REVISION 11.2

MILLSTONE POWER STATION UNIT 1
DEFUELED SAFETY ANALYSIS REPORT

REVISION 11.2

THIS DOCUMENT INCORPORATES APPROVED CHANGES TO THE MPS-1 DSAR.

REFER TO THIS DOCUMENT'S REVISION HISTORY FOR PARTICULARS.

Revision History

The NRC's "Guidance for Electronic Submissions to the NRC", in Section 2.2, Living Document Updates, requires that submittals: "...indicate the part(s) (e.g., chapter, section, or graphic) that has been changed as well as the general scope of the change." The Millstone Unit 1 DSAR's Revision History fulfills this requirement.

REPORTING PERIOD 2009 - 2010

REVISION	FSC PKG Document Number	DATE	SECTION	Summary Description of Changes
7	--	04/09	As identified in the 2009 NRC Submittal List of Changed Pages and submitted Summary of Change.	Administrative (FSAR content not affected). Change indicator (s) and page change identification (s) present in the 2009 NRC Submittal removed in preparation for the 2010 NRC Submittal. This forms the base line for changes incorporated under the Revision 7 series. Revision level of the authoring files are unchanged. This supports Revision/Change traceability.

Revision 8 (2010 – 2011 Reporting Period)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes
8	March 2011	MP1-DFCR-2010-001	S3.2.7.2	Reflects change to substation nomenclature (Northeast Utilities Distribution Project) for off site power source.

Revision 9, ((2011-2012 Reporting Period)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes
9	June 2011	Administrative	As identified in the previous NRC Submittal List of Changed Pages and submitted Summary of Change.	Administrative (FSAR content not affected). Change indicator (s) and page change identification (s) present in the 2011 NRC Submittal removed in preparation for the 2012 NRC Submittal. This forms the base line for changes incorporated under the Revision 9 series. Revision level of the authoring files are unchanged. This supports Revision/Change traceability.

Revision 10, (2012-2013 Reporting Period)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes
10	July 2010	Administrative	As identified in the previous NRC Submittal List of Changed Pages and submitted Summary of Change.	Administrative (FSAR content not affected). Change indicator (s) and page change identification (s) present in the previous NRC Submittal removed in preparation for the 2013 NRC Submittal. This forms the base line for changes incorporated under the Revision 10 series. Revision level of the authoring files are unchanged. This supports Revision/Change traceability.
10	July 2012	MP1-DFCR-2011-001	S1.2.3.2.1, S4.1, S4.4	Reflects updates for compliance with NPDES permit.

Revision 11, (2013-2014 Reporting Period)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes
11	July 2013	Administrative	As identified in the previous NRC Submittal List of Changed Pages and submitted Summary of Change.	Administrative (FSAR content not affected). Change indicator (s) and page change identification (s) present in the previous NRC Submittal removed in preparation for the 2014 NRC Submittal. This forms the base line for changes incorporated under the Revision 11 series. Revision level of the authoring files are unchanged. This supports Revision/Change traceability.
11.1	9/13/13	Administrative	List of Figures	Reflects administrative clarification of revision status for engineering controlled drawings that are coincidently FSAR Figures. FSAR figures corresponding to Controlled P&ID's are updated on a periodic basis by the FSAR Coordinator.
11.2	11/15/13	MP1-DFCR-2013-001	List of Figures	Reflects administrative clarification of revision status for engineering controlled drawings that are coincidently FSAR Figures. FSAR figures corresponding to Controlled P&ID's are updated on a periodic basis by the FSAR Coordinator.

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CHAPTER 1–INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Defueled Safety Analysis Report (DSAR) is submitted by the licensee in support of the decommissioning for Millstone Unit Number 1 at the Millstone Nuclear Power Station in Waterford, Connecticut. Dominion Nuclear Connecticut, Inc. owns and is responsible for the decommissioning of Millstone Unit Number 1.

The DSAR is the principle licensing source document describing the pertinent equipment, structures, systems, operational constraints and practices, accident analyses, and decommissioning activities associated with the existing defueled condition of Millstone Unit Number 1. As such, the DSAR is intended to serve in the same role as the Final Safety Analysis Report of Millstone Unit Number 1 during the periods of power operation between 1970 and 1998. The DSAR is applicable throughout the decommissioning of Millstone Unit Number 1. The decommissioning process is dynamic. The issuance of the DSAR does not alleviate the licensee from continuing to follow all required surveillances, procedures, technical specifications or similar documents, until those documents are officially modified using approved processes. Drawings and figures of structures, systems, or components (SSCs) included or referenced in the DSAR, are included within the licensing basis of the facility only to the extent that they show SSCs that are described in the text of the DSAR. Other contents of drawings and figures may not reflect the current configuration of the facility and are not maintained.

Construction of Millstone Unit Number 1 was authorized by a provisional construction permit CPPR-20, on May 19,1966, in AEC Docket 50-245. Millstone Unit Number 1 was completed and ready for fuel loading during October 1970. The plant went into commercial operation on December 28, 1970. On July 21, 1998, pursuant to 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii), the licensee certified to the NRC that, as of July 17, 1998, Millstone Unit Number 1 had permanently ceased operations and that fuel had been permanently removed from the reactor vessel. The issuance of this certification fundamentally changes the licensing basis of Millstone Unit Number 1 in that the NRC-issued 10 CFR 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel. Therefore, as of July 21, 1998, only those conditions or activities associated with the safe storage of fuel and radiological protection (including waste handling, storage and disposal) are applicable to the defueled Millstone Unit Number 1 plant.

Millstone Unit Number 1 was a single cycle, boiling water reactor with a Mark I containment which was designed, furnished and constructed by General Electric Company as prime contractor for the licensee. The General Electric Company engaged Ebasco Services Incorporated as architect-engineer. Millstone Unit Number 1 had a reactor thermal output of 2011 megawatts and a net electrical output of 652.1 megawatts. The Millstone site is located in the town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound.

