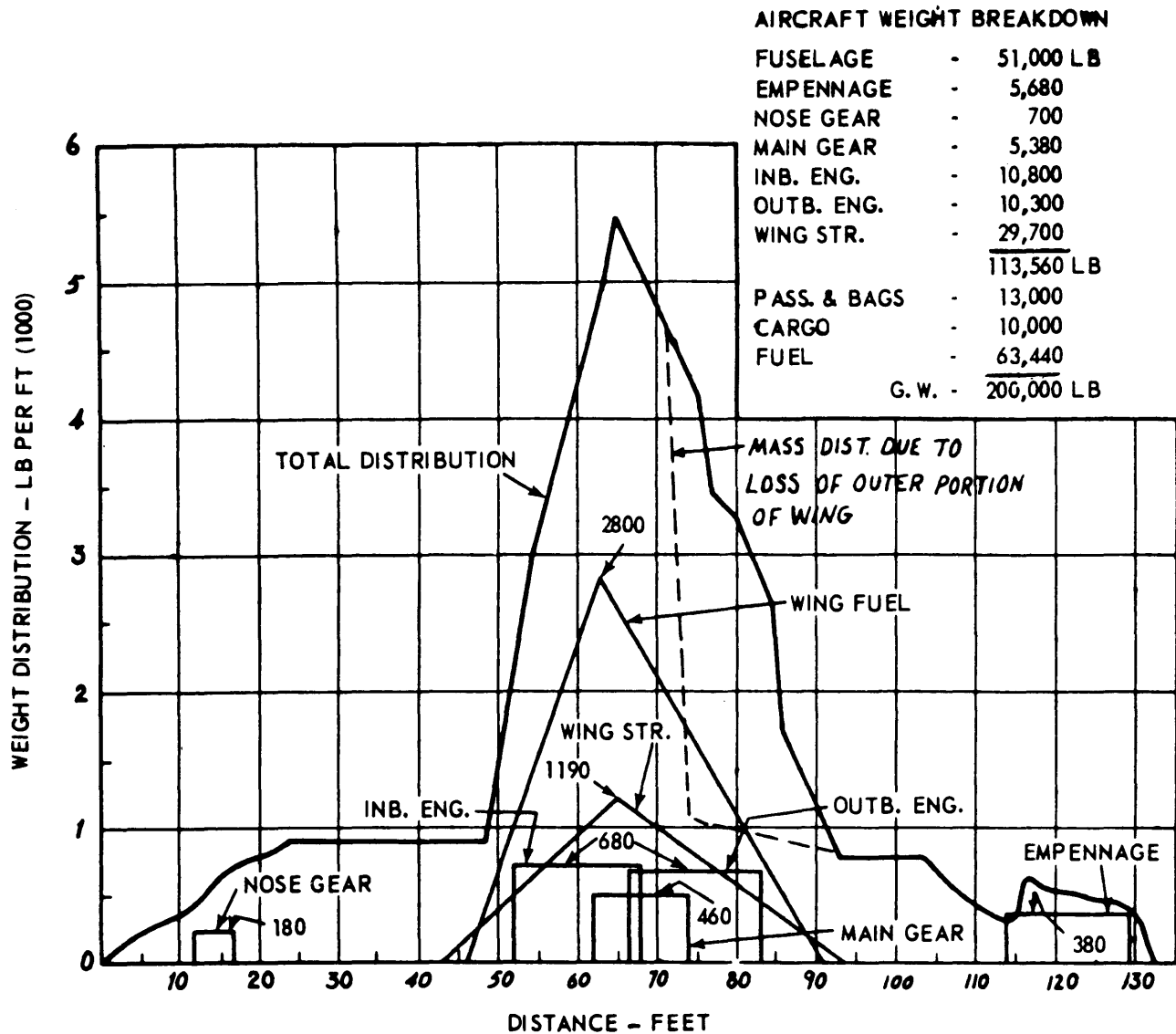
TMI Unit-1

Update - 1

7/82

**Velocity Diagram for 720 Aircraft at 200 Knots
Impact Speed with Wings and Engines Attached**

Fig. 3A-10



GPU Nuclear

TMI Unit-1

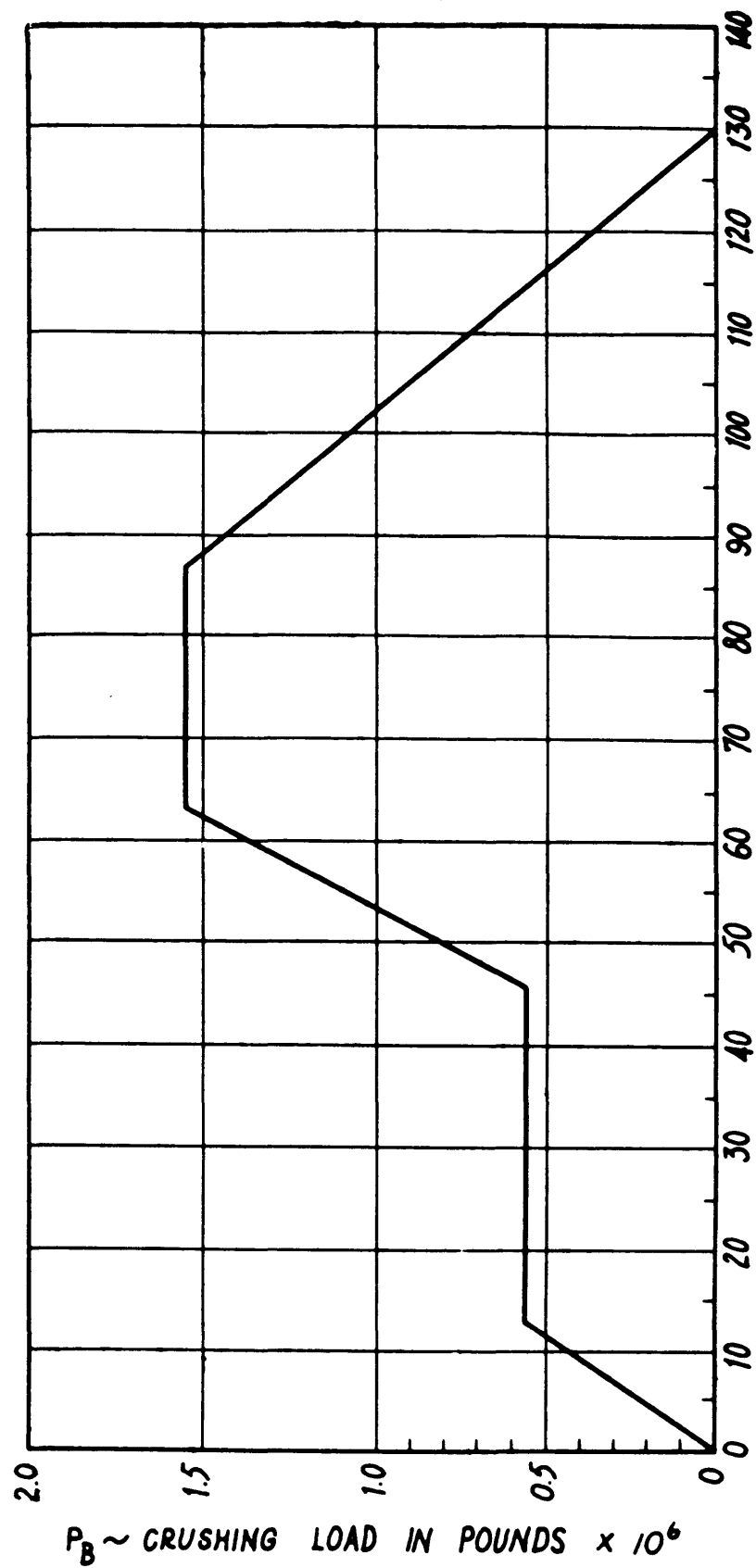
Update - 1

7/82

DSAR-3A.FIG-11 REV. 2

720 Aircraft Mass Distribution

Fig. 3A-11



DSAR-3A.FIG-12 REV. 2

GPU Nuclear

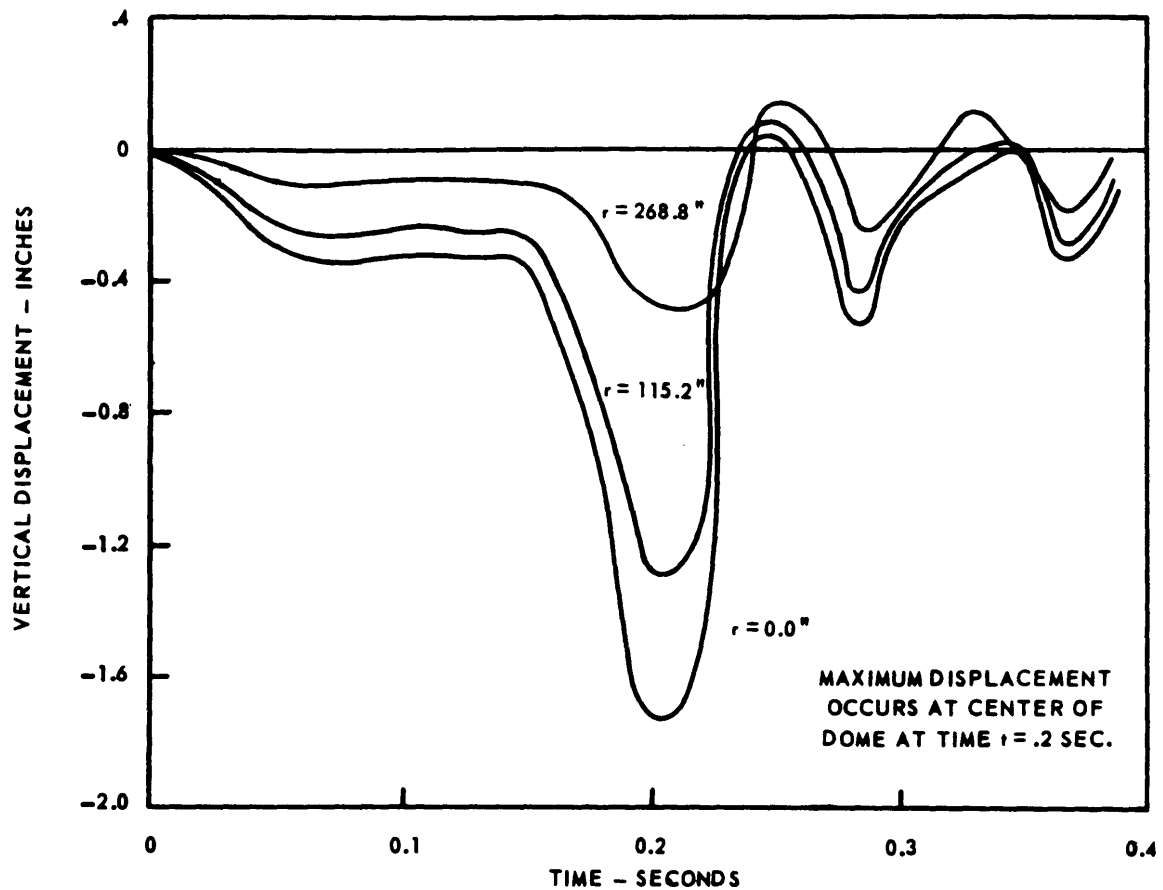
TMI Unit-1

Update - 1

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Boeing 720 Fuselage Buckling (Crushing) Load