TABLE 1.1–1 MILLSTONE UNIT NO.1 LICENSING MILESTONES

EVENT	DATE
Construction Permit Issued	May 19, 1966
FSAR Filed	November 1, 1968
Provisional operating License Issued	October 7, 1970
Full-Term Operating License Issued	October 31, 1986
Full Power License	October 7, 1970
Initial Criticality	October 26, 1970
Synchronized to the Grid	November 1970
100 Percent Power	January 6, 1971
Commercial operation	December 28, 1970
Permanently Ceased Operations	July 21, 1998

1.2 GENERAL PLANT DESCRIPTION

1.2.1 PLANT SITE AND ENVIRONS

1.2.1.1 Location and Site

The site for the Millstone Nuclear Power Station consists of a tract of land of approximately 500 acres located in the town of Waterford, Connecticut on the north shore of Long Island Sound and on the east side of Niantic River Estuary. It is located 3.2 miles west-south-west of New London, and 40 miles south-east of Hartford, Connecticut. The site is bounded on the west, south, and portions of the east sides by Long Island Sound. The nearest residential boundary is 855 meters north-east of the major structures of Millstone Unit Number 1. Chapter 2 contains more detailed information on the site and surrounding areas.

1.2.1.2 Site Ownership

The site is owned by Dominion Nuclear Connecticut, Inc.

1.2.1.3 Access to the Site

The immediate area around the station, excluding the intake and discharge canal, is completely enclosed by a security fence. This fence establishes the protected area boundary of the station. Access to the station is controlled by Security Personnel.

1.2.1.4 Description of the Environs

Adjacent to the site to the north and west is cultivated land with residential dwellings. The village of Niantic, consisting of a small commercial complex and attendant residential development, is 1.5 miles north-west of the Reactor Building. Other residential areas adjoin the site at the end of the plant access road and at distances of 1 to 3 miles.

New London, 3.2 miles ENE of the Reactor Building, is the nearest urban complex and includes mixed residential, commercial, and industrial uses.

1.2.1.5 Geology

The site area is underlain by Monson gneiss and Westerly granite. The Westerly granite intrudes the Monson gneiss, is more resistant to weathering and therefore forms ridges. Seismic surveys disclosed no unusual or extreme subsurface conditions. Chapter 2 contains more detailed information on geology and seismic qualities.

1.2.1.6 Seismology and Design Response Spectra

The Millstone Point site area is placed in Zone 2 (zone of moderate damage) on the seismic probability map of the 1964 Uniform Building Code.

The seismic design for critical items for this station is based on dynamic analysis of acceleration or velocity response spectrum curves which are based on a ground motion of 0.07g.

The preceding design criteria are for critical items only, that is, for Class I items. Class I items are defined in Chapter 3.

1.2.1.7 Hydrology

The plant site natural grade level is at an elevation of approximately 14 feet above mean sea level.

Because of the contours of the land and ground strata, and the distance of the reactor from water supplies, no water accidentally released from the plant can reach industrial or drinking water supplies.

Chapter 2.0 contains more detailed information on hydrology.

1.2.1.8 Meteorology

The meteorology of the site area is basically that of a sea-coast location with relatively favorable atmospheric dilution conditions prevailing. The inland terrain in Connecticut is not pronounced enough to produce any significant local modifications of synoptic conditions at the shoreline. The shoreline areas do, however, experience local modifications of synoptic patterns because of the temperature differences between air over land and air over water.

The site is located in an area occasionally traversed by hurricanes. The design basis hurricane for Millstone has 124 mph maximum gradient winds and a 17 mph speed of translation. This is significantly more intense than the worst on record (hurricane of 1938).

It has been estimated that a tornado can be expected to strike a point on the Millstone site about every 1,804 years. In spite of this low probability, the features of the plant important to the safe storage of irradiated fuel have been designed to withstand 300 mph winds.

It is concluded that from the viewpoint of site meteorology, the site is suitable for the station as described. (Chapter 2 contains more detailed information concerning meteorology.)

1.2.1.9 Site Environmental Radioactivity Monitoring Program

An environmental radioactivity monitoring program was initiated and has been conducted at the site since April 1967. Data are collected to measure radioactivity present in the environs. The program is continuing in order to assure prompt detection and evaluation of any changes in radioactivity.

1.2.2 SUMMARY PLANT DESCRIPTION

The plot plan (Figure 1.2-1) shows the general arrangement of Millstone Unit Number 1 on the Millstone Point site. The reactor building houses fuel storage facilities, refueling equipment and other auxiliary equipment.

The Radioactive Waste Building, located northeast of the Reactor Building, is a two story concrete structure. The overall arrangement of this building is shown in Figures 1.2-2 and 1.2-3.

1.2.3 SYSTEMS

1.2.3.1 Fuel Storage and Fuel Handling

1.2.3.1.1 Fuel Storage and Handling Equipment

The spent fuel storage pool holds fuel assemblies, control rods, and small vessel components. The pool system contains provisions to maintain water cleanliness and instrumentation to monitor water level. Makeup water is available from the Unit 2 demineralized water system and the fire water system. The racks in which fuel assemblies are placed are designed and arranged to ensure subcriticality in the pool.

The handling of spent fuel is performed within the Reactor Building. This employs a refueling platform for underwater fuel transport, storage racks for fuel and control rods in a storage pool, underwater fuel preparation stations, and floor mounted jib cranes. Control rods can be stored in the fuel pool racks or on hooks on the side of the pool.

Structural design of the fuel storage and equipment storage facilities meets all requirements for Class I structures. For additional information, refer to Chapter 3.

1.2.3.1.2 Fuel Pool Cooling System

The fuel pool cooling system provides cooling for the spent fuel pool water when required.

The fuel pool cooling system consists of a circulating pump, heat exchanger, skimmer surge tanks, system piping, valves, and instrumentation and controls. Pool cleanup is provided by an in-pool demineralizer and filter. For additional information, refer to Chapter 3.

1.2.3.2 Radioactive Waste Processing System

The radioactive waste processing system is designed to control the release of plant-produced radioactive material to within the limits specified in 10 CFR 20 and Appendix I to 10 CFR 50. This is done by collection, transfer, and evaporation.

1.2.3.2.1 Liquid Radwaste System

Liquid waste drained or transferred to the Reactor Building sumps will be processed using the Waste Water Processing System (WWPS) or using an atmospheric evaporator. The Waste Water Processing System consists of four (4) 10,000 gallon Sample Tanks, recirculation pump, demineralizer, filters and associated piping. The "A" RBFD sump will pump to the WWPS sample tanks, where the water will be batch recirculated and sampled before subsequent discharge. Radiological monitoring will be conducted using an in-line Liquid Effluent Monitor (RE-MG-110). Prior to discharge through DSN-001A (Emergency Service Water discharge piping to discharge canal), dilution flow requirements will be established by crediting Unit 2 Circulating Water Flow to the common discharge canal. Alternatively, this system could be utilized to pump the process liquids from the Reactor Building sumps to containers which would permit the process liquid to be processed onsite or offsite. 11-1

1.2.3.2.2 Solid Radwaste Handling

Solid wastes originating from nuclear system equipment maybe stored in the spent fuel storage pool and prepared for off site shipment in approved shipping containers.

Solid wastes are collected and appropriately prepared for off site shipment. Examples of these solid wastes are filter residue, spent resins, paper, air filters, rags, and used clothing. For additional information, refer to Chapter 4.

1.2.3.3 Radiation Monitoring and Control

1.2.3.3.1 Radiation Monitoring and Sampling

The Spent Fuel Pool Island ventilation exhaust is monitored for gaseous radiation and particulates. A particulate sampling skid is provided for Unit 1 Balance of Plant (BOP) exhaust to permit sampling for any significant changes. For additional information, refer to Chapter 4.

1.2.3.3.2 Area Radiation Monitors

Radiation monitors are provided to monitor for abnormal radiation at selected locations on the SFPI. These monitors actuate alarms when abnormal radiation levels are detected.

1.2.3.3.3 Liquid Radwaste Processing System Control

The liquid radwaste system is designed to safely and economically collect, store, process, and dispose of, or recycle, all radioactive or potentially radioactive liquid waste generated. The system operates on a batch basis.

1.2.3.3.4 Solid Radwaste Control

Solid radwaste can be transferred to high integrity cask containers for shipment.

1.2.3.4 Auxiliary Systems

1.2.3.4.1 Decay Heat Removal (DHR) System

The DHR system provides cooling water to the spent fuel pool cooling system. The system consists of circulating pumps, air-water heat exchangers, an expansion tank, air separator and associated piping valves and controls, and a portable ethylene glycol addition pump and tank.

1.2.3.4.2 Monitoring and Control Functions

The Millstone Unit 2 Control Room is continuously manned, and serves as the control room for Millstone Unit 1. Millstone Unit 2 Operations personnel are responsible for the monitoring and control of the Unit 1 spent fuel pool island (SFPI) and auxiliary systems via a computer console located in the Millstone Unit 2 Control Room.

1.2.3.4.3 Fire Protection System

Fire protection and detection systems are provided at Millstone Unit Number 1 to protect structures, systems, and components important to the defueled condition of the unit.

The fire protection system includes a fire water supply system that consists of two fire water tanks, fire water pumps and a distribution system that delivers fire water to all parts of the plant.

Fire water systems within the plant protect individual hazards and include sprinkler systems and deluge systems.

1.2.3.4.4 Electrical Power System

1.2.3.4.4.1 AC Power Supply

The electric power system includes the electrical equipment and connections required to supply power to station auxiliaries.

1.2.3.4.4.2 DC Power Supply

The SFPI 125 V DC system is provided via rectified AC at the point of use. In addition, a separate decommissioning 125V DC system powered by batteries and a battery charger provides a source of DC power to the decommissioning electrical system.

SFPI 24V power is provided by power supplies within the SFPI Programmable Logic Controller (PLC) panels.

1.2.3.5 Station Communication System

The plant communication system provides for reliable on site and off site communications both under normal and contingency conditions.

1.2.3.6 Station Water Purification, Treatment and Storage System

This system provides demineralized makeup water to Millstone Unit Number 1 for use in the spent fuel pool.

1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.3.1 APPLICANT'S SUBSIDIARIES

Dominion Nuclear Connecticut, Inc. is responsible for the decommissioning of Millstone Unit Number 1. Dominion Nuclear Connecticut, Inc. is a wholly owned subsidiary of Dominion Energy, which is wholly owned by Dominion Resources, Inc..

1.3.2 NUCLEAR STEAM SUPPLY SYSTEM SUPPLIER

General Electric Company was the nuclear steam system supplier for the plant.

1.3.3 ARCHITECT/ENGINEER

Ebasco Services Incorporated was the Architect/Engineer for Millstone Unit Number 1.

1.3.4 TURBINE-GENERATOR SUPPLIER

The turbine generator was manufactured by General Electric Company.

1.4 MATERIAL INCORPORATED BY REFERENCE

Specific sections of the Millstone Unit Number 2 and Number 3 FSARs are incorporated into the Unit 1 DSAR by reference. These sections are identified within the text of the DSAR.

1.5 CONFORMANCE TO NRC REGULATORY GUIDES

1.5.1 SUMMARY DISCUSSION

The AEC issued Appendix A ‘General Design Criteria’ to 10 CFR 50 in July 1971. In November 1970, Safety Guides, later to become Regulatory Guides, began to be published. These guides provided acceptable means for complying with specified general design criteria. They were not in effect at the time Millstone Unit Number 1 began operation with Provisional Operating License (POL) DPR-21, issued October 7, 1970.

Millstone Unit Number 1 submitted summaries of compliance to these guides in the early 1970s in support of the application for a full-term operating license (Reference 1.5-1).

Before acting on this application, the NRC (formerly AEC) initiated the Systematic Evaluation Program (SEP) in 1977 to review the designs of older operating nuclear reactor plants in order to confirm and document their safety. Millstone Unit Number 1 was identified as an SEP plant.

The SEP objectives were:

- To establish documentation that shows how the criteria for each operating plant reviewed compare with current criteria on significant safety issues and to provide a rational for acceptable departures from these criteria.

- To provide the capability to make integrated and balanced decisions with respect to any required backfitting.

- To provide for early identification and resolution of any significant deficiencies.

- To assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

- To use available resources efficiently to minimize requirements for additional resources by NRC or industry.

- To ensure that the safety assessments were adequate for conversion of provisional operating licenses to full-term operating licenses.