Fig. 3A-12



DSAR-3A.FIG-13 REV. 2

GPU Nuclear

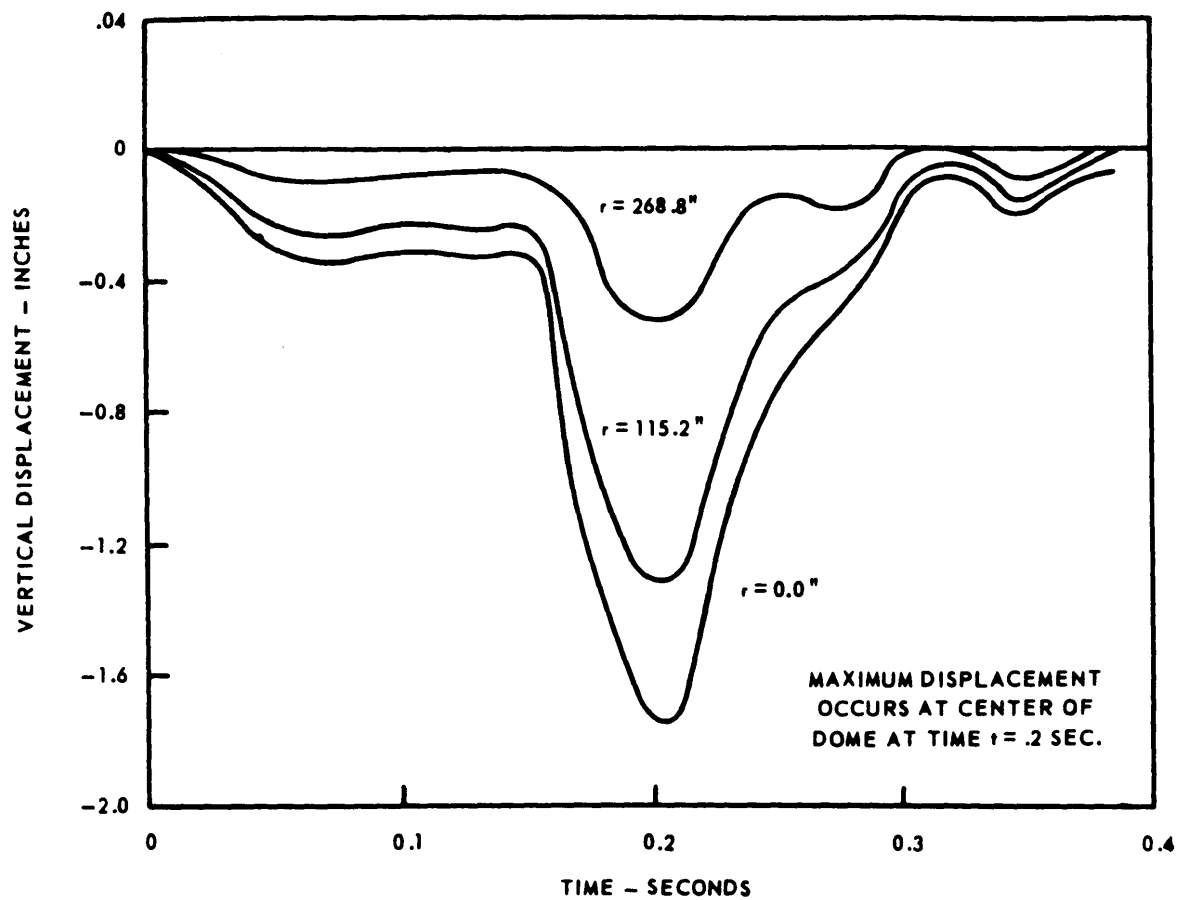
TMI Unit-1

Update - 1

7/82

**Time Variation of Shell Vertical Displacements with
Wings and Engines Detached**

Fig. 3A-13



DSAR-3A.FIG-14 REV. 2

GPU Nuclear

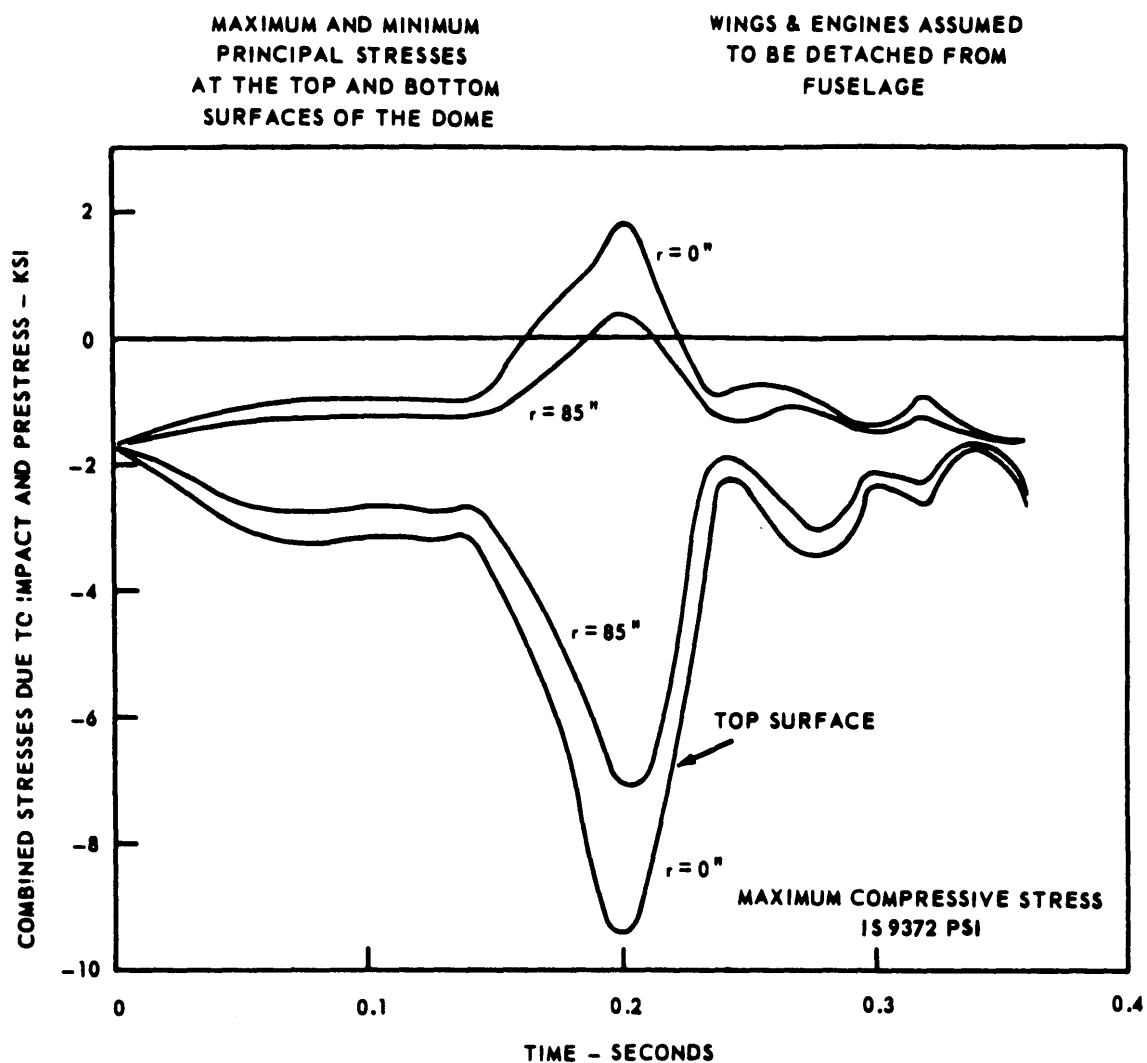
TMI Unit-1

**Time Variation of Shell Vertical Displacements with
Wings and Engines Attached**

Update - 1

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Fig. 3A-14



DSAR-3A.FIG-15 REV. 2

GPU Nuclear

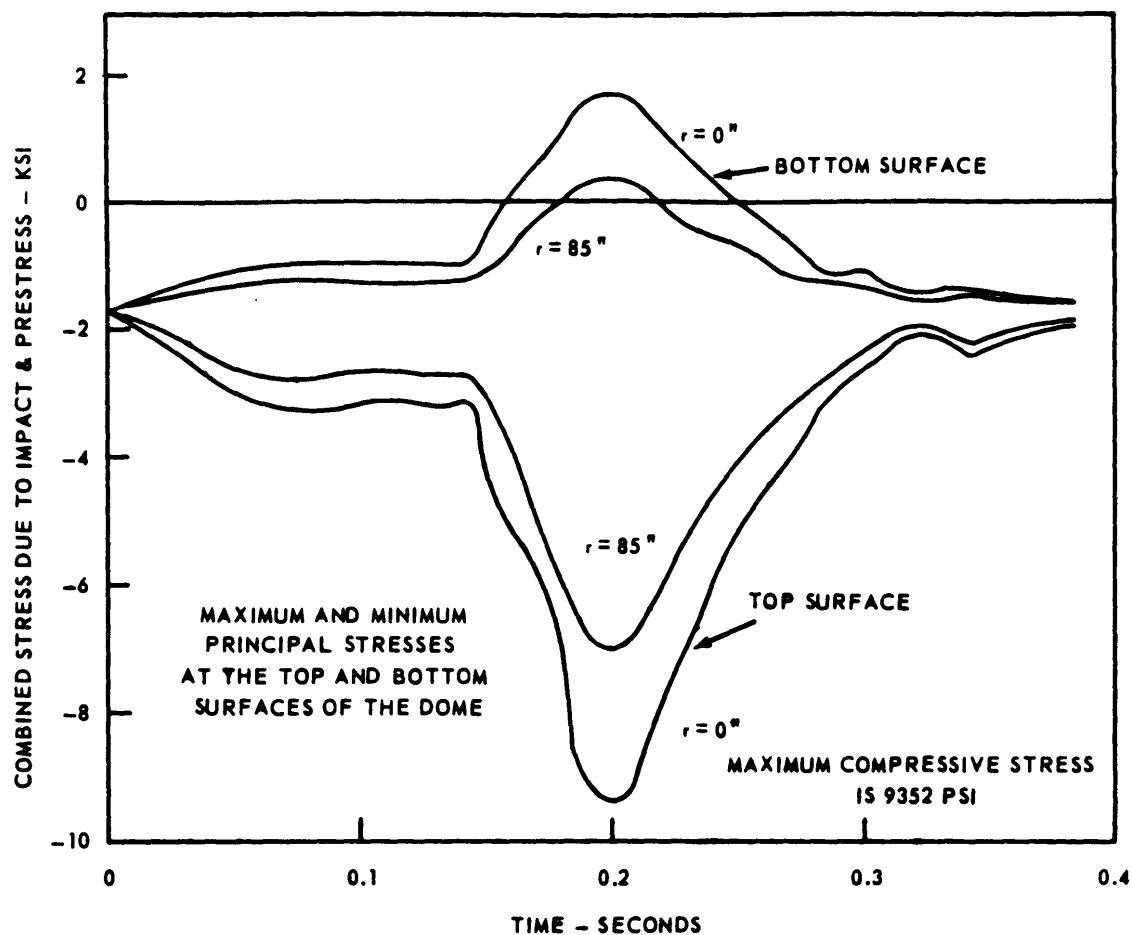
TMI Unit-1

**Time Variation of Shell Surface Stresses Aircraft
with Wings and Engines Detached**

Update -1

7/82

Fig. 3A-15



DSAR-3A.FIG-16 REV. 2

GPU Nuclear

TMI Unit-1

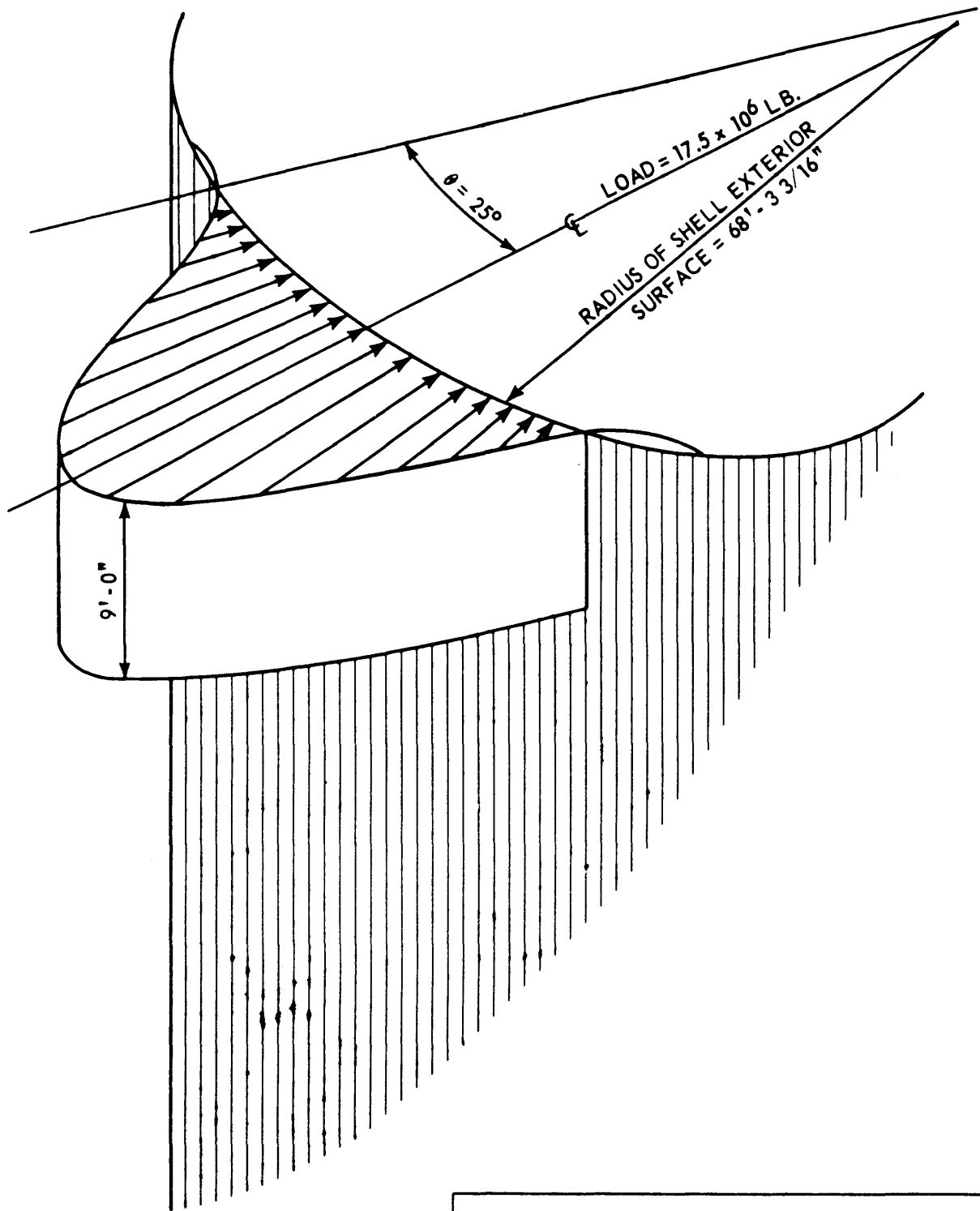
**Time Variation of Shells Surface Stresses Aircraft
with Wings and Engines Attached**

Update -1

7/82

Fig. 3A-16

FOURIER SERIES = $.07955 - .1575 \cos \theta + .1558 \cos 2\theta - .1446 \cos 3\theta + .133 \cos 4\theta$
 $- .1208 \cos 5\theta + .1058 \cos 6\theta - .0894 \cos 7\theta + .0722 \cos 8\theta$
 $- .0551 \cos 9\theta$



DSAR-3A.FIG-17 REV. 2

GPU Nuclear

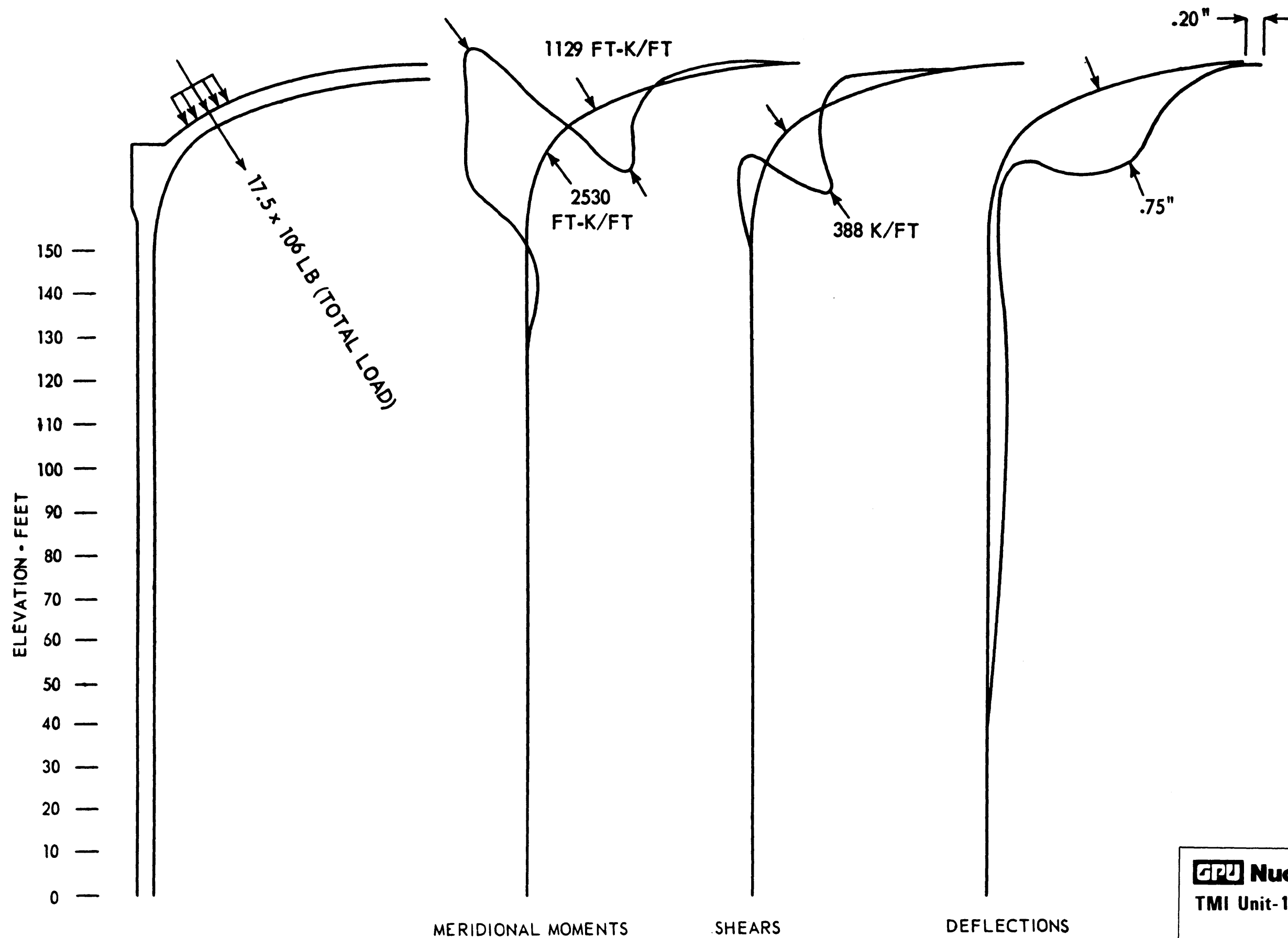
TMI Unit-1

Update - 1

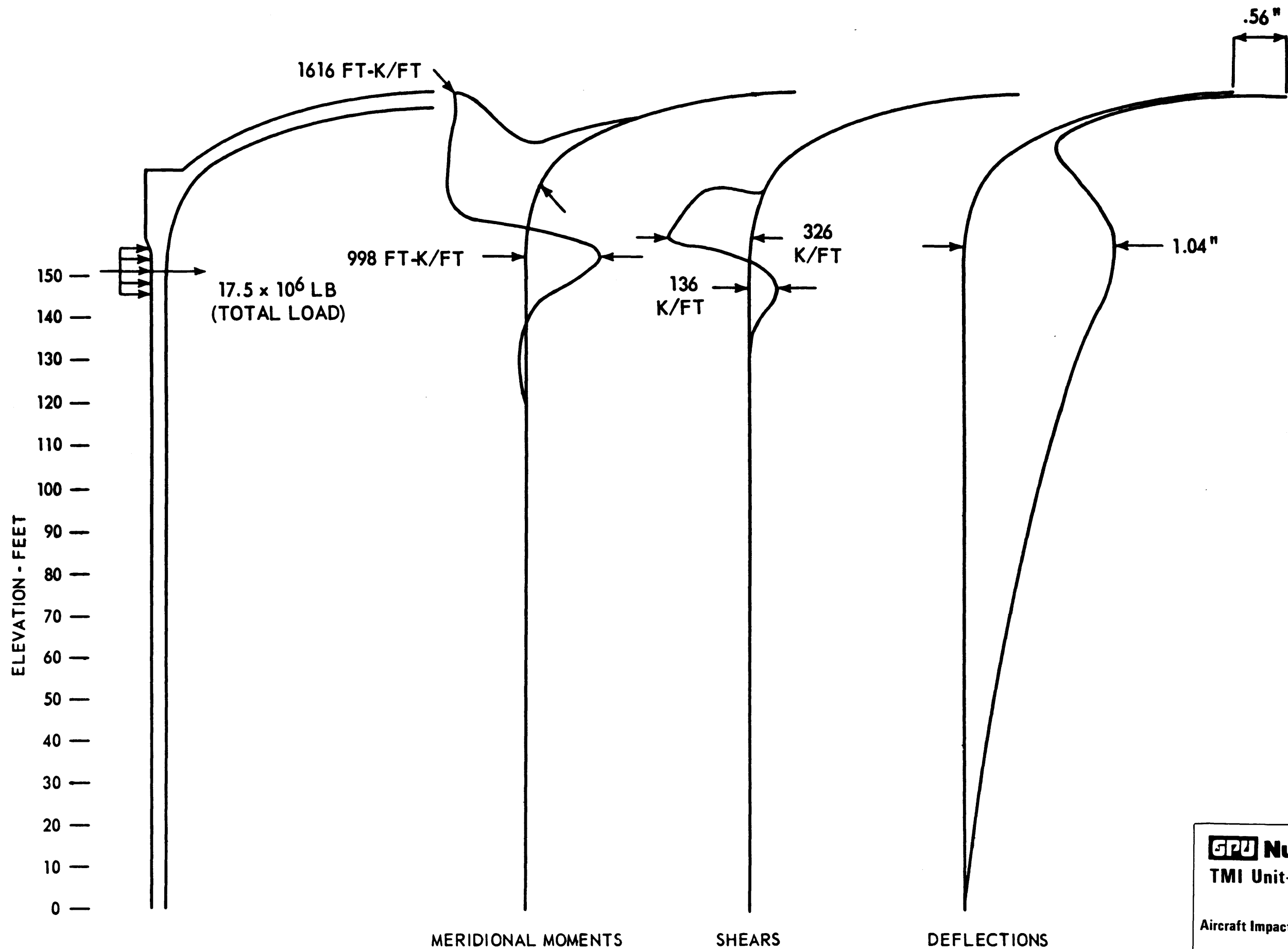
7/82

Pressure Distribution for Aircraft Impact

Fig. 3A-17



DSAR-3A.FIG-18 REV. 2



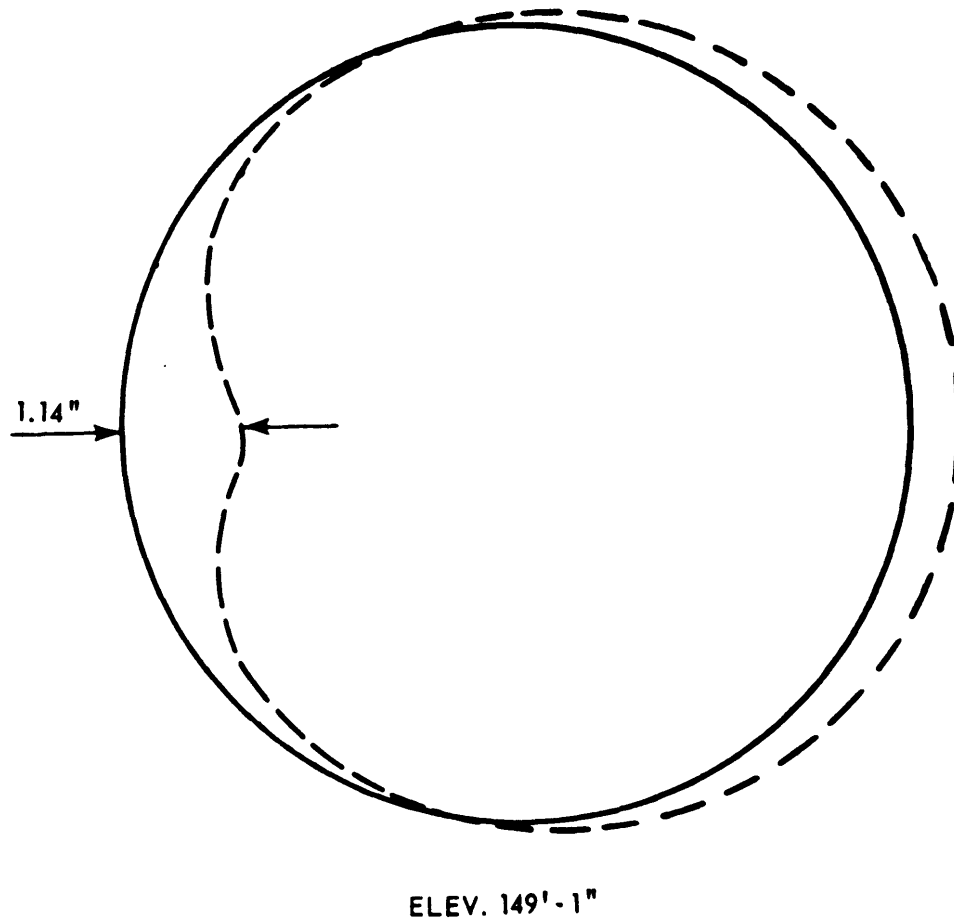
DSAR-3A.FIG-19 REV. 2

GPU Nuclear
TMI Unit-1

Update - 1
7/82

Aircraft Impact at Spring Line

Fig. 3A-19



DSAR-3A.FIG-20 REV. 2

GPU Nuclear

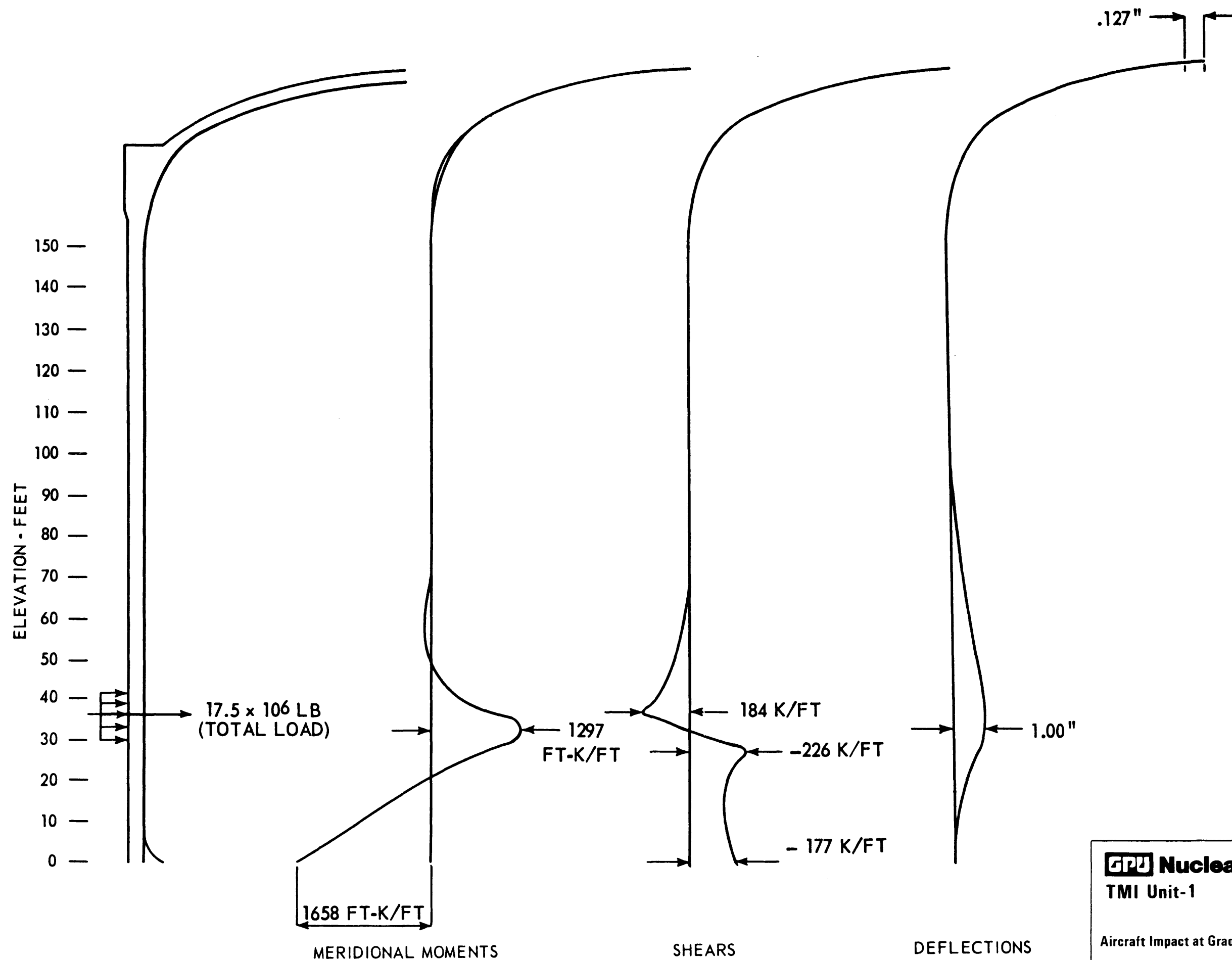
TMI Unit-1

Update - 1

7/82

Radial Deflection Impact at Spring Line

Fig. 3A-20



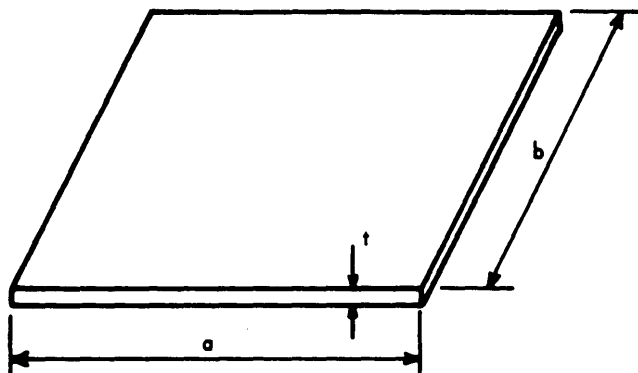
DSAR-3A.FIG-21 REV. 2

GPU Nuclear
TMI Unit-1

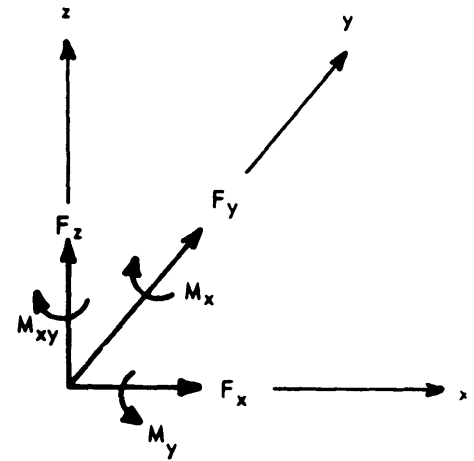
Update -1
7/82

Aircraft Impact at Grade

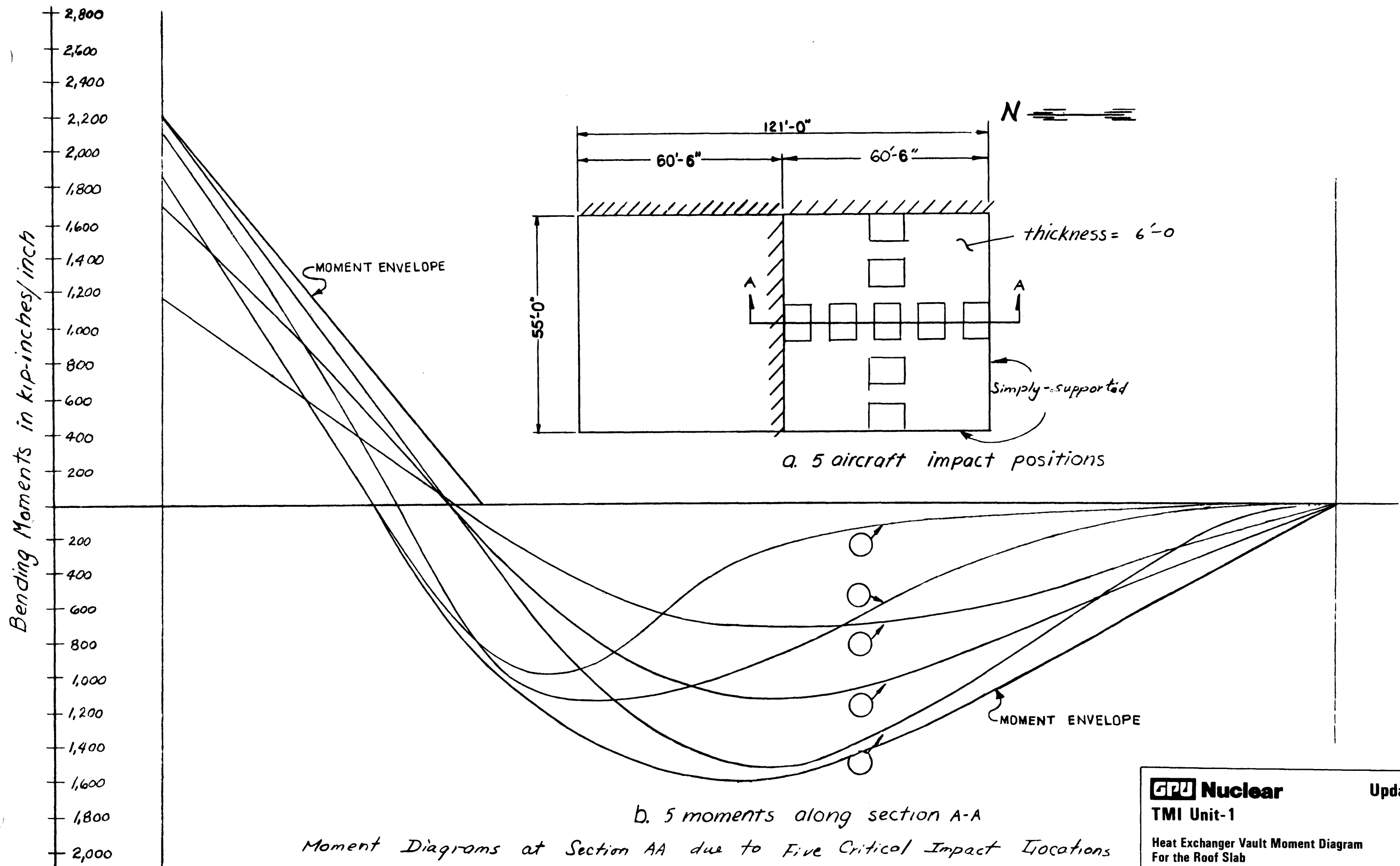
Fig. 3A-21



(a) GEOMETRY



(b) SIX FORCE COMPONENTS AT NODAL POINT

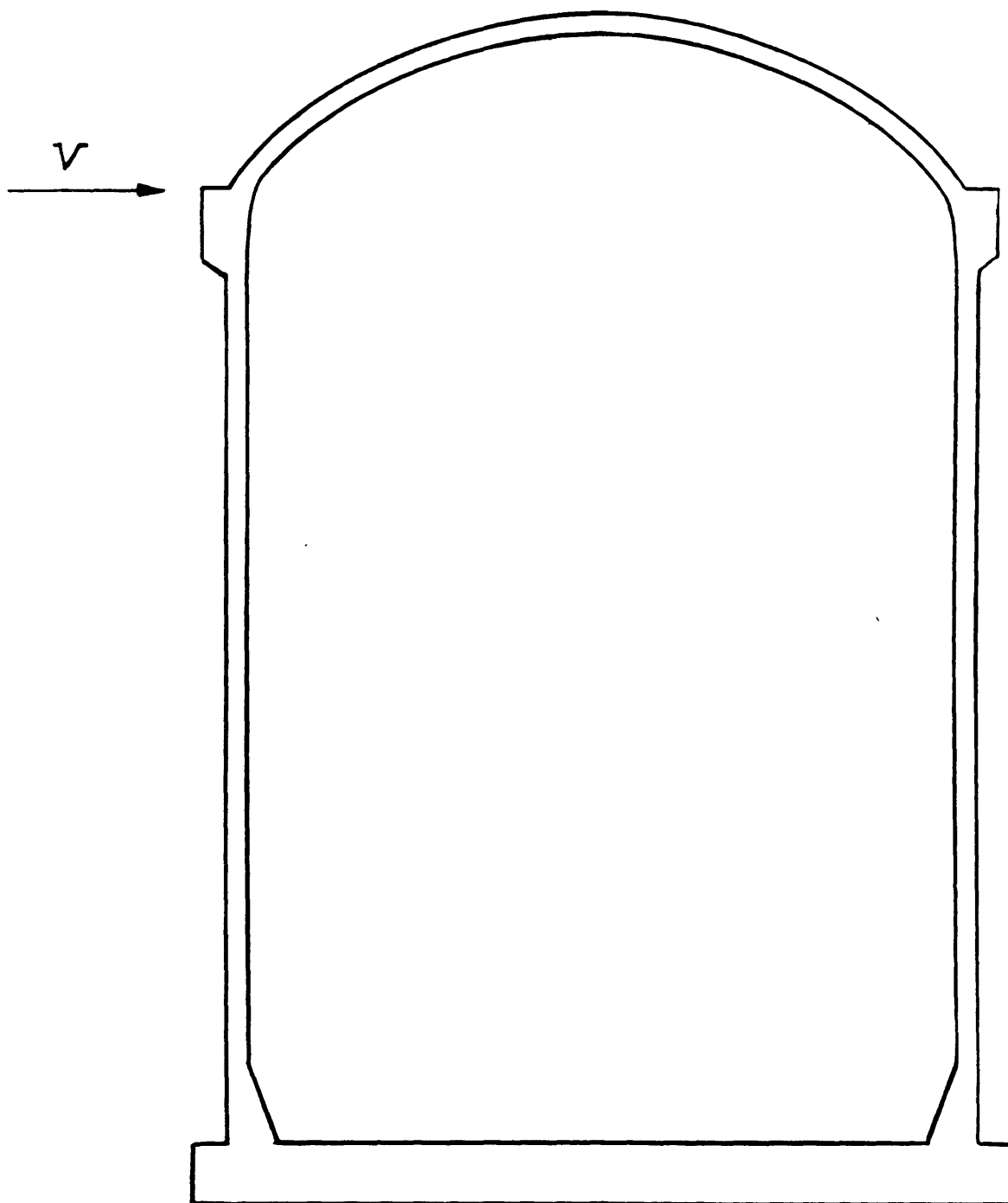


b. 5 moments along section A-A
 Moment Diagrams at Section AA due to Five Critical Impact Locations

TMI DSAR

Figures 3A-24 through 3A-25

Deleted



DSAR-3A.FIG-26 REV. 2

GPU Nuclear

TMI Unit-1

Critical Aircraft Impact-Direction 1

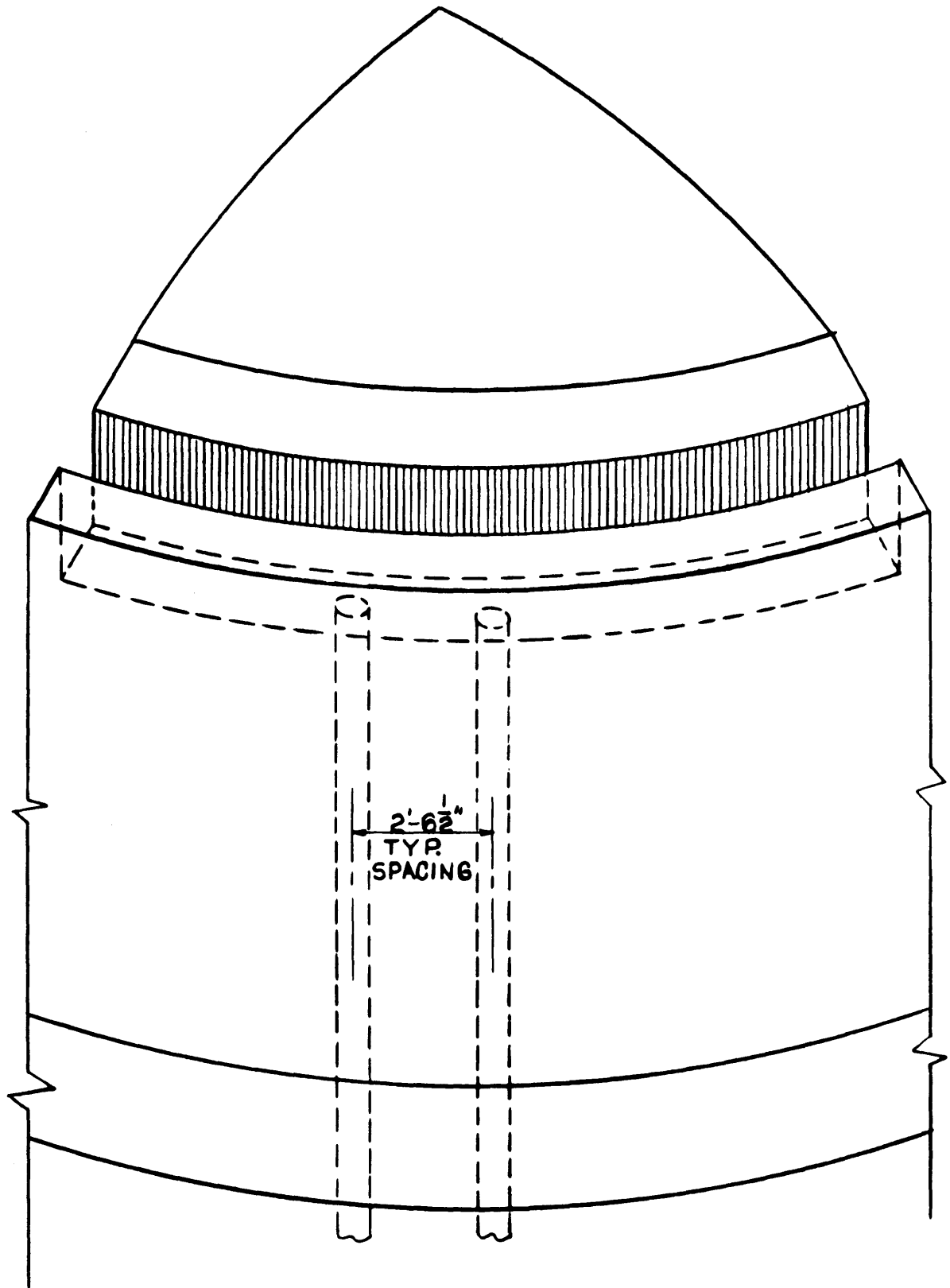
Update - 1

7/82

Fig. 3A-26



Fig. 3A-27



DSAR-3A.FIG-28 REV. 2

GPU Nuclear

TMI Unit-1

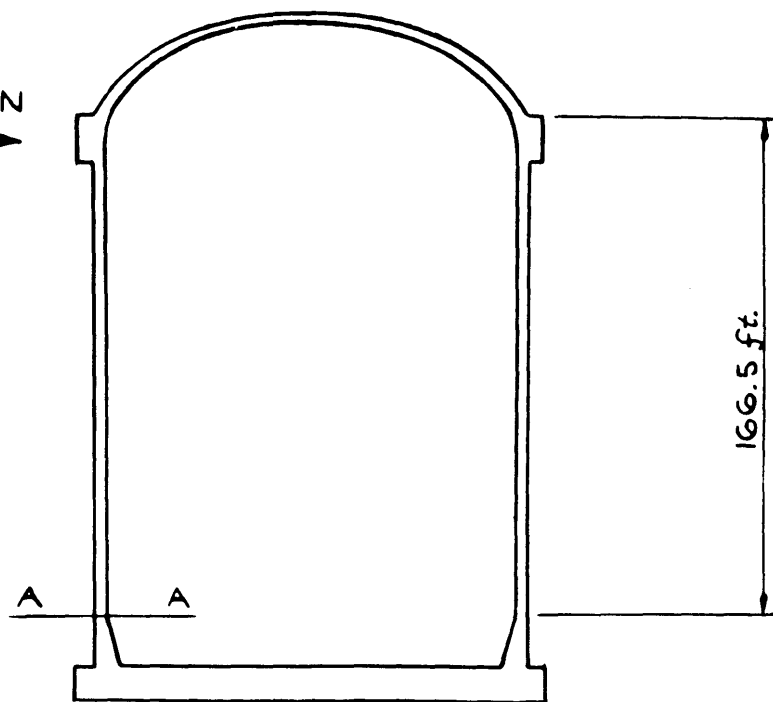
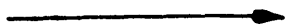
Detail of Anchor Block

Update - 1

7/82

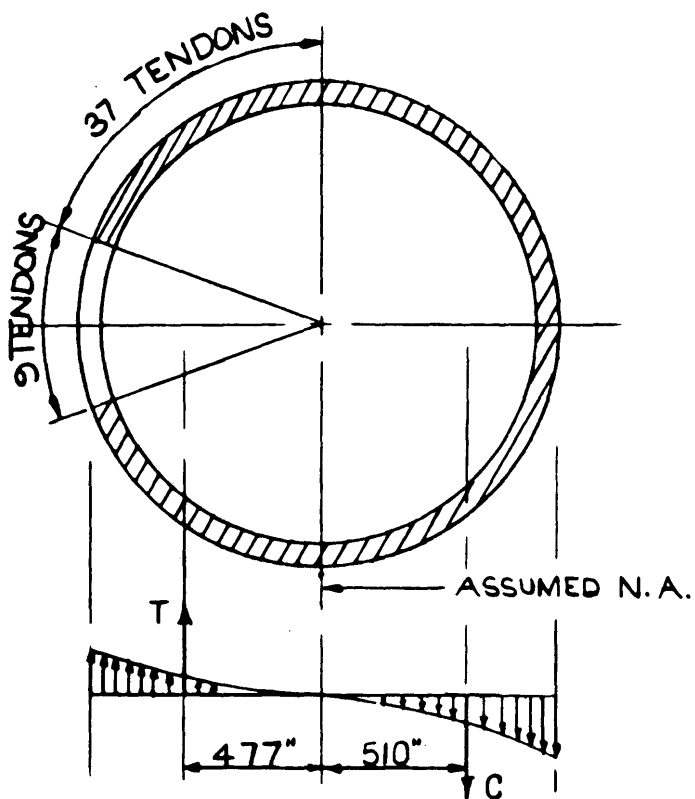
Fig. 3A-28

IMPACT DIRECTION



(a)

(b)



DSAR-3A.FIG-29 REV. 2

GPU Nuclear

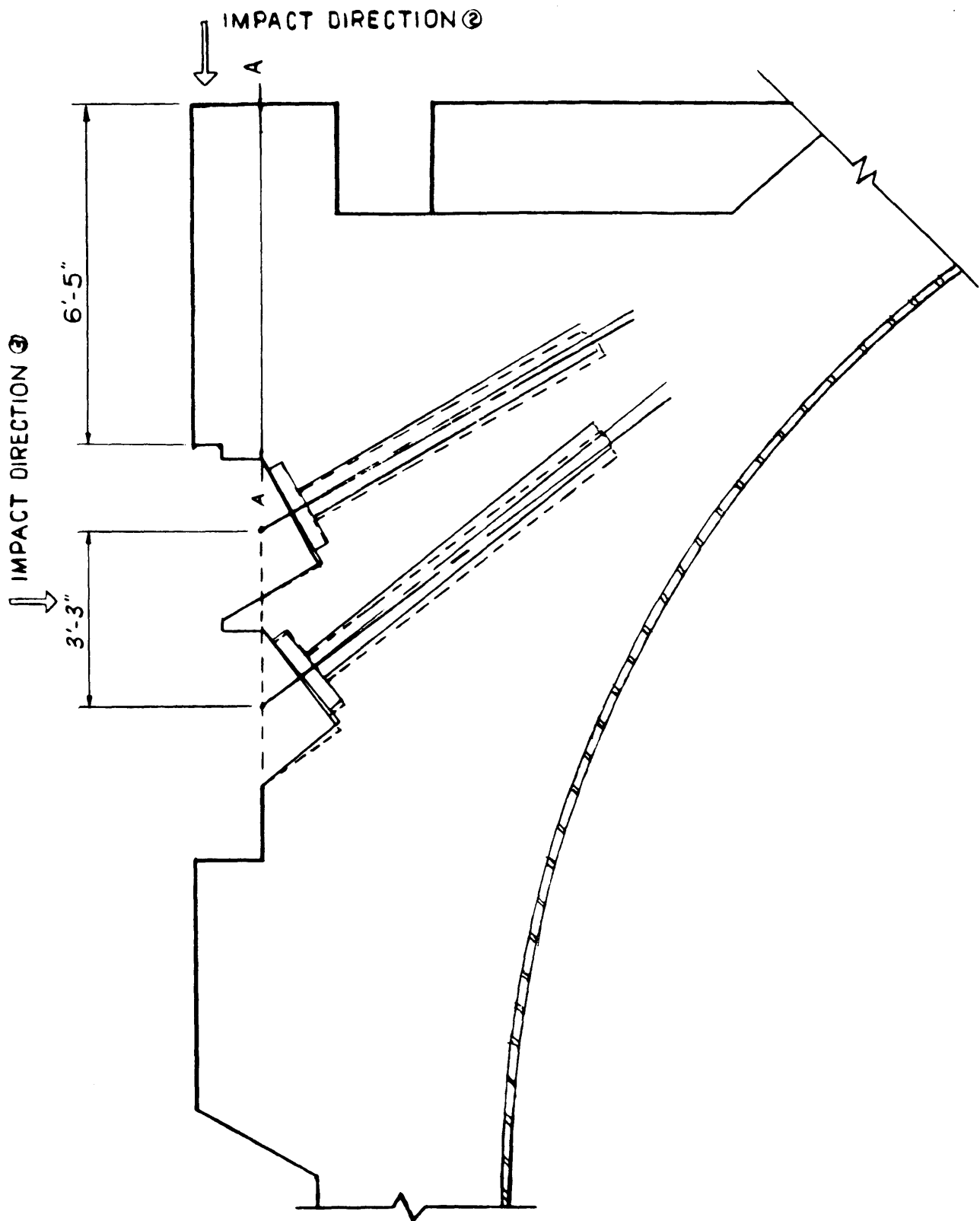
TMI Unit-1

Update -1

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Prestress Stresses after Nine Tendons Fail