The final version of the SEP program report included the status of all applicable generic activities (TMI and USIs), including those that formed the basis for the Integrated Safety Analysis Program (ISAP) being implemented by the Licensee. Based upon the acceptable conclusions reached in SEP, the NRC issued the full-term operating license for Millstone Unit Number 1 on October 31, 1986.

1.5.2 REFERENCE

1.5-1 Millstone Nuclear Power Station Unit 1 Application for Full-Term Operating License, September 1, 1972.

CHAPTER 2- SITE CHARACTERISTICS

2.1 LOCATION AND AREA

The Millstone site is located in the Town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound. The 524 acre site occupies the tip of Millstone Point between Niantic Bay on the west and Jordan Cove on the east and is situated 3.2 miles west-southwest of New London and 40 miles southeast of Hartford.

The Millstone Unit Number 1 containment structure is located immediately south of Millstone 2 and 3. The geographical coordinates of the centerline of the reactor is as follows:

Millstone Unit Number 1

Latitude and Longitude	Northing and Easting
N 41° 18'32"	N 173, 800
W 72° 10'04"	E 759, 965

The site is owned by Dominion Nuclear Connecticut, Inc. Figures 2.1-1 through 2.1-4 identify the site.

The site protected area is considered the restricted area. The restricted area has been conspicuously posted and administrative procedures, including periodic patrolling, have been imposed to control access to the area. For the purpose of radiological dose assessment of accidents, the exclusion area boundary (EAB) was considered the actual site boundary for overland sectors, except in the Fox Island / discharge channel area on the south end of the site. For all water sectors, the nearest land site boundary distance was used.

Any significant normal releases are discharged to the atmosphere via the Unit Number 1 BOP exhaust point and the SFPI ventilation exhaust point. The distance from the Unit Number 1 BOP exhaust point and the SFPI ventilation exhaust point to the nearest residential property boundary in the Millstone Point Colony development (Point A on Figure 2.1-3) is greater than 2,800 feet. This development, adjacent to the eastern site boundary, consists of single family homes on 104 half acre lots. One of the conditions of the sale of the site to the Hartford Electric Light Company and the Connecticut Light and Power Company was that permanent dwellings would never be permitted in the beach area of the development. Because of this restriction, normal release doses are calculated at Point A rather than at the nearest point on the site boundary.

The licensee has complete control of activities within the exclusion area, except for the passage of trains along the Providence & Worcester (P&W) / Amtrak Railroad track which runs east-west through the site.

To ensure the safety of people within the exclusion area during an emergency, an emergency plan for the site has been prepared. The plan includes provisions for alarms both inside and outside buildings and delineates the evacuation routes and assembly areas to be used. The State of

Connecticut Emergency Plan also provides for the control of activities in that portion of the exclusion area extending offshore through a written agreement between the licensee and the U.S. Coast Guard at their station in New London, Connecticut.

The owners have encouraged public use of portions of the site. Ownership rights have not, however, been relinquished, and the owners can, and have provision to, fulfill their obligations with respect to 10 CFR 20, "Standards for Protection Against Radiation."

A portion of the exclusion area is leased to the Town of Waterford for public recreation and is used primarily for soccer and baseball games. Figure 2.1-3 shows the general location of these activities. No attempt is made to restrict the number of persons using these facilities. Estimates of maximum attendance indicate that about 2,000 visitors could be within the exclusion area at any one time at the soccer and baseball fields. The licensee's Emergency Plan provides for removal of the visitors from the site. The number and configuration of roads and highways assure ready egress from the areas described above (Figures 2.1-2, 2.1-3, and 2.1-4).

2.1.1 POPULATION

The total 1990 population within 10 miles of the station was estimated to be 120,443. This population is expected to increase to about 129,846 people by the year 2000 and to a total of approximately 142,277 people by the year 2030 (New York State Department of Economic Development, 1989 (Reference 2.1-1); State of Connecticut Office of Policy and Management, 1991 (Reference 2.1-2); US Department of Commerce, Bureau of the Census, 1990 Census of Population (Reference 2-1-3)). The 10 mile area includes portions, or all of, New London and Middlesex Counties in Connecticut and a small portion on Suffolk County of Fishers Island which is part of the town of Southold, New York. Figure 2.1-5 shows counties and towns within the 10 mile area. Town populations and population densities are provided in Table 2.1-2.

The Town of Waterford, in which Millstone Unit Number 1 is located, contained a total population of 17,930 people in 1990 at an average density of 547 people per square mile (US Department of Commerce Bureau of the Census 1991) (Reference 2.1-3). The population growth of Waterford was small with the 1990 total representing only a 0.5 percent increase over its 1980 population. Compared to towns immediately surrounding it, with the exception of New London, Waterford had the lowest increase in population between 1980 and 1990 (US Department of Commerce Bureau of the Census, 1991 (Reference 2.1-3)).

Waterford's growth has been consistently slowing down over the past 30 years, as shown in Table 2.1-3. This slow growth is projected by state demographers to continue at a low rate through the year 2000, at which time the population is expected to reach 18,480. After that, it is projected to decrease in population. By the year 2010 (the last year of projections), the town's population is projected to be 18,080 (Connecticut Office of Policy and Management, Interim Population Projections, 1991 (Reference 2.1-2)). Population distribution by sector for the area within 10 miles of Millstone Unit Number 1 is shown for the years 1990, 2000, 2010, 2020 and 2030 in Tables 2.1-4 through 2.1-8, that are keyed to the population sectors identified in Figure 2.1-6.

Population distribution within 10 miles is based on 1990 US Census data by Census Block (Reference 2.1-3). The population within a Census Block was assumed to be distributed evenly over its land area, unless USGS 7.5 minute quadrangle maps indicated the population to be concentrated in only on portion of the Block. The proportion of each Block area in each grid sector was determined and applied to the Block total population, yielding the population in each grid sector. Population projections, by municipality, supplied by Connecticut's Office of Policy and Management provided growth factors for projection of Projections, 1991 (Reference 2.1-2).