Fig. 3A-29



DSAR-3A.FIG-30 REV. 2

GPU Nuclear

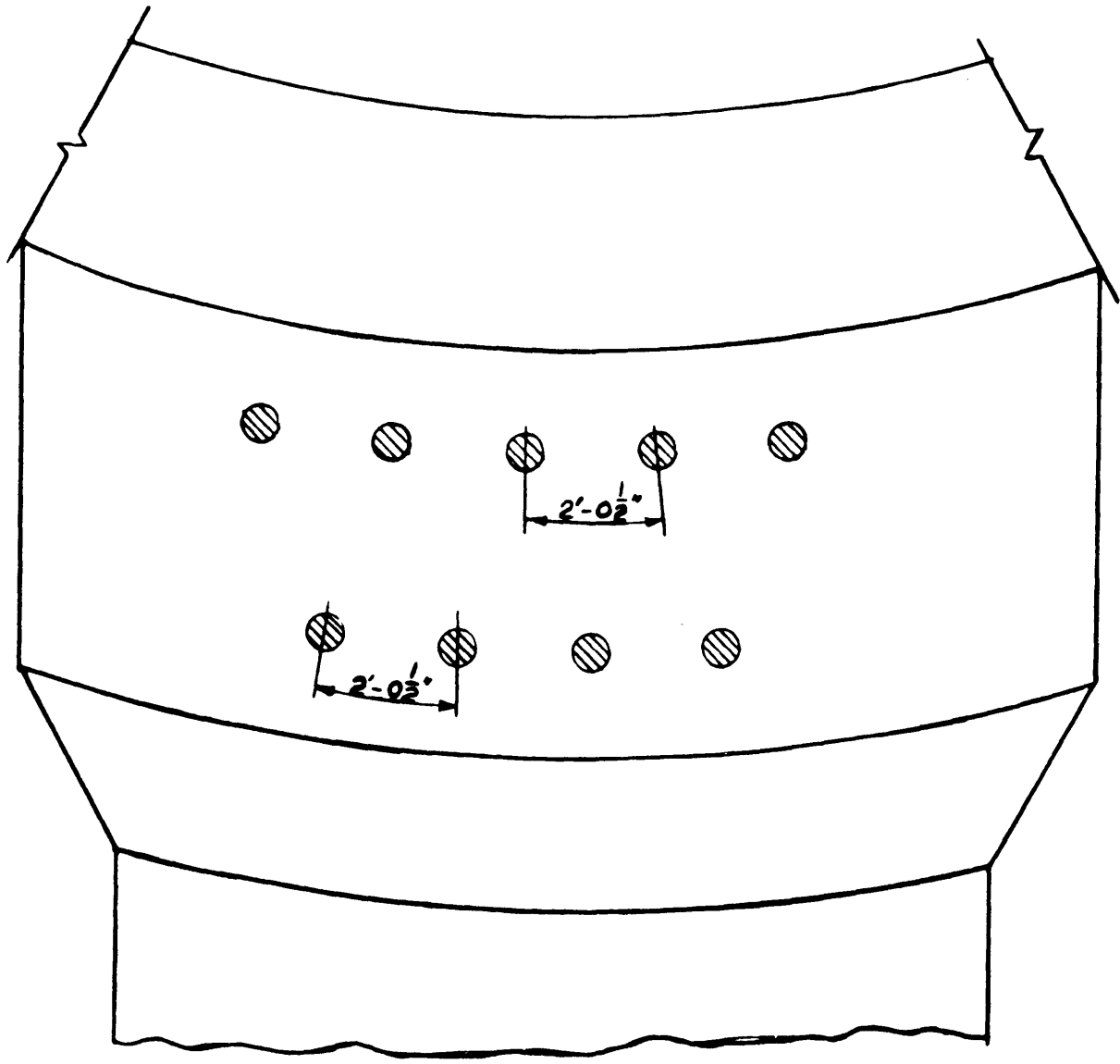
TMI Unit-1

Update - 1

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Critical Aircraft Impact - Directions 2 and 3

Fig. 3A-30



DSAR-3A.FIG-31 REV. 2

GPU Nuclear

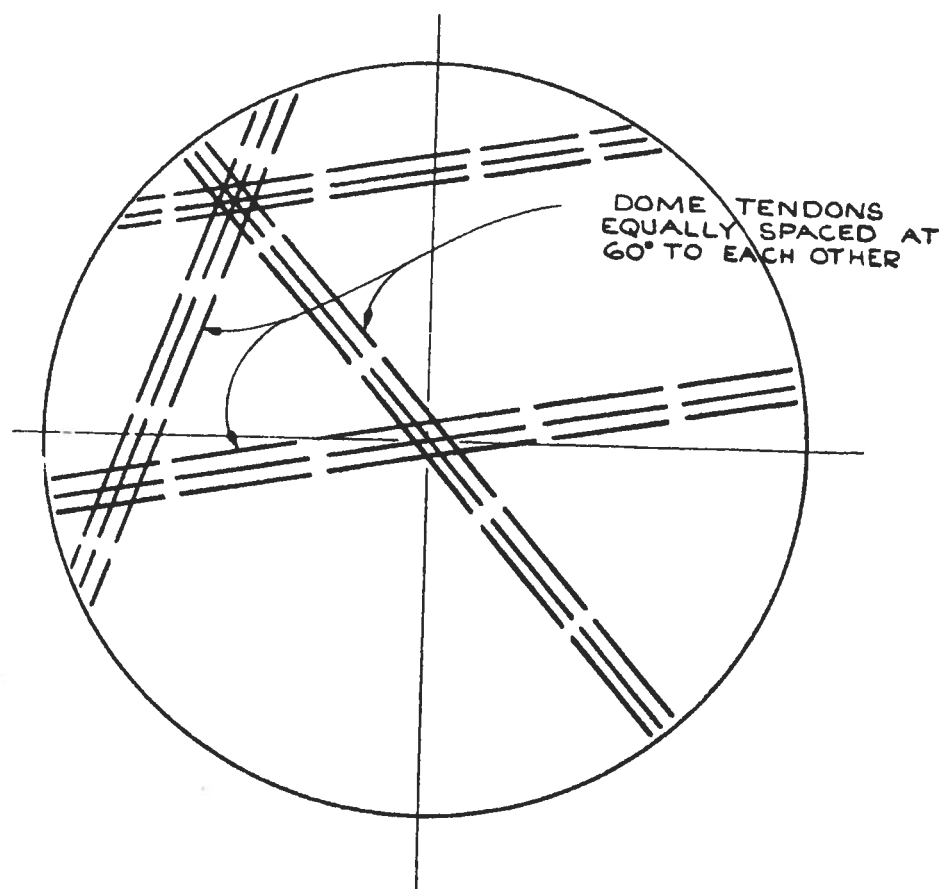
TMI Unit-1

Update -1

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Equal Spacing of Roof Tendons

Fig. 3A-31



DSAR-3A.FIG-32 REV. 2

GPU Nuclear

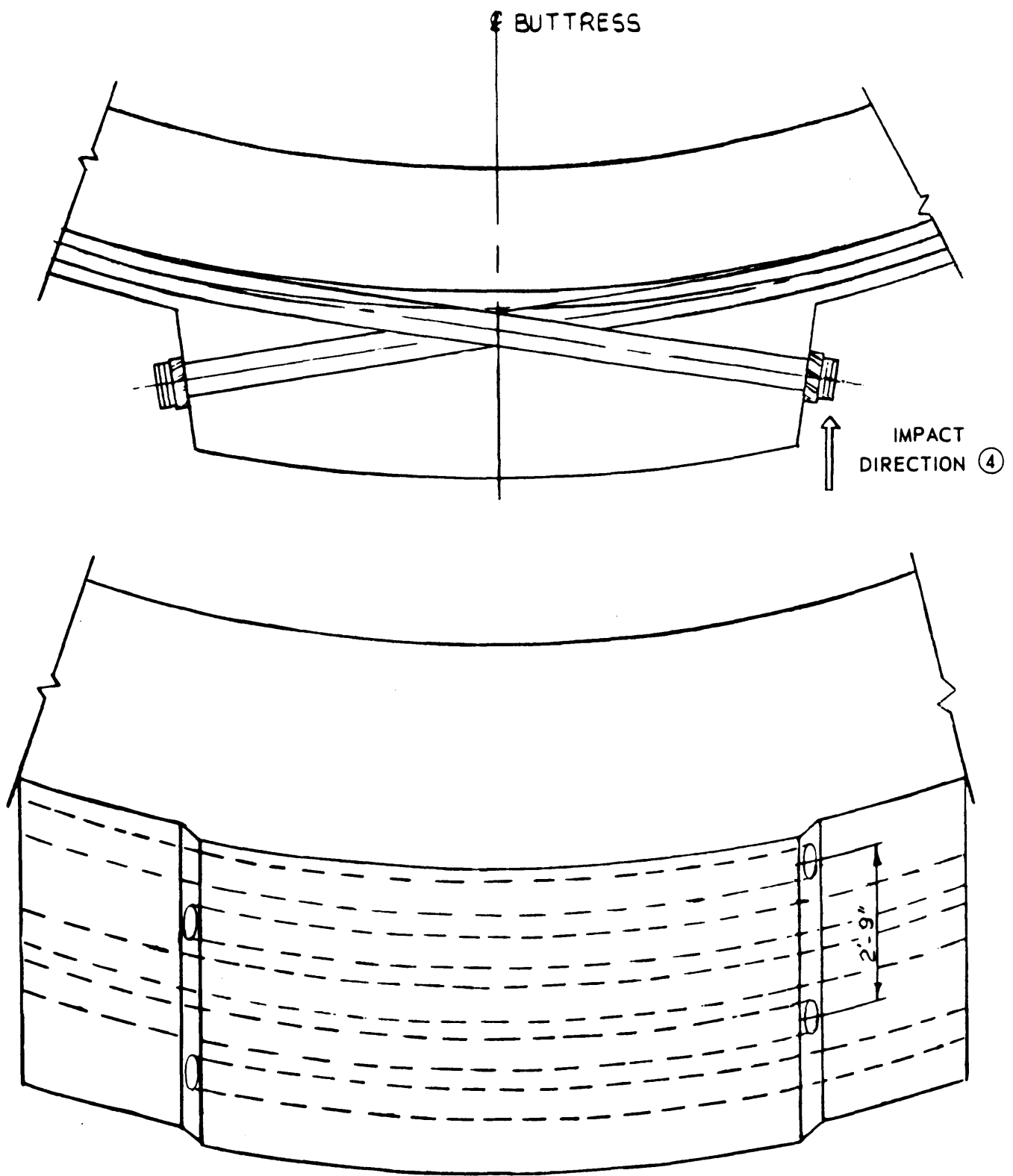
TMI Unit-1

Update -1

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Dome Tendons

Fig. 3A-32



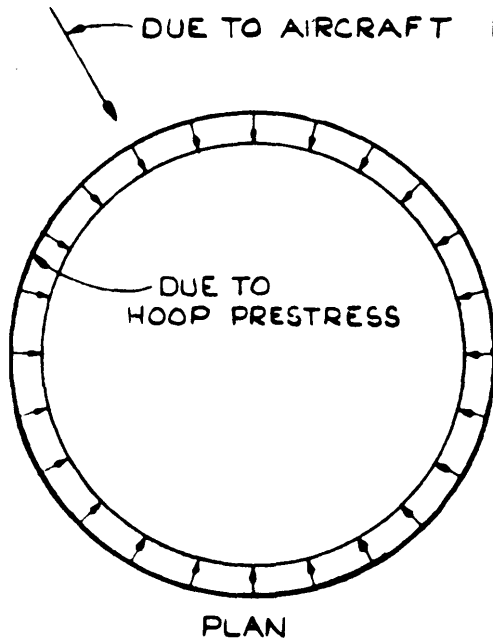
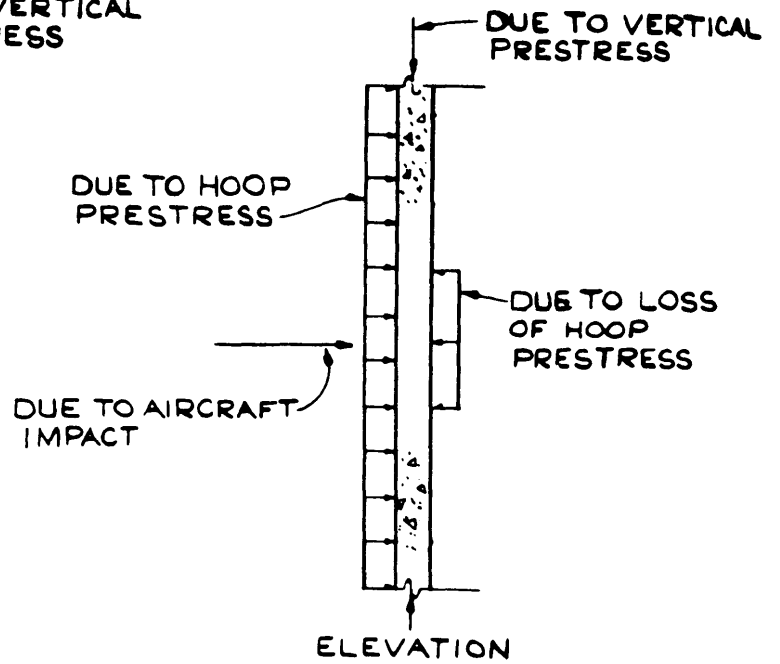
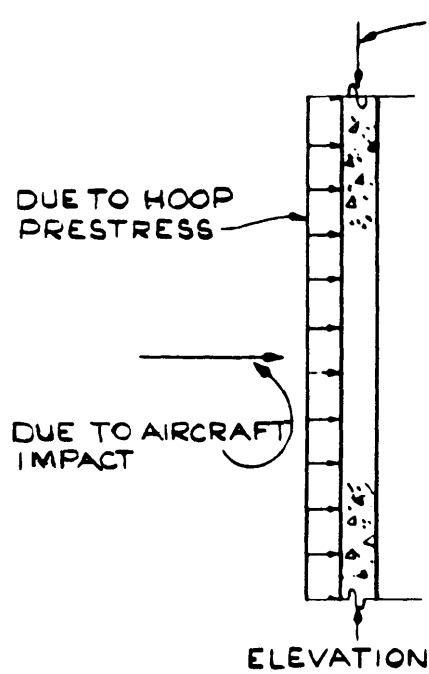
DSAR-3A.FIG-33 REV. 2

GPU Nuclear
TMI Unit-1

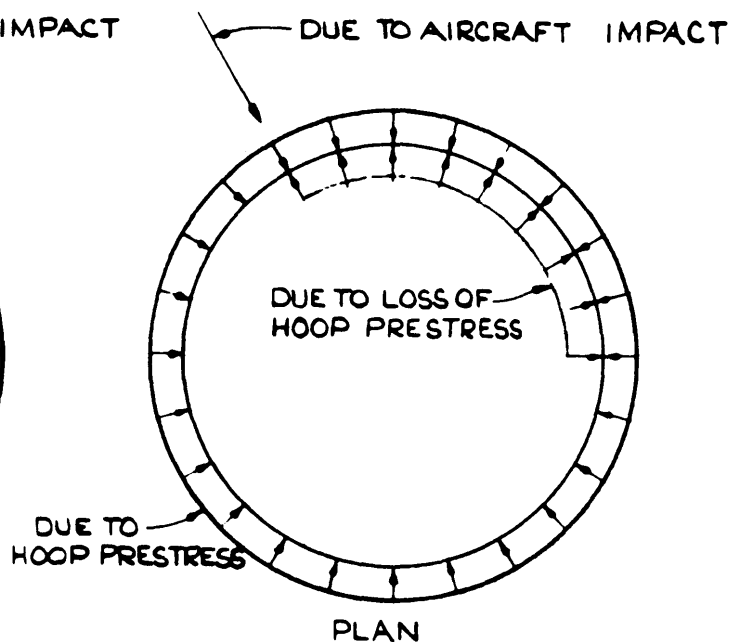
Update - 1
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Minimum Spacing of Hoop Tendons