2.1.1.1 Population Distribution Within 50 Miles

The area within 50 miles of Millstone Unit Number 1 includes portions, or all, of eight counties in Connecticut, four counties in Rhode Island and one county in New York. Figure 2.1-7 shows counties and towns within the 50 mile area. In 1990, the 50 mile area contained approximately 2,835,159 people (U.S. Department of Commerce), 1990 Census of Population and Housing (Reference 2.1-4). This population is projected to increase to about 3,223,654 by the year 2030 (Connecticut Office of Policy and Management, 1991 (Reference 2.1-2); New York State Department of Economic Development, 1989 (Reference 2.1-1); Rhode Island Department of Administration, 1989 (Reference 2.1-5); US Department of Commerce, 1990 Census of Population and Housing, 1991 (Reference 2.1-4)). Population distribution by sector for the area within 50 miles of Millstone Unit Number 1 is shown for the years 1990, 2000, 2010, 2020 and 2030 in Tables 2.1-9 through 2.1-13, which are keyed to the population sectors identified in Figure 2.1-9.

Population distribution and projections within the 50 mile region surrounding Millstone Unit Number 1 were calculated based on population by municipalities and were assigned to sectors based on land area allocation. Projections for the 50 mile area were based on country-wide projections.

2.1.1.2 Transient Population

Seasonal population increases resulting from an influx of summer residents total approximately 10,500. However, many of the beaches and recreation facilities in the area are used by residents, and therefore, do not represent any increase in population but instead a slight shift in population. There are, however, a number of schools, industries, and recreation facilities which create daily and seasonal variations in sector populations. Tables 2.1-14 through 2.1-16 show annular sector population variations resulting from school enrollments, industrial employment, and recreation facilities (with documented attendance).

2.1.1.3 Low Population Zone

The low population zone (LPZ) surrounding Millstone Unit Number 1 encompasses an area within a radial distance of about 2.4 miles. The distance was chosen based on the requirements of 10 CFR 100.11. Figure 2.1-8 shows topographical features, transportation routes, facilities, and institutions within the LPZ.

The LPZ contained approximately 9,846 people in 1990, with an average density of 545 people per square mile. By the year 2030, the LPZ population is projected to increased to about 11,629, or an average density of 643 people per square mile (US Department of Commerce, Bureau of the Census, 1991 (Reference 2.1-3); Connecticut Office of Policy and Management, 1991 (Reference 2.1-2); US Geological Survey (Reference 2.1-6)). The LPZ population distribution for 1990 and 2030 is shown in Table 2.1-17. Table 2.1-18 shows the 1991-1992 school and employment distribution within the LPZ. Both tables are keyed to Figure 2.1-9.

Daily and seasonal variations due to transient population are minimal within the LPZ. Several beaches are located within the area; however, they are predominantly used by local residents and generally have no facilities for parking or accommodation of large groups. Three schools, Great Neck Elementary and Southwest Elementary in Waterford, and Niantic Elementary in East Lyme, are located within the LPZ. Major employment consists of the Camp Rell Military Reservation and Hendel Petroleum. The New London Country Club is also located within the LPZ.

2.1.1.4 Population Center

The closest population center to Millstone Unit Number 1 (as defined by 10 CFR 100 to contain more than 25,000 residents) is the city of New London which contained a 1990 population of 28,540 people at an average population density of 5,189 people per square mile (US Department of Commerce Bureau of the Census 1991). The distance between Millstone Unit Number 1 and the city's closest corporate boundary is about 3.3 miles to the northeast, just beyond the minimum distance requirement set by 10 CFR 100.

The region within 50 miles of Millstone Unit Number 1 includes portions, or all, of 11 Metropolitan Statistical Area's. The populations of these areas are shown in Table 2.1-19.

There were 38 population centers within 50 miles of Millstone Unit Number 1, containing 25,000 or more people in 1990. They are listed in Table 2.1-20 with the populations indicated.

The population of the area within 50 miles of Millstone was approximately 2,800,000 in 1990, with an average density of 361 people per square mile. This density is lower than the NRC comparison figure of 500 people per square mile (NRC Regulatory Guide 1.70, Revision 3, Reference 2.1-7). Within 30 miles of Millstone, the population density is considerably less, at an average of 189 people per square mile. By 2030, the 50 mile population is projected to increase to 3,200,000 or an average population density of about 410 people per square mile, considerable lower than the NRC comparison figure for end-year plant life of 1,000 people per square mile. Within 30 miles, the average density will be 223 persons per square miles by the year 2030. Population densities by sector for 1990 and 2030 are shown for within 10 miles of Millstone in Tables 2.1-21 and 2.1-23 respectively, which are keyed to Figure 2.1-6, and for within 50 miles of Millstone in Tables 2.1-23 and 2.1-24, respectively, which are keyed to Figure 2.1-7. Cumulative population densities 1990 and 2030 are shown in Tables 2.1-25 and 2.1-26, respectively.

2.1.2 LAND USE

The area around the Millstone site contains three major industrial facilities (Dow Chemical Corporation, Pfizer Corporation, and Electric Boat division of General Dynamics Corporation); two transportation facilities (Groton/New London) Airport and the New London Transportation Center; and four military installations (U.S. Navy Submarine Base, U.S. Coast Guard Academy, Camp Rell, and Stone's Ranch Military Reservation).

There is also an interstate highway (Interstate 95), passenger and freight railroad lines, gas distribution lines, above ground gas and oil storage facilities and two major waterways (Long Island Sound, Thames River) in the vicinity of the Millstone site.

There are no major gas transmission lines, oil transmission or distribution lines, under ground gas storage facilities, drilling or mining operations, or military firing, or bombing ranges near the site.

Aircraft patterns and routes are shown of Figures 2.1-10 and 2.1-11. Figure 2.1-12 shows the road and highway system in the area of the Millstone site.

2.1.2.1 Description of Facilities

A summary of the significant industrial, transportation, military, and industrial related facilities, and products and materials used, is shown in Table 2.1-27 as listed below.

1. Dow Chemical Corporation of Allen Point, Ledyard, Connecticut is located on the east bank of the Thames River approximately 10 miles north-northeast of the site. Dow Chemical employs approximately 115 people and produces organic compounds, such as Styron, Styrofoam, and a base product of latex paints. All materials are moved to and from the company by truck and/or railroad.

2. Pfizer Corporation of Eastern Point Road, Groton, Connecticut is located on the east bank of the Thames River, approximately 4.9 miles east-northeast of the site. Pfizer Corporation employs approximately 3,000 persons and produces organic compounds and pharmaceutical materials, such as citric acid, antibiotics, synthetic medicines, vitamins and caffeine. All materials are moved to and from Pfizer corporation by truck and/or railroad.

3. Electric Boat Division of General Dynamics of Eastern Point Road, Groton, Connecticut is located approximately 5 miles east-northeast of the site. Electric boat employs approximately 12,000 persons, and is a producer of submarines and oceanographic equipment for commercial industry and the U.S. Navy. The nature of products produced at Electric Boat requires that they handle substantial amounts of nuclear material which is licensed under the Naval Reactors Division. All material is moved by truck, railroad, and/or barge to and from the company with the exception of completed ships which leave under their own power.

4. Groton / New London Airport, approximately 6 miles east-northeast of the site, handles regularly scheduled commercial passenger flights. Approximately 13 persons are employed at Groton/New London Airport on a full-time basis, excluding airline and car rental employees. The National Guard has an aircraft repair facility at the airport that has approximately 140 full time employees.

5. The New London Transportation Center, located at City Pier, New London on the west bank of the Thames River, is approximately 4 miles northeast of the site. Approximately 20 persons are employed there on a full-time basis. The New London Transportation Center is a large complex in downtown New London in the City Pier area. It encompasses numerous facilities, including a train station, several ferry companies, commercial and private boat slips, an interstate bus terminal, local bus inter-changers, and commercial land transportation facilities. It serves as the prime entrance and exit for New London for civilian and commercial travel.

6. U. S. Navy Submarine Base, Groton, Connecticut is located on the east bank of the Thames River, approximately 7 miles northeast of the site. The base population includes approximately 8,500 military personnel. In addition, there are about 1,800 civilian employees at the base. The U.S. Navy Submarine Base provides logistics as well as training and operation of the base and its ships (nuclear and non-nuclear). All materials are moved by truck, railroad, barge and / or ship, to and from this government installation.

7. The U.S. Coast Guard Academy, New London, Connecticut is located on the west bank of the Thames River, approximately 5.6 miles northeast of the site. Approximately 900 cadets attend the academy, while approximately 360 military and civilian personnel are employed here.

8. Camp Rell, located approximately 2 miles northwest of the site, is a training headquarters for the Connecticut Army National Guard. It is owned and operated by the Military Department of the State of Connecticut. On a full-time basis, it employs 16 persons (military and civilian), including the headquarters for the Connecticut Military Academy, post Operations personnel, and 745th Signal Company. On a part-time basis, during various weekends, Camp Rell is occupied by varying numbers of troop units for administrative training maneuvers, billeting, and supply functions for the Connecticut Army National Guard. During the training maneuvers there may be from 300 to 1,200 people at the facility. Camp Rell is an administrative training center for troops of the Connecticut Army National Guard. Because of the solely administrative nature of its occupancy, the camp's operation has no effect on the Station's operation.

9. In addition to Camp Rell, the Military Department of the State of Connecticut also maintains a field training facility known as Stone's Ranch Military Reservation, located approximately 7 miles northwest of the site. Fourteen persons are employed there full-time for two regional motor vehicle and equipment maintenance shops. It is also occupied on a part-time basis by varying numbers of troop units for periods of field training for the Connecticut Army National Guard. During some weekend training sessions there may be up to 500 people at the facility.

Limited quantities of munitions and explosives are stored in underground bunkers at this facility. These materials are used in quarry operations for the Connecticut Army Corps of Engineers. No live ammunition is used at the facility. All materials are moved to and from Stone's Ranch by truck.

In addition, a small paved utility landing strip is located at Stone's Ranch. While capable of handling light, fixed-wing aircraft, the strip is not routinely used except for occasional rotary-wing operations. Because of its distance from the site, the limited quantity of materials stored and used, and the type of aircraft operations occurring at the facility, Stone's Ranch Military Reservation does not pose any hazard to the Millstone station.

10. Hess Oil Corporation of Eastern Point Road, Groton, Connecticut is located on the east bank of the Thames River, approximately 5 miles east-northeast of the site. It is located north of Pfizer Corporation, and south of General Dynamics-Electric Boat Division and services as a fuel storage facility. There are about 14 persons employed there on a full time basis. Hess Oil Corporation operates a fuel distribution and storage facility for home heating oil and kerosene. There are large above ground tanks capable of storing heating oil, residual fuel oil, and kerosene. The fuel arrives by ships or barges and is distributed by trucks.

11. There is one medium-sized propane storage area in the proximity of the Millstone site. Hendel Petroleum Company, is located in Waterford, approximately 2.5 miles northeast of the site on Great Neck Road, and employs about 75 people. Hendel Petroleum Company operates a fuel distribution facility for commercial and residential use. There are 5 above ground tanks (3-30,000 gallons and 2-16,000 gallons) which are capable of storing 126,000 gallons total of propane gas. The facility also stores 40,000 gallons of gasoline, and 40,000 gallons of Number 2 fuel oil. The propane for the facility arrives by train and truck, and is distributed by truck.

    On the Millstone site, at the Fire Training Facility located approximately 2,800 feet to the north of the protected area are two 1,000 gallon propane cylinders. The two cylinders are used to supply propane to the fire simulator.The Fire Training Facility was constructed in 1994 for the purpose of training fire brigade members. The Training Facility consists of six live burn "mock-ups" which replicate nuclear power plant fire hazards. Propane is used to fuel these "fireplaces." The two storage cylinders are positioned such that their ends are pointed away from the Millstone site. Both cylinders are above ground domestic storage cylinders designed per ASME Code for Pressure Vessels, Section VIII Division 1-92.

12. Montville Station is a Fossil Fuel powered electric generating plant operated by Connecticut Light & Power Company in Montville, Connecticut. It is located on the west bank of the Thames River, approximately 9.5 miles north-northeast of the site. Approximately 67 people are employed there. It is capable of providing 498 MW of electric power. The fuel is stored in three large above ground tanks, capable of storing approximately 175,000 barrels of fuel each; two medium above ground tanks, capable of storing approximately 12,000 barrels of fuel each; and two small above ground tanks,

capable of storing approximately 250 barrels of fuel each. The fuel arrives by barge or trucks.