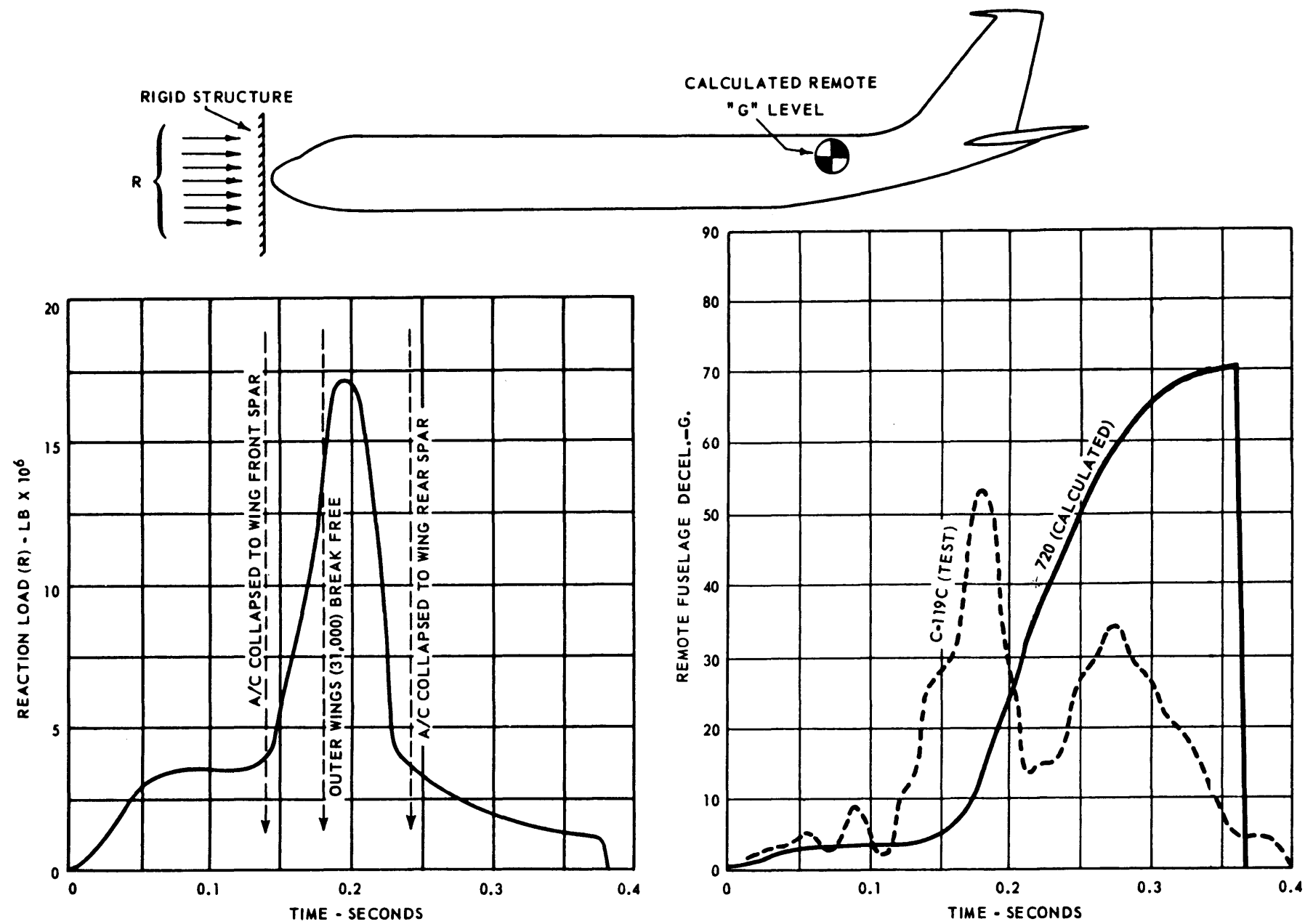
Fig. 3A-33



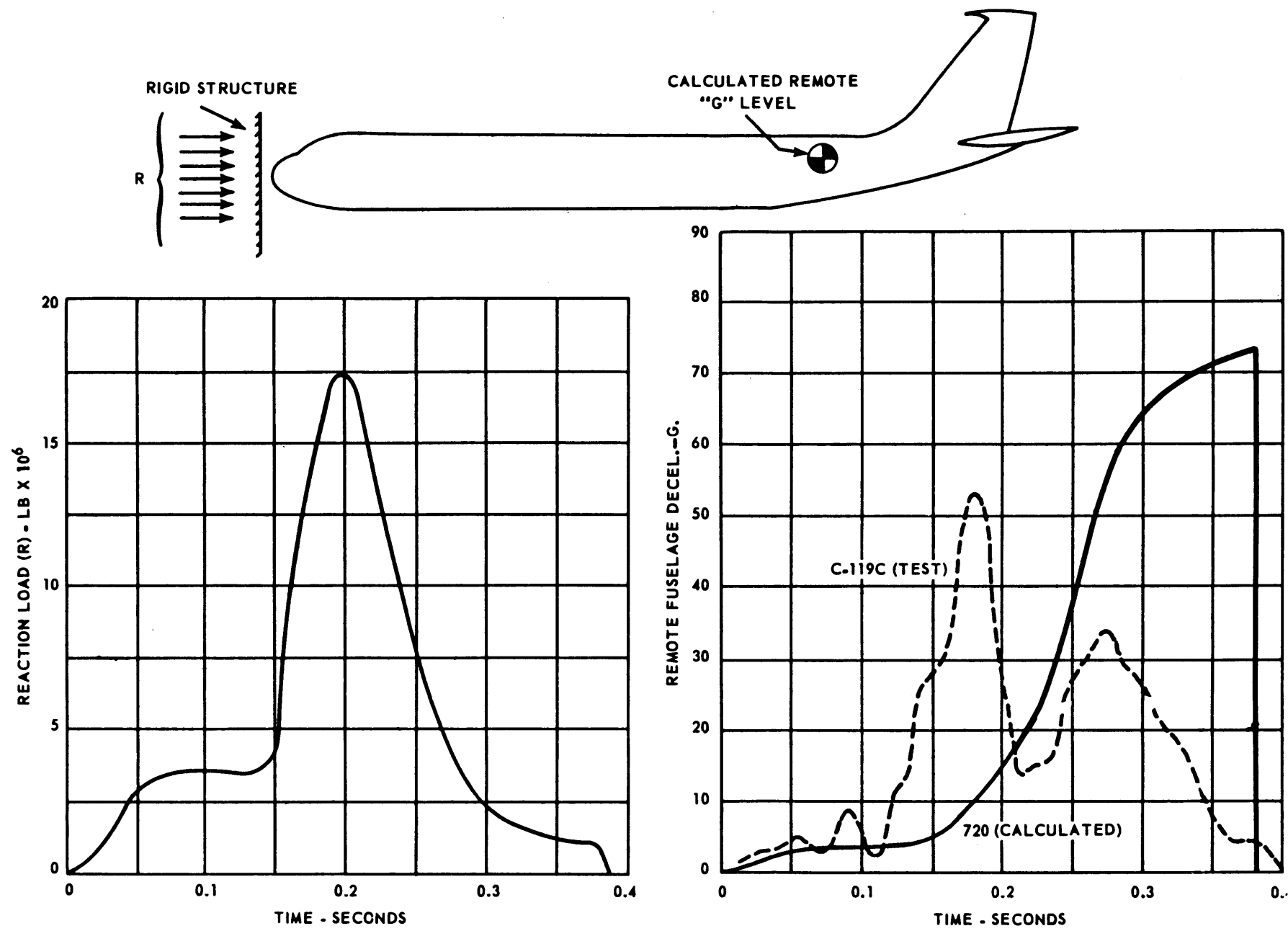
LOADING 1
TOTAL PRESTRESS



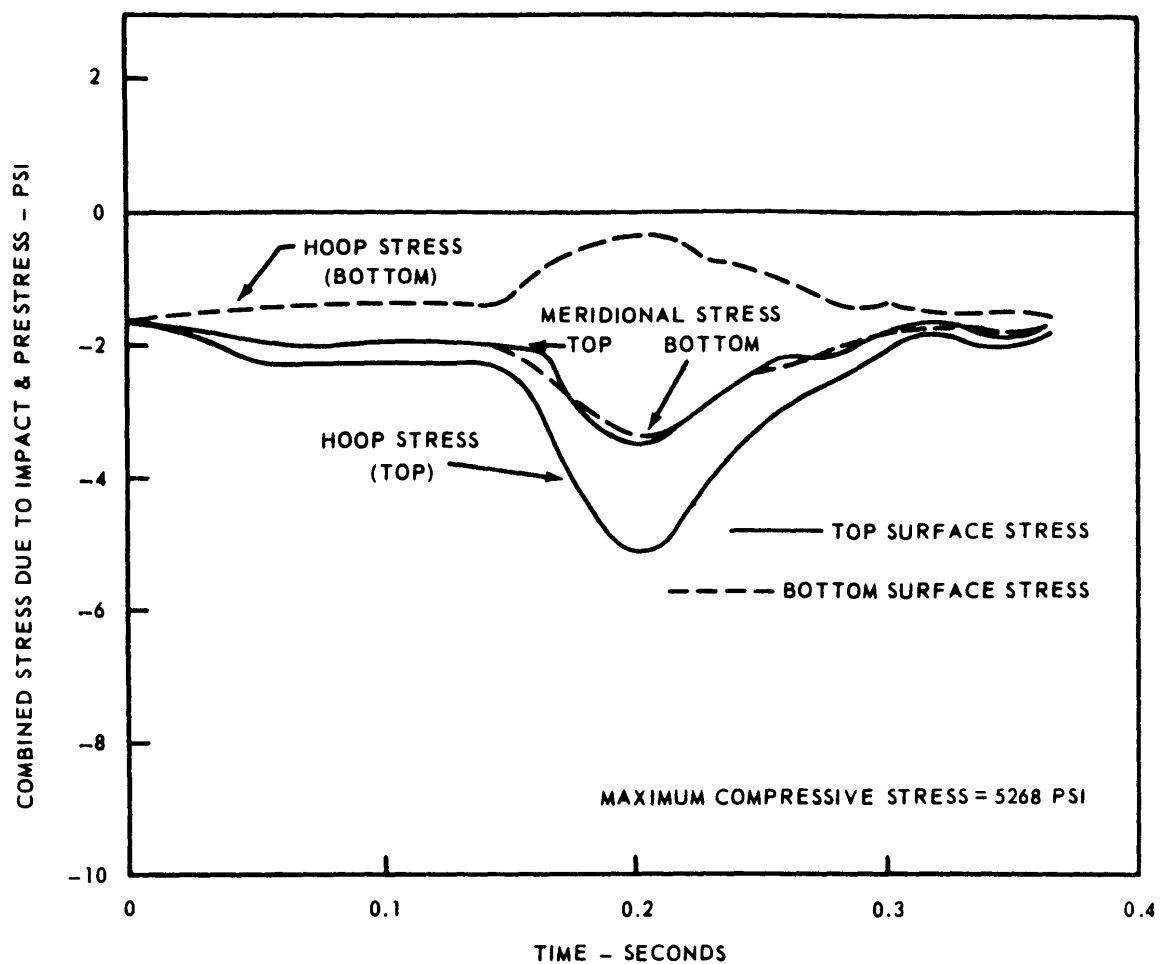
LOADING 2
LOSS OF HOOP PRESTRESS



SEE TABLE 5A-6, REACTION LOAD CALCULATIONS
WITH WINGS & ENGINES DETACHED



SEE TABLE 5A-7, REACTION LOAD CALCULATIONS
WITH WINGS AND ENGINES ATTACHED



DSAR-3A.FIG-37 REV. 2

GPU Nuclear

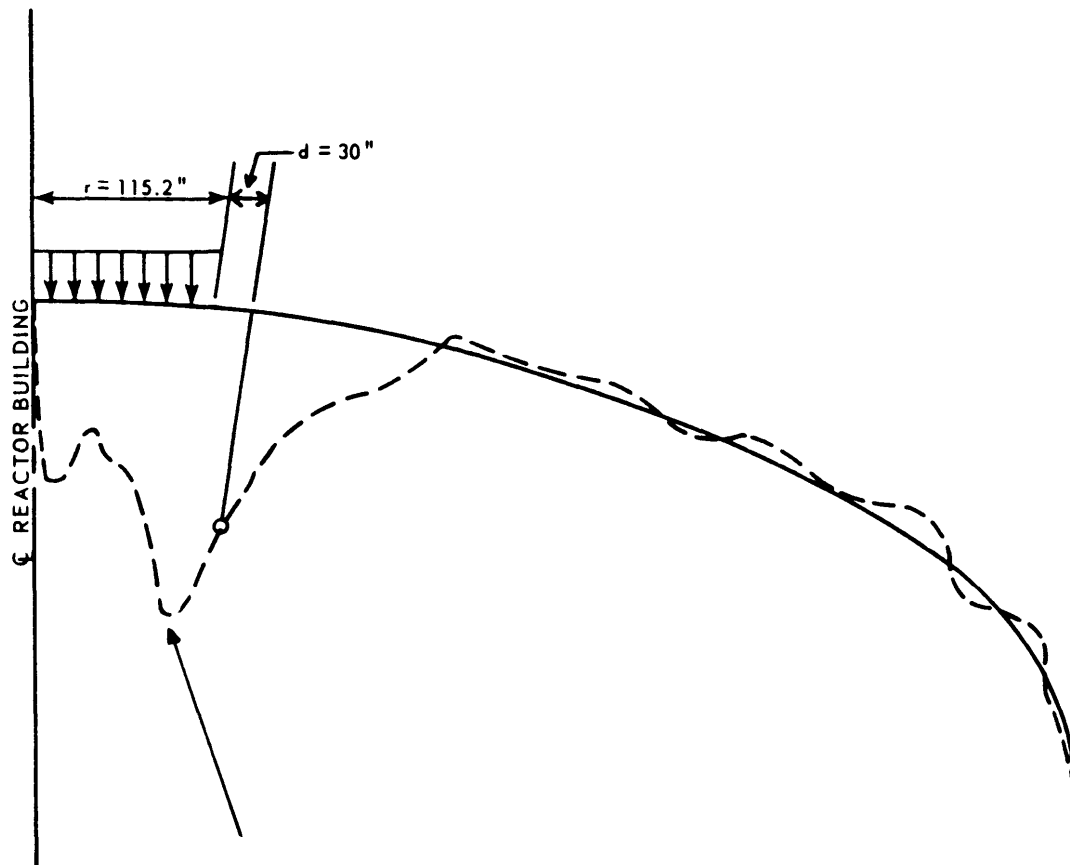
TMI Unit-1

Update - 1

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Hoop and Meridional Stresses at 36 Inches from the Edge of the Load Area

Fig. 3A-37



MAXIMUM AVERAGE SHEAR STRESS ACROSS
SECTION = 482 PSI

SHEAR AT $r = 145.2'' = 354$ PSI

ULTIMATE SHEAR STRESS FROM CHAPTER 26 OF ACI 318 $= v = 3.5 \sqrt{f'_c} + 0.3 f_{pc}$
 $v = 3.5 \times 70.7 + 0.3 \times 1600 = 728$ PSI > 354 PSI OK

DSAR-3A.FIG-38 REV. 2

GPU Nuclear

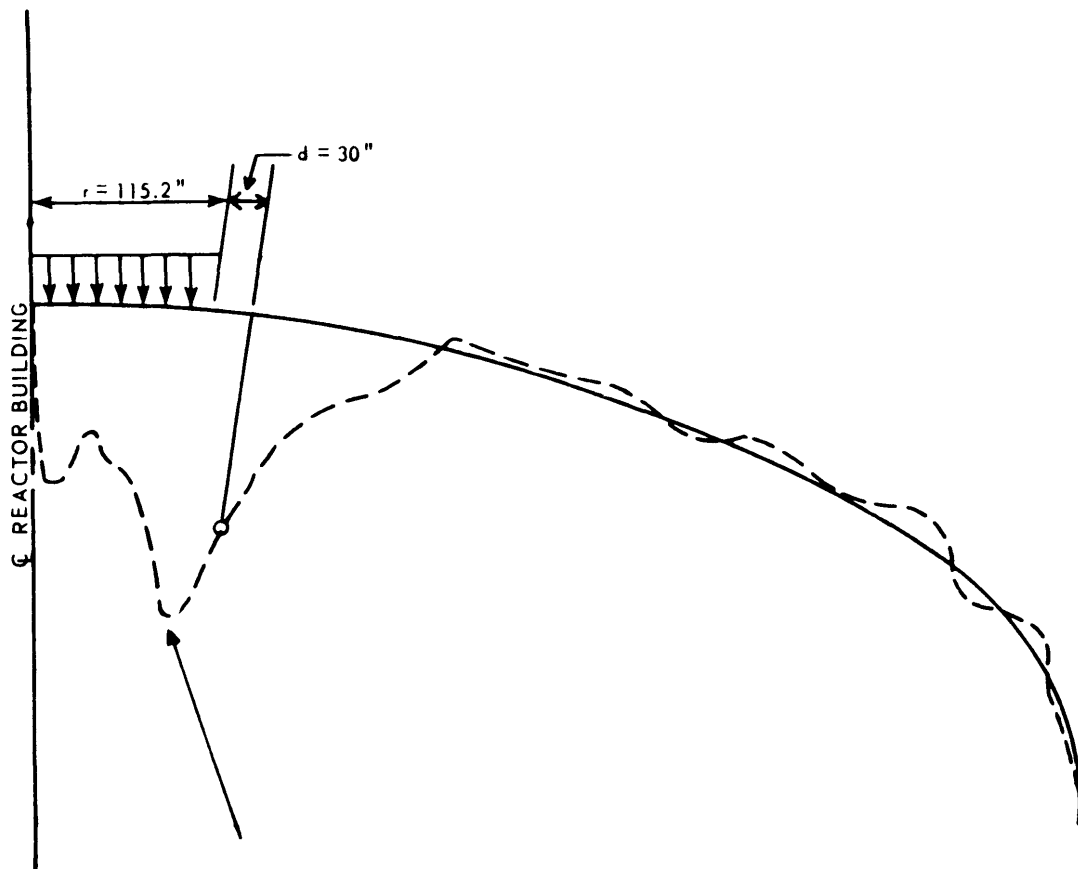
Update - 1

TMI Unit-1

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Average Shear Stress in the Dome at the Time
 $t = 0.20$ Seconds, Wing and Engines Remain Attached
 to Fuselage

Fig. 3A-38



MAXIMUM AVERAGE SHEAR STRESS ACROSS
SECTION = 482 PSI

SHEAR AT $r = 145.2'' = 354$ PSI

ULTIMATE SHEAR STRESS FROM CHAPTER 26 OF ACI 318 $= v = 3.5 \sqrt{f'_c} + 0.3 f_{pc}$

$v = 3.5 \times 70.7 + 0.3 \times 1600 = 728$ PSI > 354 PSI OK

DSAR-3A.FIG-39 REV. 2

GPU Nuclear

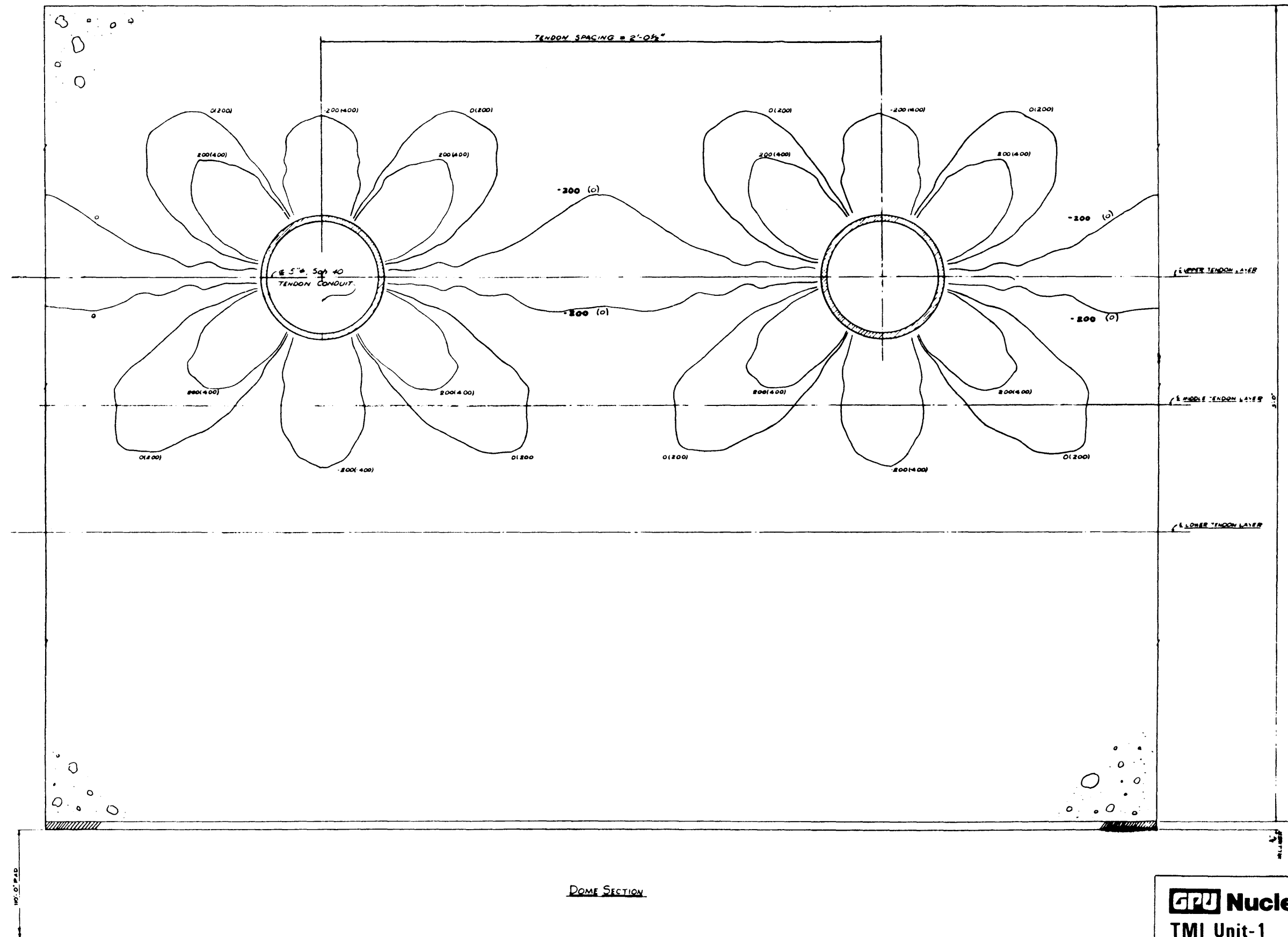
TMI Unit-1

Update - 1

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**Average Shear Stress in the Dome at the Time
 $t=0.20$ Seconds, Wing and Engines Detached from
Fuselage**

Fig. 3A-39



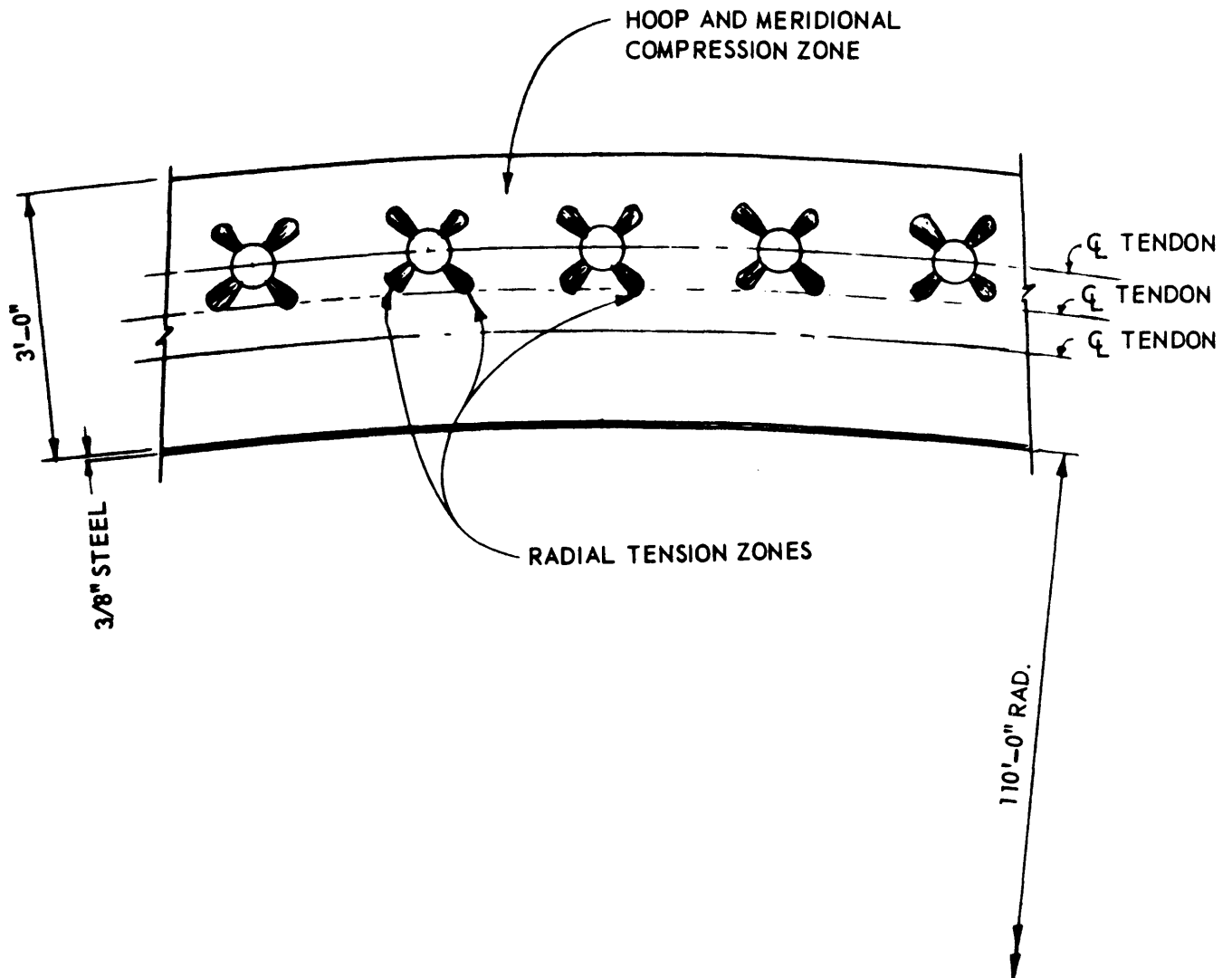
DSAR-3A.FIG-41 REV. 2

GPU Nuclear
TMI Unit-1

Update - 1
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Radial Stresses Due to Prestress and Aircraft Impact

Fig. 3A-41



DSAR-3A.FIG-42 REV. 2

GPU Nuclear

TMI Unit-1

Update -1

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Zones in Compression or Tension Due to Prestress of Aircraft Impact

Fig. 3A-42

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CHAPTER 4 – RADIOACTIVE WASTE AND RADIATION PROTECTION

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4.0 RADIOACTIVE WASTE AND RADIATION PROTECTION

4.1 DESIGN PERFORMANCE OBJECTIVES

The radioactive waste disposal systems and the structures housing them are designed and constructed to meet or exceed the applicable federal regulations for the containment, control, and release or disposal of radioactive liquids, gases, and solids generated as a result of normal and emergency operation of the plant.

Plant structures are designed and constructed, and equipment is located within them, in order to attenuate the radiation received by plant personnel in the discharge of their normal duties to levels not exceeding applicable federal regulations for occupational exposure. The structures also provide shielding of the general public such that applicable federal regulations for radiation exposure are not exceeded during operation of the unit.

The radiation monitoring systems are designed to detect, measure, and alarm or control, as required, radiation and releases of radioactive solids, liquids, and gases produced as a result of normal and emergency operation of the plant. These systems aid in the protection of the general public and plant personnel from exposures to radiation or radioactive materials in excess of those allowed by the applicable federal regulations.

4.2 RADIOACTIVE WASTE DISPOSAL SYSTEMS

The radioactive waste disposal systems are described in the following sections for liquid, gas and solid waste disposal.

TMI-1 was designed and constructed to different criteria than those contained in Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management SSCs Installed in Light Water-Cooled Nuclear Power Plants.". This guide is applicable with the following clarifications:

A. For modifications to existing plant systems, items will be classified by Site Engineering according to the original design basis, or this Guide. This classification will not degrade the safety of the system being modified.

B. Additions to existing plant systems will be designed and constructed to the same codes, standards, and technical requirements which were originally applied to the system to which the addition is to be made, or more recent versions of these codes, standards, and technical requirements. The addition will not degrade the safety of the system being added to.

C. For new construction, the latest applicable codes will be utilized, unless such utilization would result in hardship or unusual difficulty without providing an equivalent level of safety.

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D. Hose may be used in lieu of pipe where the connections are temporary. The anticipated applications of hose would normally be (1) connection to contractor owned skid mounted radioactive waste processing equipment, (2) connections to a non-mounted, frequently-changed component such as a burial liner/HIC, or (3) connection to non-mounted pieces of radioactive waste processing or collection equipment which must be readily removable (e.g., items placed on equipment hatches). The pressure rating of such hoses and connections shall equal or exceed those of the systems or components to which they are connected.

1. Prior to use, the hoses shall be hydro-tested to the appropriate pressure for the system or component to which they will be connected. After installation, they will receive regular hydrotesting or in-service inspections.

2. A 50.59 review process is required to justify the use of such hose connections.

4.2.1 RADIOACTIVE LIQUID WASTE DISPOSAL SYSTEMS

4.2.1.1 Design Basis

The Liquid Waste Disposal System functions to collect, contain, and disposition potentially contaminated water. The disposal process complies with the requirements of TMI1 Decommissioning Quality Assurance Program (Reference 4), the Offsite Dose Calculation Manual (Reference 1) and the NPDES permit.

Proper containment of radioactive material is a primary design consideration. **The major volumes of radioactive liquid are contained within tanks and structures which were originally designed and installed to withstand earthquake forces.** These systems do not have a safety function.

4.2.1.2 System Description

The system is used to collect, contain and dispose of liquid leakage in the Reactor Building (RB), Auxiliary Building (AB), Fuel Handling Building (FHB) or Chemical Cleaning Building (CCB). The radioactive liquid waste disposal system process flow is shown on drawings 302-719.

1. **The primary sources of leakage are rain or ground water intrusion. Any liquid leakage within or into these buildings will be directed to one of following collection points:**

- A. **AB Sump - (AB 254' elev.: depth 7'10': capacity 1380 ft3).**

Any leakage within the RB will be contained and directed to the RB sump. There is an open path from the RB sump to drain to the AB Building sump. Leakage from areas within the AB and AB except as described in Section C, D or E below will be contained and drained to the AB Sump.

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The water level in the AB sump is periodically monitored. The AB sump level is maintained by pumping water to “B” or C” Reactor Coolant Bleed Tanks (RCBTs).

B. Chemical Cleaning Building (CCB) sump:

Any leakage within the CCB would be collected in the CCB sump. The water level in the CCB sump is periodically monitored.

C. Spent Fuel (SF) Pit Sump - (FHB 281’ elev.: depth 6’8”: capacity 204 ft3).

Floor drains from (1) AB 331’ elevation, (2) Intermediate Building (IB) 305’ elevation in front of RB personnel hatch and (3) FHB 305’ elev. drain to the SF Pit Sump. The water level in the SF Pit Sump is periodically monitored. The SF Pit Sump level is maintained by pumping water to a portable tank in the Outage Equipment Storage Building (OESB).

D. Heat Exchanger Vault (HEV) Sump - (AB 271’ elev.: depth 6’: capacity 120 ft3).

The HEV sump is periodically monitored. The HEV sump level is maintained by pumping water to a portable tank in the OESB.

E. Decay Heat & Building Spray Vaults - (AB 261’ elev.).

The “A” Building Spray vault provides a collection sump for ground water intrusion between this area and the Tendon Access Gallery (Reference EC 635009). Ground water intrusion into this area is monitored and pumped to the Turbine Building floor drains to maintain level.

Any leakage into the “B” Building Spray Vault, “A” Decay Heat Vault or “B” Decay Heat Vault will drain to the AB sump.

The following collection points are not within the RB, AB, FHB or CCB, but are located in the immediate vicinity outside of these buildings.

A. Borated Water Storage Tank (BWST) Tunnel Sump - (BWST Tunnel 287’ elev.: depth 3’: capacity 42 ft3).

The BWST tunnel sump is periodically monitored. The BWST Tunnel sump level is maintained by pumping water to a portable tank in the OESB.

B. Air Intake Tunnel (AIT) Sump – (AIT Tunnel 269’ elev.).

The AIT sump is periodically monitored and pumped to the storm drain system to maintain level in the sump.

C. RB Tendon Access Gallery (TAG) – (TAG 262’ elev.).

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Ground water intrusion in this area is monitored and pumped to the Turbine Building floor drains to maintain level.

Ground water or rain water intrusion into other areas where radioactive material is stored (Solid Waste Storage Facility or Interim Solid Waste Storage Facility) will be screened for the presence of any radioactive material. If there is none, the water will be released into the storm drainage system. If any radioactive material is present, then the water will be transported to the OESB and handled as described below.

2. The OESB is a steel structure on a concrete pad which shares the AB west wall (i.e., provides the east wall of OESB). The OESB is utilized for remote monitoring of sumps, to control the electrical power in the AB and FHB, and to provide a temporary storage location for potentially contaminated sump water.

- A. Monitoring of sumps

The monitoring station in the OESB provides access to cameras to view sump float actuated level indicators in the AB sump, BWST Tunnel sump, TAG sump, 'A' BS Vault sump, SF Pit sump, and the Turbine Building sump.

- B. Control of electrical power

Temporary 120 VAC power (breaker in OESB) can be used to supply portable fans, portable light strings, frisker, etc. when access to the AB and FHB is required.

48 VDC power for sump pumps in the BWST Tunnel sump, SF Pit sump, AB sump, TAG sump, and 'A' BS Vault sump are controlled from breakers in the OESB.

- C. Temporary storage of potentially contaminated water

Water will be pumped from sumps as described above, to portable 1000 gallon tanks in the OESB. When the transfer is complete, the contents will be circulated, sampled and analyzed. If the water quality is acceptable, then the water will be released in accordance with ODCM requirements. If the water quality is not acceptable, then the water will be transferred to a qualified water treatment and disposal vendor in accordance with DOT transport requirements. ODCM reporting requirements apply.

3. RCBTs – (WDL-T-1B & WDL-T-1C)

These SS tanks are located on 275' elev. in the AB. These tanks were designed for a temperature of 200° F, with internal pressure between 25 psig and -3 psig in accordance with ASME III-C. Each tank has a USABLE CAPACITY of 11050 ft³.

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The RCBT piping connections (below the potential water level) have been isolated (Reference EC 635009). The tank is used for long term storage of water which is known to or likely to have radioactive material content above acceptable release criteria.

4.2.1.3 Methods of Operation

The potential for radioactive material content in the water was a determining factor for establishing the disposal plan for each location (Reference Tech Eval 632005). The design of these systems is described in EC 635009.

The RCBTs provide storage for water where treatment may be required prior to disposal. A plan for disposal of RCBT contents will be implemented before the tank water level reaches 90% of capacity in both tanks.

Water collected in portable tanks in the OESB will be discharged as a “batch” process through the main station discharge (outfall DSN 001). Prior to the release a representative sample (Reference CY-AA-110-200, Sampling) will be obtained. Release will be permitted if the requirements of the ODCM are satisfied. The radioactive material concentrations and a conservative estimate of the volume will be recorded and used for 10CFR50 Appendix I dose assessments.

Water collected in sumps where radioactive material presence is very unlikely and the storm water system discharge path to the river are periodically sampled to confirm that no radioactive material is present.

4.2.1.4 Evaluation

Since most of the liquid waste disposal system equipment is contained within shielded cubicles and located inside structures originally designed to withstand the hypothetical aircraft incident, it is not considered credible that any single accident could violate the multiple barriers required to release significant quantities of radioactive liquids to the environment.

The portable tanks in the OESB will be located within a water collection berm until the contents is confirmed to meet water quality limits for release. The potential dose consequences of a spill were shown to be well below regulatory limits (Reference EC 637363 Attachment 4).

The volume of water collected in the AB Sump and CCB sump are projected to be small. It is unlikely that disposal of water from these areas would be required for many years. However, the CCB sump and RCBT level will be monitored, and a disposal plan would be implemented if the storage capacity is challenged.

This process of managing ground water and rain water intrusion into areas with radioactive material has been evaluated to ensure that radioactive liquid effluents from TMI-1 will remain below the limits established in 10CFR20 and 10CFR50 Appendix I (Reference EC 638348 Attachment 7).

4.2.2 DELETED

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4.2.3 DELETED

4.2.4 RADIOACTIVE SOLID WASTE DISPOSAL SYSTEMS

4.2.4.1 Design Basis

The function of the Solid Waste Disposal System is to package radioactive solid and concentrated liquid wastes in such a manner as to ensure minimum exposure of unit personnel during the packaging process, produce waste packages that provide protection for the public during their transportation from the unit to an offsite processor or to the ultimate disposal site, and meet ultimate disposal requirements for waste packages sent for direct disposal.

All waste materials are packaged for shipment, processing and disposal activities comply with applicable requirements of 49 CFR 173, 10CFR 61 and 10 CFR 71 (References 9, 10 and 11).

4.2.4.2 System Description

The Solid Waste Disposal System consists of the following facilities:

- A. The Interim Solid Waste Staging Facility (ISWSF) is an above ground storage facility used for temporary storage of low level radioactive material that is properly packaged for off-site shipment prior to shipment offsite to a licensed burial facility or volume reduction vendor prior to disposal.
- B. The Solid Waste Staging Facility (SWSF) is an underground facility used for storage of high dose rate resins. The SWSF design meets the requirements of Reg. Guide 1.143, July 1978.
- C. The Original Steam Generator Storage Facility (OSGSF) is a solid waste storage facility. The OSGSF meets all requirements for this OTSG storage, and in addition will be used to store RCS hot leg piping that was removed for replacement as part of the TMI-1 SGR. The OSGSF is designed to be used as a non-occupied mausoleum whose use is limited to the storage of these large contaminated components until plant decommissioning.

4.2.4.3 Methods of Operation

Dry compactible or non-compactible trash is the only type of waste produced, processed, and shipped from the TMI1 as solid radioactive waste.

Dry trash is either shipped offsite directly or shipped to an offsite processor for decontamination and/or volume reduction prior to recycle or disposal.

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4.2.4.4 Evaluation

Packaging and shipping of dry trash is performed in accordance with applicable shipping and disposal regulations (References 9,10 & 11).

4.2.5 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

The policy for the control of radioactive discharges to the environment from the unit is (1) to discharge the minimum quantity of radioactive materials (either liquids or gases) to the environment which is consistent with the capability of the equipment provided in the unit, and (2) to meet all the conditions of the operating license and all applicable federal, state, and local regulations. The program requirements are described in the ODCM (Reference 2). The station Gaseous Effluent Release Points and Liquid Effluent Outfall Locations are shown on ODCM "Map 1.1".

Major changes to the radioactive waste treatment systems will be evaluated prior to installation in accordance with the ODCM to ensure continued compliance with 10 CFR 50 Appendix I.

4.2.5.1 **DELETED**

4.2.5.2 Radioactive Liquid Release

Rain or ground water intrusion into areas where radioactive material is present will be managed in a controlled manner. The water is collected in portable tanks which are normally staged in the OESB as described in 4.2.1. Based on the batch sample analysis and the river flow a maximum flow rate for the disposal of the batch is determined. The release process requirements are defined in the ODCM (Reference 2).

Any release of radioactive **liquid** from the **site** will be made as follows:

- a. Analysis of **portable/fixed** tank contents prior to their release.
- b. The minimum and maximum river flow rates used to determine the release flow rate are: (1) 1700 cfs (765,000 gpm) which is the minimum daily river flow rate recorded, and (2) 34,000 cfs (15,300,000 gpm) which is the annual average river flow rate (Refer to section 4.2.5.3 for additional basis).
- c. **Release tank** pH is adjusted if required,
- d. Positive manual operation is required to establish a release path.

After liquid waste enters the effluent from the **radiation monitoring pit**, the mixture travels approximately **150** ft before discharging into the west channel of the Susquehanna River, approximately 600 ft downstream of the river water intake structure.

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With respect to the discharge of tritium to the environment, there is no commercial equipment available to remove tritium from the liquid wastes prior to dilution in the **radiation monitoring pit**.

4.2.5.3 Dilution Factors in Susquehanna River

Three Mile Island is located in a section of the Susquehanna River known as Lake Frederick (refer to Figure 2.1-3). Lake Frederick is ponded by the York Haven Dam and is divided into three major flow channels by Three Mile Island and Shelley Island. A section of the York Haven Dam blocks the east channel approximately one mile downstream of the TMINS. The remainder of the York Haven Dam extends in a south, southeasterly direction from the southern tip of Three Mile Island to the west bank of the Susquehanna River, where the York Haven Hydroelectric Station is located. At this point, the west bank of the river lies east of the unit. The river flow in this area is from north to south. TMINS is located near the north end of the Three Mile Island and discharges into the center channel of the pond about 1500 ft downstream of the unit.

The nominal flow capacity of the York Haven Hydroelectric Station is 16,600 cfs with all the turbines running. Therefore, river flows below approximately 17,000 cfs can pass entirely through the hydrostation. The hydrostation can induce spillage over the dam below these flows, however, by reducing the number of turbines operating. Flows in excess of about 17,000 cfs or in excess of the capacity of the operating turbines spill over the York Haven Dam and pass through the Conewago Falls. The hydrostation operates as a run-of-river unit when river flows drop below the operating capacity of the turbines. The operating schedule of the turbines is adjusted such that river flows entering Lake Frederick leave at the same flow rate.

Travel time and mixing characteristics of TMINS liquid effluents were studied by Sutron Corp. in the early 1980s. Sutron modeled characteristics at river flows ranging from 4700 cfs to 85,500 cfs (Reference 12). This range accounts for about 92% of the river flow conditions near TMINS. Sutron found the distribution of river flows through the three channels varies with total river flow. The west channel remains the most constant, ranging from 55% of the total flow at extremely low river flows to 53% during river flows of 85,500 cfs. The west channel also contains the talweg of the river. The proportion of flow in the center channel varies inversely with flows in the east channel. The east channel flow varies from 0% when the river is not spilling over the east dam to 15% at flows of 85,500 cfs. Conversely, about 45% of the river passes through the center channel when the east channel has zero flow. The proportion of river flow in the center channel drops to about 32% when the east channel flows increase to 15%.

When the entire river flow passes through the hydrostation, liquid effluents from TMINS enter the center channel of the pond, travel along the west bank of Three Mile Island, finally reaching the York Haven Dam. The surface areas of the center channel impacted by TMINS effluents increases to about 625 feet at the dam. From there, this band travels along the face of the dam narrowing to approximately 200 feet at the head race section of the hydrostation. Throughout this reach, the amount of river flow impacted by discharges from TMINS remains fairly constant at about 15%. The York Haven Hydrostation discharges towards the eastern bank of the river downstream of the York Haven Dam. At extremely low flows, this results in a complete mixing of the water impacted by TMINS effluents with the rest of the river.

At river flows above the operating capacity of the hydrostation, the portion of the river impacted by TMINS effluents travels along and eventually over the dam and through the Conewago Falls.

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Flows over the dam are then compressed towards the east side of the river by the discharge from the hydrostation. This area of the river, the York Haven Dam, Conewago Falls and the discharge from the hydrostation, accounts for the most significant contribution to mixing of TMINS effluents with the rest of the river. At medium flows, approximately 35% of the area of the river just downstream of the hydrostation is impacted by TMINS effluents. This increases to about 68% as the total river flow increases to about 85,000 cfs. At medium and high flows, effluents from TMINS do not mix completely with the Susquehanna River as far downstream as Lake Clark.