2.1.2.2 Pipelines

There are no major gas transmission lines within 5 miles of the site. There are two medium pressure gas distribution lines in near proximity of the site. The nearest gas distribution line is approximately 2.9 miles from the site, located along Rope Ferry Road in Waterford. This 35 psi gas distribution line is a 6-inch plastic pipeline, buried approximately 3 feet deep. The control valve for this line is located at the intersection of Clark Lane and Boston Post Road in Waterford. The second gas distribution line, ends at and serves the shopping center complex, near the intersection of I-95 and Parkway North, approximately 4 miles north of the site. This 35 psi gas distribution line is an 8 inch plastic pipeline buried approximately 3 feet deep. The control valve for this line is located at the complex where it intersects with Parkway North.

There are no oil transmission or distribution lines within 5 miles of the Millstone site.

2.1.2.3 Waterways

Ships that pass by the site in the shipping channels of Long Island Sound are of two types: general cargo freighters, usually partially unloaded, with drafts of 20 to 25 feet, and deep draft tankers with drafts of 35 to 38 feet. Both of these classes of ships must remain at least 2 miles offshore to prevent running aground on Bartlett Reef.

No oil barges pass to the shore side of Bartlett Reef, and since there are no tank farms in Niantic Bay, no oil barges pass with 2 miles of the site.

Barge traffic in the vicinity of the site has been diminishing over the past several years due to the decrease in the amount of oil used by area facilities. Barge traffic is heaviest during the winter months, and averages only 1 barge per day during these months. On the average of once a month, a barge carrying 15,000 barrels of sulfuric acid is towed past the site outside of Bartlett Reef. Approximately 10 ships per day traverse the Reef in the vicinity, 6 miles of the site.

For these reasons, it is concluded that shipping accidents would not adversely affect Millstone 3 safety related facilities.

2.1.2.4 Airports

Groton / New London Airport, approximately 6 miles east-northeast of the site, handles regularly scheduled commercial passenger flights. It is served by U.S. Air Express. It has two runways: 5-23, which is 5,000 feet long; and 15-33, which is 4,000 feet long. Both runways are illuminated. There is a control tower at Groton / New London, with ILS (Instrument Landing System) and VOR (Very High Frequency Omni Range) navigation aides located on the airfield. The ILS is associated with runway 5. As shown on Figure 2.1-10, the landing patterns used do not direct traffic near the Millstone site.

The largest commercial aircraft to use Groton / New London Airport on a regularly scheduled basis are Beechcraft 1900's which carry approximately 19 passengers. The only jets using the airport on a regular basis are two small chartered Cessna Citation which carry 10 passengers.

The largest military aircraft to use Groton/New London Airport on an occasional basis are C-130's and C-23's. Additionally, there are several military helicopters stationed at the airport.

In 1995 there were approximately 4,490 military flights, approximately half of which were military helicopters. Millstone Station is not in the flight path of these flights, and pilots are briefed to avoid the site.

As shown on Figure 2.1-11, the air lane nearest the site is V58 which is approximately 4 miles northeast of the site. Other adjacent air lanes include V16, which is approximately 6 miles northwest of the site, and V308, which is approximately 8 miles east of the site.The nearest high-altitude jet route, J121-581, passes approximately 9 miles southeast of the site. A second jet route, J55, passes approximately 12 miles northwest of the site.

2.1.2.5 Highways

The area around the Millstone site is served by interstate, state and local roads. These are shown on Figure 2.1-12. The nearest major highway which would be used for frequent transportation of hazardous materials is U.S. Interstate 95, which is located 4 miles from the Millstone site. Other principal highways which pass near the site include U.S. Highway 1 which is located 3 miles from the site, and State Highway 156, located 1.5 miles from the site.

These separation distances exceed the minimum distance criteria given in Regulatory Guide 1.91, Revision 1 and provide assurance that any transportation accidents resulting in explosions or toxic gas releases of truck size shipments of hazardous materials would not have a significant adverse effect on the safe operation or shutdown capability of the unit.

2.1.2.6 Railroads

The site is traversed from east to west by a Providence & Worcester (P&W)/Amtrak railroad right-of-way. The mainline tracks are more than 2,000 feet from the Millstone Unit Number 1 Reactor Building structure.

The motive force for the rail stock is both diesel and electric locomotives. Overhead electric lines power the former. These lines affect neither the site nor the overhead transmission lines leaving the site and traversing the railroad right-of-way above the tracks.

The Department of Transportation and P&W/Amtrak have been contacted for information concerning rail traffic on the mainline tracks. Approximately eighteen scheduled passenger trips per day pass along the tracks near the Millstone site.

Approximately one freight train per day passes by the site. Hazardous material shipped on the track include chlorine, anhydrous ammonia, carbon dioxide, propane, ethyl alcohol, rosin, ammonium nitrate, and hydrochloric acid. See Table 2.1-28 for a list of hazardous materials handled over this track which are potentially capable of producing significant missiles.

Records of hazardous materials incidents dating back a number of years show no incident occurring between East Lyme and New London, Connecticut, including the trackage near the Millstone site. See Section 2.1.3 for a more detailed evaluation of potential accidents.

The railroad spur serves the Millstone Nuclear Power Station exclusively. The switch for that spur is normally set for through traffic. In order to reach any station facility, a train car must also pass through a second switch, which is normally set to direct traffic past the station to a dead end near the Sound. Therefore, the possibility of unauthorized transport of hazardous materials on the spur is very remote.

There are no grade crossings on or adjacent to the site at which hazardous materials might be transported across the tracks.

2.1.2.7 Projections of Industrial Growth

<u>Pipelines</u>

No expansion of facilities is presently planned in the area for oil distribution within the southeastern region of Connecticut. The gas distribution line along Rope Ferry Road ends at Waterford high School, approximately 2.9 miles from the Millstone site. The gas distribution line at I-95 and Parkway North ends at, and serves the shopping complex approximately 4 miles from the Millstone site.

<u>Waterways</u>

As previously mentioned, ship and barge traffic in the area of Millstone site has decreased over the past several years. No new ship or barge traffic is anticipated at this time in the Niantic Bay area on Long Island Sound near location of the intake structures.