Dilution of TMINS liquid effluents follow similar patterns to the mixing characteristics. The most substantial dilution occurs as the TMINS effluent enters the Susquehanna River at the main station outfall. Only a small amount of additional dilution occurs from the outfall to the York Haven Dam. However, total dilution exceeds 80% (dilution factor of 5) prior to spilling over the dam or reaching the York Haven Hydroelectric Station during all flow conditions. Additional dilution, second only to the initial discharge, occurs as the river passes over the dam, through the Conewago Falls or through the hydrostation. Effluents are diluted by essentially 100% by the time effluents reach the head waters to Lake Clark.

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4.2.5.4 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

For any major change to the radioactive waste treatment systems (liquid and solid), the next Annual Radioactive Effluent Release Report (Reference: **Decommissioning Quality Assurance Program, Appendix E**) will include a discussion of the change and the following:

- a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.
- b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information.
- c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems.
- d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto.
- e. An evaluation of the change which shows the expected maximum exposures to individuals in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto.
- f. A comparison of the predicted releases of radioactive materials, in liquid and in solid waste, to the actual releases for the period prior to when the changes are to be made.
- g. An estimate of the exposure to plant operating personnel as a result of the change.
- h. Documentation of the fact that the change was reviewed and approved.

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TABLE 4.2-1
(Sheets 1 through 3)

DELETED

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4.3 RADIATION SHIELDING

Radiation shielding for Unit-1 is provided principally by concrete walls, floors, and ceilings or roofs that are also structural members of buildings, which are designed to meet the requirements of the controlling criterion (i.e., structural strength or radiation shielding). Therefore, where structural strength is the controlling criterion, the radiation shielding provided is in excess of requirements.

To comply with radiation exposure limits specified in 10CFR20, the shielding is designed to attenuate radiation levels throughout the plant, from direct and scattered radiation, to the dose rate levels specified for each of the areas indicated below:

4.3.1 PRIMARY SHIELD

The primary shield serves to protect plant personnel, surrounding structures, and primary system components from the gamma radiation escaping the reactor vessel. It is an annular cylinder of reinforced concrete which surrounds the reactor vessel and extends upward from the Reactor Building floor to the bottom of the fuel transfer canal. The shield thickness is 5 ft up to the height of the reactor vessel flange, where the thickness is reduced to 4.5 ft. It is designed to sufficiently reduce the radiation, escaping the reactor vessel after shutdown, to permit limited access to the reactor coolant system equipment.

4.3.2 SECONDARY SHIELD

The secondary shield is a 4 ft thick reinforced concrete structure surrounding the primary shield and the bulk of the primary system equipment, including piping, pumps, and steam generators. It was designed to attenuate radiation sufficiently to permit limited access to certain areas within the Reactor Building during full power operation. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield.

4.3.3 REACTOR BUILDING

The Reactor Building is a reinforced, prestressed containment structure with 3.5 ft thick cylindrical wall and a 3.0 ft thick dome that completely surrounds the nuclear steam supply system. Other significant radiation shielding inside the Reactor Building is listed in Table 4.3-1.

4.3.4 CONTROL ROOM

The Control Room shield is 2 ft thick and was designed to permit continuous occupancy by essential Control Room personnel following a Maximum Hypothetical Accident.

4.3.5 AUXILIARY BUILDING

Auxiliary building includes all concrete walls, covers, doors, and removable blocks which provide radiation protection of personnel from the numerous sources of radiation occurring from equipment housed within the Auxiliary Building. The major components and their required shielding are given in Table 4.3-1.

4.3.6 DELETED

4.3.7 ORIGINAL STEAM GENERATOR STORAGE FACILITY (OSGSF) SHIELDING

The Once-Thru Steam Generators (OTSGs) and hot leg piping which were replaced in 2009 are stored in the Original Steam Generator Storage Facility (OSGSF). The OSGSF meets the radiation shielding requirements of 40 CFR 190, 10 CFR 20, and the Plant License. The building is designed to have a maximum contact dose rate of 0.2 mR/hour on the exterior wall surface, and also provides a locking access control entrance door behind a concrete labyrinth designed to provide shielding.

4.3.8 MATERIALS

The material used for the primary, secondary, Reactor Building, Control Room, auxiliary, and spent fuel shields is concrete with a density of approximately 150 lb/ft³.

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TABLE 4.3-1
(Sheet 1 of 3)
PRINCIPAL SHIELDING

Reactor Building

<u>Component</u>	<u>Concrete Thickness (ft)</u>
Primary shield (below flange)	5
Primary shield (above flange)	4.5
Secondary shield	4
Reactor Building vertical walls	3.5
Reactor Building dome	3
Side walls of fuel transfer canal	4
End walls of fuel transfer canal	3 to 4
Floor of fuel transfer canal	4
Reactor coolant drain tank room walls	1.5
Minimum water over active fuel during transfer (Reference Sterns Rogers Dwg. No. 21804-1)	6.5

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TABLE 4.3-1
(Sheet 2 of 3)
PRINCIPAL SHIELDING

Auxiliary Building

<u>Component</u>	<u>Adjacent Concrete Wall Thickness (ft)</u>
Miscellaneous waste storage tank	3
Reactor coolant bleed tanks	3
Waste transfer pumps	2 to 4
Waste gas compressors	2 to 4
Waste gas delay tank	2 to 6
Reactor coolant waste evaporator	2 to 4
Miscellaneous waste evaporator	2 to 4

Auxiliary Building

<u>Component</u>	<u>Concrete Thickness (ft)</u>
Boric acid recycle pumps	2 to 4
Reclaimed boric acid tanks	3
Used filter precoat tank	3
Spent resin storage tank	3
Makeup tank	3
Makeup and purification pumps	2 to 3
Waste gas decay tanks	3 to 4
Condensate demineralizer	2.5
Makeup and purification demineralizers	3.5
Deborating demineralizers	3 to 5
Cation demineralizers	3 to 5
Precoat filters	2 to 3
Waste drumming area	3

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TABLE 4.3-1
(Sheet 3 of 3)

PRINCIPAL SHIELDING

Fuel Handling Building

<u>Component</u>	<u>Concrete Thickness (ft)</u>
Side walls of spent fuel pools	5
End walls of spent fuel pools	5
Bottom of spent fuel pools	5
Pools water-gate wall	5
Spent fuel shipping cask pit walls	1.5
Minimum water over active fuel during transfer	6.5 (water)
Minimum water over active fuel in storage racks	23 (water)

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4.4 RADIATION MONITORING SYSTEM

4.4.1 DESIGN BASIS

The Radiation Monitoring System detects, indicates, annunciates, and records the radiation level at selected locations inside and outside the plant as required for compliance with 10CFR20 limits.

4.4.2 SYSTEM DESCRIPTION

The Radiation Monitoring System, as shown on Table 4.4-1 is divided into the following subsystems:

a. Atmospheric monitoring

All of the monitors except RM-A12 **and** RM-L12 display their measured variable and alarm condition in the Unit 1 Control Room. Each monitor is equipped with two adjustable alarm levels and a channel failure and/or loss of power indication. Interlock functions performed by the radiation monitoring system are shown on Table 4.4-1.

4.4.2.1 **DELETED**

4.4.2.2 Atmospheric Monitoring Subsystem Description

The Atmospheric Monitoring Subsystem provides effluent monitoring and in-plant air monitoring. The atmospheric monitoring subsystem includes channels located inside and outside the plant with the following functions:

a. **DELETED**

b. Fuel Handling Building Exhaust Ventilation Duct: Channel RM-A4 and Auxiliary Building Exhaust Ventilation Duct: Channel RM-A6

These in-plant air monitors are used for Auxiliary and Fuel Handling Building air activity monitoring. Their function is to provide an early detection of activity. It measures gaseous activity and contains a particulate filter which is periodically removed and analyzed. Channels RM-A4 and RM-A6 will initiate shutdown of the related ventilation air supply system should their preset radiation setpoint be exceeded.

c. **DELETED**

d. Auxiliary and Fuel Handling Building Exhaust: Channel RM-A8

This channel is part of the effluent monitoring system that monitors the activity released from the Auxiliary and Fuel Handling Buildings through the unit vent.

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This channel measures gaseous activity and contains a particulate filter which is periodically removed and analyzed. Should the gas channel's preset level be exceeded, air supply for the Auxiliary and Fuel Handling Buildings will be shut down and the waste gas decay tanks' discharge valve is closed.

e. **DELETED**

4.4.2.3 **DELETED**

4.4.2.4 Supplemental Samplers and Monitors

Composite proportional flow samplers are installed on (1) the common discharge from the turbine building sump and the powdex sump to the Industrial Waste Treatment System and (2) the station river water discharge near RM-L7.

Additional air sampling instruments are located at various locations in the TMI-1 Auxiliary and Fuel Handling Buildings. Numerous regulated air samplers, and beta particulate air monitors are in service. These air monitors supplement the Radiation Monitoring System.

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TABLE 4.4-1
(Sheet 1 of 1)
RADIATION MONITORING SYSTEM DATA

Monitor	System / Location	Detector Type	Range	Interlocks
RM-L-12	IWTS/IWFS Discharge IWST/IWFS bldg.	Nal / Liquid Monitor	10 – 10 ⁶ cpm	SHUT DOWN IW-P-16, 17, 18, 29 & 30 CLOSE IW-V-73 & 279
RM-A-4	Fuel Handling Area Discharge AB 305' elev. (Radwaste Pnl)	Gas /B _{SCINT} Atmospheric Monitor NOTE 1	10 – 10 ⁶ cpm	SHUT DOWN AH-E-10 CLOSE AH-D-120, 121 & 122
RM-A-6	Aux & FHB 281 & 305 Discharge AB 305' elev. (Radwaste Pnl)	Gas /B _{SCINT} Atmospheric Monitor NOTE 1	10 – 10 ⁶ cpm	SHUT DOWN AH-E-11
RM-A-8	Aux & FHB Common Discharge RM Bldg. 310' elev. (yard north of Aux Bldg.)	Gas /B _{SCINT} Atmospheric Monitor NOTE 1	10 – 10 ⁶ cpm	SHUT DOWN AH-E-10 & 11 CLOSE WDG-V-47

NOTE 1: Particulate filter is installed in monitor and can be removed and analyzed.

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4.5 RADIOLOGICAL HEALTH AND SAFETY

4.5.1 RADIATION WORK PERMITS

A Radiation Work Permit is required by all personnel prior to any work or entry into restricted areas containing radioactive material as specified in applicable procedures. Only in the event that the safety of the plant or its personnel is endangered may entry be made into a Radiation Work Permit area without a Radiation Work Permit. In such a case, the entry will be made by qualified personnel carrying monitors, and a Radiation Work Permit will be executed shortly thereafter.

4.5.2 PERSONNEL MONITORING SYSTEM

Dose is maintained within the limits established in 10CFR20. All personnel entering a radiation area will wear dosimetry devices as specified by procedure or the Radiation Work Permit. Types of dosimetry that are readily available include self-reading dosimeters, alarm capable dosimeters, thermoluminescent dosimeters, and extremity dosimetry. This personnel monitoring equipment will be available on a day-to-day basis for those persons, employees, or visitors not assigned to the station who have occasion to enter Radiation Control Areas, or to perform work involving possible exposure to radiation. Records of radiation exposure history and current occupational exposure are maintained by the Radiological Controls section for each individual requiring personnel monitoring. The external radiation dose to personnel will be determined for each Radiation Work Permit entry and assessed annually or more frequently as necessary.

4.5.3 PERSONNEL PROTECTIVE EQUIPMENT

Special protective or anticontamination clothing will be furnished and worn as necessary to protect personnel against contact with radioactive contamination. Change areas are conveniently located for proper utilization of this protective clothing.

Respiratory protective equipment is also available for the protection of personnel against airborne radioactive contamination. The equipment consists of full-face filter masks, self contained air breathing units, or air-supplied masks and hoods. The first line of defense against airborne contamination in the work area is the ventilation system. However, respiratory protective equipment is provided should its use become necessary.

Maintenance of the above equipment is in accordance with the manufacturer's recommendations and 10CFR20, "Standard for Protection Against Radiation." The use and maintenance of this equipment are under the direct control of the Radiological Controls Section. All personnel will be trained in the use of this equipment, utilizing it in the performance of work.

4.5.4 ACCESS CONTROL

Change area facilities are provided where personnel may obtain clean protective clothing required for station work. These facilities are divided into clean and contaminated sections. The contaminated section of the change area is used for the removal and handling of contaminated protective clothing after use. Showers, sinks, and necessary monitoring equipment are also provided to aid in the decontamination of personnel.

Equipment decontamination facilities are provided at the station for large and small items of plant equipment and components.

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All facilities and equipment necessary to exclusively support the radiation protection and decontamination activities of Unit 1 are provided.

Provision is made for decontamination of work areas throughout the station. Appropriate written guidance governs the proper use of protective clothing, where and how it is to be worn and removed, and how the change area and decontamination facilities for personnel, equipment, and station areas are to be used.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any high radiation areas. Warning signs, audible and visual indicators, barricades, and locked doors will be used as necessary. The Radiation Work Permit system is also utilized to control access to high radiation areas.

4.5.5 RADIOLOGICAL CONTROL LABORATORY FACILITIES

The unit includes a Radiological Control Laboratory with facilities and equipment for detecting, analyzing, and measuring alpha, beta, or gamma radiation and for evaluating any radiological problem which may be anticipated. Counting equipment (such as G-M, scintillation, and proportional counters) is provided in an appropriately shielded environment for detecting and measuring all types of radiation as well as a High Purity Germanium detector/MCA for the identification of specific radionuclides.

Site procedures are written to provide instructions and guidance on radiation protection, sampling, laboratory analyses used to determine concentration and species of radioactive liquids and gases, and calibration of laboratory equipment. Surveillance procedures are written to ensure compliance with Technical Specification requirements.

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4.5.6 RADIOLOGICAL CONTROL INSTRUMENTATION

Portable radiation survey instruments are provided. A variety of instruments have been selected to cover the entire spectrum of radiation measurement problems anticipated at TMI-1. Sufficient quantities have been obtained to allow for use, calibration, maintenance, and repair. This includes instruments for detecting and measuring alpha, beta, gamma, and neutron radiation. In addition to portable radiation monitoring instruments, fixed monitoring instruments, i.e., count rate meters, are located at exits from radiation control areas. These instruments are intended to prevent any contamination on personnel, material, or equipment from being spread into unrestricted areas. Appropriate monitoring instruments are available at various locations within the radiation control areas for contamination control purposes. The station has permanently installed process radiation monitoring systems as described in DSAR section 4.4.

4.5.7 MEDICAL PROGRAM

A physical examination and medical surveillance program is provided to establish and maintain records of the physical status of those employees who must meet specific OSHA and regulatory physical criteria in order to perform their duties at the Three Mile Island nuclear facility. Medical examinations shall be performed as required.

4.5.8 RADIATION PROTECTION PROGRAM

4.5.8.1 Foundation for the TMI-1 Radiological Controls Program

The objective of the Radiological Controls Program is to control radiation hazards, to avoid accidental radiation exposures, to maintain exposures within the regulatory requirements, and also to maintain exposures to workers and the general population as low as is reasonably achievable. These philosophies, policies, and objectives are based on and implement the regulations of the NRC as contained in Title 10 of the Code of Federal Regulations, Parts 19, 20, 50, and 71, and appropriate Regulatory Guides, specifically 8.2 (1973), 8.8 Rev. 3 (1978), 8.9 Rev. 1 (1993), 8.10 Rev. 1-R (1975), 8.13 Rev. 3 (1999), and 8.14 (1977).

Specific details as to how the Radiation Protection Program is implemented is promulgated in site and company procedures and includes those applicable procedures addressed in Reg. Guide 1.33 Rev. 2 (1978), App. A, paragraph 7, and paragraph 8 (1) (aa), (bb).

The Radiological Controls procedures are written to provide an effective Radiological Controls Program and ensure **compliance** with 10CFR20. Procedures provide adequate guidance and specify appropriate methods or techniques to ensure that the performance of each activity is in accordance with sound radiological control principles and is in compliance with applicable regulatory provisions. The Radiological Controls procedures are prepared, reviewed, approved, and controlled as described in company administrative procedures.

The TMI-1 Radiological Controls Program is fully integrated into each and every phase of operations at TMI-1. The TMI-1 Radiological Controls Program, when carried out as specified, will assure that the operation of Unit 1 will be performed with personnel who work at the site incurring radiation exposure as low as can reasonably be achieved (ALARA).

In order to meet this objective, the program must be carried out by each person involved in the TMI-1 activities. There is no group or person involved in the TMI-1 operations who does not have some degree of responsibility for the Radiological Controls Program.

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4.5.8.2 Audits, Reviews, And Reports on the TMI-1 Radiological Controls Program

Each individual is responsible for maintaining his or her radiation exposure as low as reasonably achievable while completing the scope of work he or she is required to perform. Each is required to comply with the applicable procedures of the TMI-1 Radiological Controls Program and the specific radiological controls prescribed for work in which he or she is engaged.

To ensure that these requirements are being met and to assist all site personnel in understanding and complying with these requirements, a system of audit and review procedures has been established, including criteria for timely and appropriate corrective action. The following assessments and review procedures are used:

- a. Radiological control technicians monitor and aid the performance of each individual's radiological work practices.
- b. Self-assessments of the Radiological Controls Program will be conducted.
- c. Independent assessments (in addition to NRC) of the Radiological Controls Program are conducted by technically qualified persons from outside the Radiological Controls Department. These audits are conducted in accordance with procedures as outlined in the Decommissioning Quality Assurance Plan (Reference 4). These audits are performed at least every 24 months.
- d. If a violation of an established procedure or practice is identified, the deficiency is recorded in Internal Assessments, Surveillance Inspection Reports, Corrective Action Program Process, or other reporting mechanisms. There is a reporting system for individuals to report deviations from good radiological practices.
- e. In the event of a radiological incident, an investigation shall be conducted to determine the causes of the incident and to determine the corrective actions and improvements needed.

4.5.8.3 Radiological Controls Training

- a. Periodic radiological controls training is given to ensure that each person understands the radiological conditions to which he is exposed, understands his responsibility to minimize his own exposure to radiation, and understands his own responsibilities for complying with radiological control procedures. Personnel occupationally exposed to radiation receive instruction on the effects of radiation and the risks associated with radiation exposure.
- b. General radiological indoctrination is given to those not directly involved with radiation so that they understand not to enter areas requiring **DLRs** and not to cross radiation barriers. The indoctrination includes explanation of the radiological environment in which they work.
- c. Those who require access to areas controlled by Radiation Work Permits receive training and are required to pass a radiological examination on their practical abilities, including use of dosimetry, frisking, anticontamination clothing, and response to unusual situations. In addition, those who need to use a respirator receive respirator training.

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Retraining and both written and practical examinations are conducted at least annually. In addition, spot checks are made to ensure that they retain the required knowledge during the period between examinations. Special briefings and extra training, including use of mockups where applicable, may be required for work involving higher than usual exposures to radiation and radioactivity.

4.5.8.4 Control of External Radiation Exposure

Control of radiation exposure is based on the assumption that any exposure, no matter how small, involves some risk; however, exposure within the accepted limits represents a risk small compared with normal hazards of life. Therefore, the policy of TMI-1 is to maintain exposures to individuals and total man-rem as low as is reasonably achievable (ALARA). Line management from all departments as well as each individual worker shall take an active role in radiation exposure reduction.

To aid in exposure reduction, administrative radiation exposure control levels shall be established. Radiation person-rem exposure goals shall be established for each major job and for each year. Work involving radiation exposure shall be preplanned. Major exposure jobs shall require that radiological controls be incorporated in the design, that written procedures be prepared, and that pre-job briefing and rehearsals be conducted prior to commencing work.

Restricted areas used to control personnel access to radiation and radioactive materials are defined, access controlled, and posted in accordance with 10CFR20 except as allowed by Technical Specification step 6.12.1. Radiological Controls personnel are exempt from the RWP issuance requirements during the performance of their assigned radiation protection duties providing they are following radiological control procedures for entry into High Radiation Areas.

To evaluate radiological conditions, radiation surveys are conducted for air activity, removable surface contamination, and external radiation at regular intervals. Surveys are performed in order to (1) monitor the suitability of control measures, (2) evaluate the needs for additional controls, (3) evaluate trends for ALARA purposes, and (4) evaluate radiological conditions in areas routinely entered without radiation work permit coverage. Surveys in unrestricted areas are provided to ensure the effective control of radioactive material. Unusual conditions detected in the performance of either a routine or special survey are immediately brought to the attention of Radiological Controls management.

Portable radiation survey instruments are calibrated at least annually, except for direct reading dosimeters which will be calibrated at least semiannually, to assure a consistent, reliable, and predictable response to radiation levels. Records of surveys are maintained on file. An administrative program is used to verify the calibration of personnel and field monitoring instruments.

4.5.8.5 Control of Internal Radiation Exposure

It is the policy of the licensee to prevent significant unplanned exposure to personnel from radioactivity associated with TMI-1. Intakes may be planned consistent with TEDE-ALARA concepts and other occupational health considerations. (TEDE – Total Effective Dose Equivalent).

4.5.8.6 Control of Radioactive Contamination

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Radioactive surface contamination is controlled to minimize possible inhalation or ingestion of radioactivity and to minimize buildup of radioactivity in the environment. Measures are taken to contain radioactivity and to minimize the number and extent of areas contaminated to minimize personnel radiation exposure, to simplify subsequent personnel and area or facility decontamination, and to minimize the need to rely on anticontamination clothing.

The surface contamination level release limit for uncontrolled use of items is 5000 dpm/100cm² for total beta-gamma contamination, 300 dpm/100cm² for total alpha contamination, 1000 dpm/100cm² for smearable beta-gamma contamination, and 20 dpm/100cm² for smearable alpha contamination.

The skin contamination action level is ≥ 100 counts per minute above background, as indicated by a pancake probe frisk. (This is equivalent to 5000 dpm/100cm² beta-gamma.)

Emphasis is placed on planning, training, and working in order to minimize the number of occurrences of radioactive surface contamination of a person's skin or of areas not controlled for radioactive surface contamination, and the amount of radioactivity involved. Each such occurrence is reviewed in detail to determine how to correct deficiencies and improve control of radioactivity.

4.6 RADIOACTIVE MATERIALS SAFETY

4.6.1 MATERIALS SAFETY PROGRAM

4.6.1.1 Control of Radioactive Materials

The following program will be implemented to assure the safe storage, handling, and use of sealed and unsealed special nuclear, source, and byproduct materials. In addition to the definition of 10CFR20, any material with surface contamination in excess of the limits specified in Section 4.5.8.6 shall be handled as radioactive. A radioactive material control system shall be established to ensure radioactive material is not lost or misplaced in a location where personnel could unknowingly be exposed to radiation and to prevent the uncontrolled spread of radioactivity to areas where the public might be affected.

This system shall include the following requirements:

- a. The number of areas in which radioactive materials are stored shall be minimized.
- b. Any long term radioactive materials storage area shall be approved by the Manager of Radiation Protection and Chemistry (or specified designee).
- c. The numbers of radioactive items and the amount of radioactivity in storage shall be minimized.
- d. Radioactive items shall be identified as radioactive before removing them from a restricted area.
- e. Radioactive materials removed from the Security Area or removed from a restricted area outside the Security Area shall be controlled in accordance with an accountability procedure which ensures the materials are not lost or

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improperly handled during transfer or subject to unauthorized removal. This accountability procedure shall require inventory of radioactive materials which remain outside such areas.

- f. Each incoming or outgoing shipment of radioactive material shall be handled in strict compliance with detailed written procedures.

Each case in which radioactive material is lost or unaccounted for shall be reviewed in detail to determine the potential radiation exposure personnel might unknowingly receive, to correct deficiencies, and to improve control of radioactive materials.

4.6.1.2 Receipt of Radioactive Sources

The method of receipt, date, and description of the contents will be recorded. Each radioactive sealed source will be surveyed for removable contamination prior to use. Records of the survey will be maintained. Each sealed source and/or source container will be smeared using cotton swabs or small filter papers. These smears will be counted in the counting laboratory for alpha and/or beta gamma activity, as appropriate. If leakage is confirmed from any source, the source will be repaired or replaced.

Each source will be numbered and recorded in the Source Accountability System.

4.6.1.3 Storage of Radioactive Sources

Each source will have an assigned storage location and an assigned custodian responsible for the security of the source.

4.6.1.4 Source Preparation and Handling

Only personnel trained in Radiological Controls practices are permitted to handle radioactive sources. These personnel will wear dosimeters/TLDs while handling radioactive sources as required by a Radiation Work Permit. When radioactive material containers are opened, a radiation and contamination survey will be taken as appropriate.

Protective clothing, consisting of rubber gloves, lab coats, and shoe covers, will be worn, as required, during the preparation of the radioactive calibration sources.

All liquid or gaseous radioactive sources will be opened under a ventilation hood. The area where the opened radioactive material is handled will be appropriately marked and posted. Radiological Controls practices, appropriate to the material being handled, will be followed.

Standards prepared at the station will be carefully made so as to limit the spread of radioactivity. For liquid sources, a cover is provided for the source container to prevent spillage during transport from the source storage area to the source preparation area and vice versa.

A smear sample will be taken of each sealed source for detection of leaks or removable contamination. The detection limit of instrumentation used should be able to detect the presence of 0.005 uCi of removable contamination.

4.6.1.5 Disposal of Radioactive Waste Resulting from Source Preparation

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Liquid waste, which is expected to result from source preparation, will be stored in a radioactive waste storage area prior to disposal.

Solid waste consisting of absorbent paper and kimwipes, etc., will be handled in accordance with solid waste handling procedures.

4.6.1.6 Radiological Surveys

A radiation survey and a survey for contamination will be conducted in the laboratory source preparation area after the sources are prepared. In addition, a radiation and contamination survey will be conducted once per week for laboratory areas in use. Leak testing for radioactive surface contamination will be conducted in accordance with 10CFR35.67. Sealed sources are exempt from such leak tests when the source contains 100 μCi or less of beta and/or gamma emitting material or 5 μCi or less of alpha emitting material. The survey detection limit of contamination must be equal to or less than 0.005 μCi .

Routinely scheduled radiation and contamination surveys will be conducted in and around the source storage areas.

4.6.1.7 Source Inventory

A source inventory will be maintained current in accordance with 10CFR35.67. Records of the inventory will be kept by Radiological Controls.

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4.6.2 FACILITIES AND EQUIPMENT

Three Mile Island Unit 1 includes a Radiological Control Laboratory with facilities and equipment for detecting, analyzing, and measuring all types of radiation and for evaluating radiological problems which may occur. Counting equipment (such as G-M, scintillation, and proportional counters) are provided in an appropriately shielded counting room for detecting and measuring all types of radiation.

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4.7 REFERENCES

1. TMI OFFSITE DOSE CALCULATION MANUAL (ODCM), CY-TM-170-300.
2. TMI-1 TECHNICAL SPECIFICATIONS.
3. **DELETED.**
4. NO-DC-10, DECOMMISSIONING QUALITY ASSURANCE PLAN
5. REGULATORY GUIDE 1.109, CALCULATION OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10 CFR PART 50, APPENDIX I"
6. REGULATORY GUIDE 1.110 "COST-BENEFIT ANALYSIS FOR RADWSOST-BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS"
7. REGULATORY GUIDE 1.111
8. REGULATORY GUIDE 1.112
9. 49 CFR 173, GENERAL REQUIREMENTS FOR SHIPMENTS AND PACKAGINGS OF HAZARDOUS MATERIALS
10. 10 CFR PART 61, LICENSING REQUIREMENTS FOR LAND DISPOSAL OF RADIOACTIVE WASTE
11. 10 CFR PART 71, PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL
12. SUTRON REPORT NO. SCR-358-82-063, "STUDY OF TRAVEL TIME AND MIXING CHARACTERISTICS FOR THE SUSQUEHANNA RIVER BELOW THREE MILE ISLAND," THE SUTRON CORPORATION, JULY 1982.
13. **DELETED**
14. **DELETED**

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CHAPTER 5 – CONDUCT OF OPERATIONS

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5.0 CONDUCT OF OPERATIONS

5.1 ORGANIZATIONAL STRUCTURE

Constellation Energy Generation, LLC (the licensee) is a limited liability company responsible for the safe, reliable, and efficient operation of its nuclear facilities. In addition, the license is responsible for appropriate standards, programs, processes, management controls, and support for the nuclear facilities. In keeping with these responsibilities, the licensee is committed to providing sufficient personnel having appropriate qualifications to both operate and technically support the facility.

The overall responsibility for operation, maintenance, inspection, test, modification, decommissioning, and storage of spent fuel resides with the Sr. Vice President, Constellation Energy Generation, LLC and President and Chief Nuclear Officer (CNO). The TMI1 Site Decommissioning Director is responsible for the administration and implementation of the Decommissioning Quality Assurance Program (DQAP) at Three Mile Island Unit 1. The overall organizational structure and reporting relationships are defined in the Decommissioning Quality Assurance Plan (Reference 1).

5.1.1 OFFSITE ORGANIZATION

The licensee corporate organization, functions and responsibilities are described in Chapter 1 of the Decommissioning Quality Assurance Plan (Reference 1).

5.1.2 STATION MANAGEMENT

A decommissioning management position shall be responsible for overall safe operation of the facility and shall have control over those onsite activities necessary for safe storage and maintenance of spent nuclear fuel, including maintaining the facility within the constraints of applicable regulatory requirements, license, DC technical specifications and training.

This management position or specified designee, shall approve, prior to implementation, all tests, experiments, and modifications to systems or equipment that affect the safe storage and maintenance of spent nuclear fuel. Supervisory direction is provided for the technical review program, including approval of individuals as technical reviewers as applicable.

Dependent on the decommissioning status the functional areas will be supported by management individuals either assigned to the facility or in the corporate organization who perform the following under the jurisdiction and direction of this position. In some cases, a single management position may support multiple responsibilities.

A. Qualification of Plant Personnel

Each member of the facility staff shall meet or exceed the minimum qualifications specified in the DQAP (Reference 1).

B. Safety Reviews

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The safety review process ensures the quality of procedure changes, Technical Specification changes, Licensee Event Reports (LERs), plant modifications, and other documents within the scope of the DQAP. The process requires all management personnel to control the preparation, review, implementation and reporting activities in their area. The subjects addressed include operating procedure changes, plant hardware modifications, security and radiological and environmental control activities, etc. All aspects of nuclear plant design and operation that are important to safe operation (i.e., within the scope of the DQAP) utilize the safety review process. "Station Qualified Reviewers (SQR)" are charged with reviewing the document for safety and technical adequacy. They must be knowledgeable and experienced individuals, different from the preparer, but may be from the same organization as the individual who prepared the document. The SQR must be qualified in accordance with the provisions of the DQAP.

The Safety Review Committee (SRC) is an on-site review body that reviews changes to design, operation, or operating experience for decommissioning activities which may affect Nuclear Safety and makes recommendations to the Site Decommissioning Director.

5.1.2.1 Operating Department

A management position is responsible for Operations, with responsibility for operating strategies that support nuclear and personnel safety within the constraints of the decommissioning license and regulatory requirements.

5.1.2.2 Radiation Protection, Chemistry and Environmental

A management position is responsible for ALARA planning activities
A management position is responsible for chemistry activities.
A management position is responsible for environmental activities

The radiation protection and radioactive material programs are described in DSAR section 4.

5.1.2.3 Security

A management position is responsible for implementation of the site security plan. The security program is described in DSAR section 5.8.

5.1.2.4 Regulatory Assurance and Emergency Preparedness

A management position is responsible for maintaining an interface between the facility and federal and state regulators. A management position is responsible for Emergency Preparedness. The Emergency Preparedness program is described in DSAR section 5.7.

5.1.2.5 Engineering

A management position is responsible for engineering support activities, development and maintenance of engineering programs, policies, procedures, and providing

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engineering services in accordance with the DQAP. The position is also, responsible for document control and records management functions.

The Design Control program is described in DSAR section 5.4.

5.1.2.6 Decommissioning Project Management

A management position is responsible for managing decommissioning projects that support nuclear and personnel safety within the constraints of the decommissioning license and regulatory requirements. This includes the routine maintenance and repair functions.

A. Routine Maintenance and Repairs

Procedures for performing maintenance and repairs are prepared and maintained by the unit staff and reviewed and approved through the approved Safety Review process and the 10CFR50.59 review process, as applicable.

The written repair and maintenance procedures specifically indicate the inspections and checks which must be performed, and also indicate the records and data which must be kept. The procedures indicate where independent verification of inspections or checks should be performed.

Outside personnel may be brought in to supplement the unit personnel but will work under the supervision of the unit personnel.

Materials and parts utilized in the repair and maintenance of nuclear-related portions of the unit will be of the same quality as, or better than, the original materials, for SSC with the original design function. The procurement documents will be selectively reviewed in accordance with the DQAP (Reference 1) to ensure that appropriate quality control requirements are fulfilled. Storage and material-identification procedures assure that purchased materials and parts do not deteriorate in storage and are properly identified prior to their installation or use.

A management position reviews all maintenance records to ensure that required maintenance is performed and that the necessary records of maintenance and repairs are kept.

5.2 TRAINING

5.2.1 TRAINING PROGRAMS

The training programs are maintained for aspects of operator training, technical training, and support personnel. Training programs utilize formal objectives and written lesson plans. Line management ensures that the content, test material, mode of presentation, and schedule are appropriate.

5.2.2 TECHNICAL TRAINING PROGRAM

Deleted

5.2.3 OPERATOR TRAINING PROGRAM

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Deleted

5.2.4 GENERAL EMPLOYEE TRAINING PROGRAM

General Employee Training is conducted for all non-visitor personnel (company and contractors) at Three Mile Island Unit 1. All personnel receive instruction on topics such as use of procedures, security, emergency preparedness, quality assurance, industrial safety, worker and community right to know laws, fitness for duty, radiation effects and risks and basic radiation protection. Employees who enter radiation work permit required areas receive additional training (refer to DSAR section 4.5.8) on topics such as federal standards, and radiological protection practices and procedures. Employees required to wear respiratory protection devices receive training in respiratory protection.

Practical factors training is conducted for radiation workers and for wearers of respiratory protection devices. In order to maintain continuing access to the specified areas of the station, all workers must successfully complete an appropriate refresher program.

5.2.5 FIRE PROTECTION TRAINING PROGRAM

The Fire Protection Training Program is designed to meet the requirements of Section 600 of the NFPA Code-1996, and the Fire Protection Program as described in Section 3.7.

Fire Watch Training is conducted annually for those personnel who are assigned "hot work" fire watch duties. This training includes instruction in fire watch duties/responsibilities, basic fire chemistry, fire classification, and the type and operation of portable fire extinguishers.

5.2.6 EMERGENCY PREPAREDNESS TRAINING PROGRAM

Emergency preparedness training is intended to prepare all personnel to successfully perform their emergency duties as outlined in the Emergency Plan and the Emergency Plan Implementing Document.

All personnel receive a familiarization with the salient features of the Emergency Plan as part of General Employee Training. The training includes identification of specific emergency conditions and associated alarms, and proper employee responses. Retraining is conducted annually.

Personnel who have specific emergency related duties receive specialized emergency preparedness training. That training is tailored to the specific emergency duty positions and emergency response roles. Retraining is conducted annually.

5.2.7 SECURITY TRAINING PROGRAM

The Security Force Training Program is designed to train the site protection personnel to perform their assigned duties as identified in the Security Personnel Training and Qualification Plan. Security Force Training is presented as an initial training program and an annual retraining program. Initial training includes topics on basic security skills, site security procedures, the Safeguards Contingency Plan, specific security tasks identified in the Security Personnel Training and Qualification Plan, and weapons training. Annual retraining provides refresher training on special topics of security skills, security plan and procedures, assigned

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security tasks, and weapons qualification at least once per year. Security Force Training is accomplished through classroom instruction, hands-on performance, demonstrations, and practical field exercises.

5.3 PROCEDURES

Written procedures are established, implemented and maintained. Independent checks and audits are conducted to verify that the procedures are followed.

Changes to procedures are subject to the requirements of section 5.1.2.B and 10CFR50.59 as applicable.

5.3.1 EMERGENCY AND ALARM RESPONSE PROCEDURES

Emergency procedures to cope with failure of system components are reviewed by the Safety Review process and approved by the Operations Lead. In addition to the emergency procedures, alarm response procedures are written to convey to the operator the necessary, immediate corrective action required to establish a safe condition. Operators are required to demonstrate from memory their familiarity with the immediate actions of emergency procedures.

5.4 DESIGN CONTROL

The Manager of Engineering is responsible for controlling design changes and the administering of design control activities including design interfaces for modifications that affect nuclear safety of structures, components, and systems. Proposed modifications to the unit are prepared by Site Engineering, Corporate Engineering, or an authorized contractor, and are reviewed by the Safety Review process. The Site Decommissioning Director or specified designee is responsible to authorize design changes.

Plant modifications are designed in accordance with approved Engineering procedures. The procedures comply with the DQAP and other applicable codes and standards. The same (or upgraded) codes and requirements as used in the original plant design are applied to modifications.

Plant modifications are installed by unit staff or outside personnel under TMI Unit 1 supervision and/or management. The installation of modifications is accomplished in accordance with approved engineering design change package and approved implementing procedures. The procedures comply with the DQAP and ensure:

- a. Adequate post installation testing;
- b. Compilation of installation records and record retention and;
- c. Monitoring of installation by the Quality Verification group.

Where the installation of modifications is performed by an outside contractor, the contractor may utilize the TMI procedures and the DQAP or may utilize their defined quality control plan and construction procedures. The contractor's defined quality control plan and construction procedures must comply with intent of the above and be approved by Nuclear Oversight.

5.5 RECORDS

Records of plant activities shall be maintained as described in the DQAP (Reference 1).

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5.6 QUALITY ASSURANCE

Activities are conducted in accordance with the NRC-approved Decommissioning Quality Assurance Program (DQAP) which describes the formal and comprehensive control methods established to assure compliance with 10CFR50 Appendix B. The DQAP describes how the Quality Assurance Program is to be functionally implemented with due regard to the safety and health of the public and the personnel on site. The DQAP is incorporated into the DSAR by reference.

5.7 EMERGENCY PLAN

The prime objectives of emergency planning are to: (1) Develop a plan and implementing procedures that will provide the means for mitigating the consequences of emergencies (including very low probability events) in order to protect the health and safety of the general public and site personnel and to prevent damage to property, and (2) Ensure operational readiness of emergency preparedness capabilities.

The Emergency Plan for Three Mile Island Nuclear Station assures that all emergency situations, including those which involve radiation or radioactive material are handled logically and efficiently. It covers the entire spectrum of emergencies from minor, localized emergencies to major emergencies involving action by offsite emergency response agencies and organizations.

The Emergency Plan Implementing Document provides a single source of pertinent and significant information and data and the procedures that would be required by or useful for various emergency response agencies and organizations in the event of an emergency.

The Emergency Plan Implementing Document consolidates and integrates specific material detailed in documents such as the Emergency Plan, the State Plans, and the Various County Plans.

This Emergency Plan has been developed in accordance with the provisions of 10CFR50, Appendix E, and 10CFR50.47 and is consistent with the guidelines given in (1) Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3 and (2) "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG 0654/FEMA-REP-1, dated November 1, 1980.

For emergencies, the control room (**or the ISFSI security building**) has a communication system which includes a dedicated telephone system that connects TMI with NRC headquarters. This is identified as the Emergency Notification System (ENS). The ENS is used for reliably reporting emergencies to the NRC.

5.8 PHYSICAL SECURITY

The physical security program for the TMI ISFSI is in accordance with the applicable portions of 10 CFR Part 72.212 (Modified 10 CFR Part 73.55). Upon placing all TMI-1 Greater than Class C Waste on the ISFSI pad, a stand-alone 10 CFR Part 37 Security Plan provides physical protection of Category 1 and Category 2 quantities of radioactive material in TMI-1.

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5.9 REFERENCES

1. NO-DC-10, Decommissioning Quality Assurance Program (DQAP)
2. TMI-1 Technical Specifications
3. NRC Letter to Bryan Hanson, December 29, 2017, Approval of Certified Fuel Handler Training and Retraining Program (ML17228A729).
4. 10CFR50.120 "Training and qualification of nuclear power plant personnel"

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CHAPTER 6 – SAFETY ANALYSIS

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LIST OF FIGURES – None

6.0 **SAFETY ANALYSIS**

All fuel has been moved to the ISFSI. There are no credible accidents for the 10 CFR 50 facility.

6.1 **DELETED**

6.2 **DELETED**

6.3 **DELETED**

6.4 **DELETED**

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TABLE 6.2-1 - **DELETED**

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