<u>Airports</u>

No expansion of facilities at Groton / New London Airport is proposed although some improvements to the facility, such as expansion of the approach lights, and upgrading of the terminal and runways in planned. Southeastern Connecticut Regional Planning Agency (SCRPA) recommends that a master plan be prepared for the airport before any major physical improvements are made. The agency has previously adopted the policy that Groton / New London Airport should remain a small feeder airport providing connection to larger airports and direct service to a limited number of cities with a 500 mile radius.

2.1.3 DETERMINATION OF DESIGN BASIS EVENTS

The area around the Millstone site was investigated and found to contain no explosives, chemicals, airborne pollutants, flammable or dangerous gases, nor tanks or pipelines near enough to the site to pose a danger if they were to explode or burn.

A railroad right-of-way of P&W/Amtrak companies transverses the site from east to west. The mainline tracks are about 0.5 miles from the Millstone Unit Number 1 Reactor building and upgrade from the plant. Traffic on the spur of the mainline track which extends onto the site is controlled to minimize the possibility of railroad traffic-related accidents.

A spur of the P&W/Amtrak railroad serves the Millstone Nuclear Power Station exclusively. The switch for that spur is normally set for through traffic. To reach any station facility, the locomotive must pass through a second switch, which is normally set to direct traffic past the station to a dead end near the Sound. Therefore, the possibility of unauthorized transport of hazardous materials does not exist on the spur.

Hazardous materials that are shipped on the track which crosses the site between New Haven and New London include chlorine, anhydrous ammonia, carbon dioxide, propane, ethyl alcohol, rosin, ammonium nitrate, and hydrochloric acid. Among these materials, only the shipment of propane (about 44 carloads per year) is in the “frequently shipped quantities of hazardous material” category as defined in Regulatory Guide 1.78.

The nearest major highway which would be used for frequent transportation of hazardous materials is U.S. Interstate 95, which is located at a distance of 4 miles from the Millstone site. This separation distance exceeds the minimum distance criteria given in Regulatory Guide 1.91, Revision 1; and therefore, provides assurance that any transportation accidents resulting in explosions of truck size shipments of hazardous materials will not have an adverse effect on the safe operation of the plant.

Based upon the size of Groton / New London airport and the location of flight paths, the impact of an airplane on Millstone Unit Number 1 is highly unlikely.

There are no major gas transmission lines within 5 miles of the site. The nearest low pressure gas distribution line is 2.9 miles from the site and is located near Waterford High School on Rope Ferry Road.

The closest oil transmission line is approximately 5 miles from the site in Groton Connecticut.

Because they are 5 miles or more away from the site, both the major gas and oil transmission lines constitute no threat to the safe conduct of activities associated with storage of irradiated fuel or decommissioning of Millstone Unit Number 1 or to the site in general.

2.1.4 EFFECTS OF DESIGN BASIS EVENTS

Propane gas is heavier than air and can form a potentially explosive mixture in air. However as shown on topographic maps of the area, the rail line on which propane is shipped through the site runs through an excavation in the hill which is approximately 20 feet below the natural contour of the ground immediately north of the reactor facilities. This railroad cut would channel a heavier-than-air propane cloud in an east-west direction away from the plant. The map indicates that the remainder of the topography of the site is about the same grade as the rail line and therefore would not cause a gravity flow of the cloud toward the plant site.

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TABLE 2.1–1 THIS TABLE HAS BEEN INTENTIONALLY DELETED

TABLE 2.1–2 1990 POPULATION AND POPULATION DENSITIES - CITIES AND TOWNS WITHIN 10 MILES OF MILLSTONE

Municipality	1990 Population Total	1990 Population Density (People/Square Mile)	1980-1990 Change (%)
East Lyme	15,340	451	10.6
Groton (including City)	45,144	1,442	9.9
Ledyard	14,913	391	8.6
Lyme	1,949	61	7.0
Montville	16,673	397	1.3
New London	28,540	5,189	-1.0
Old Lyme	6,535	283	6.1
Old Saybrook	9,552	637	2.9
Waterford	17,930	547	0.5
Southold, New York (Fishers Island)	19,836	394	3.5

NOTES:

Based on 1990 US Census of Population and Housing.

Includes total 1990 population of all municipalities totally or partially within 10 miles of the site.

TABLE 2.1–3 POPULATION GROWTH 1960 - 1990

Municipality	Total Population				% Change		
	1960	1970	1980	1990	1960-1970	1970-1980	1980-1990
East Lyme	6,782	11,399	13,870	15,340	68.1	21.7	10.6
Groton	29,937	38,523	41,062	45,144	28.7	6.6	9.9
Ledyard	5,395	14,558	13,735	14,913	169.8	-5.7	8.6
Lyme	1,183	1,484	1,822	1,949	25.4	22.8	7.0
Montville	7,759	15,662	16,455	16,673	101.9	5.1	1.3
New London	34,182	31,630	28,842	28,540	-7.5	-8.8	-1.0
Old Lyme	3,068	4,964	6,159	6,535	61.8	24.1	6.1
Old Saybrook	5,274	8,468	9,287	9,552	60.6	9.7	2.9
Waterford	15,391	17,227	17,843	17,930	11.9	3.6	0.5

SOURCES:

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TABLE 2.1–4 POPULATION DISTRIBUTION WITHIN 10 MILES OF MILLSTONE - 1990 CENSUS

Sector	Distance to Plant										Total
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	
N	16	722	866	784	116	213	542	209	536	1,717	5,721
NNE	13	359	1,146	1,978	1,861	1,622	2,242	2,242	2,192	3,142	16,221
NE	165	455	839	3,888	10,584	7,752	8,164	8,129	911	1,961	42,646
ENE	22	455	292	4,963	971	7,186	3,748	3,748	1,008	2,662	24,354
E	0	636	413	1,804	193	552	0	63	1,434	904	5,999
ESE	0	143	36	0	0	0	0	0	115	214	508
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	
SW	0	0	14	0	0	0	0	0	0	0	14
WSW	0	0	489	91	86	312	472	158	0	74	1,682
W	0	178	1,061	1,014	440	763	476	562	881	408	5,782
WNW	0	476	1,165	1,946	346	239	211	1,654	509	4-17	6,981
NW	0	634	873	1,192	1,140	644	599	101	209	81	5,473
NNW	148	314	892	522	646	918	221	429	456	314	4860
Total	354	4,372	8,086	18,200	16,383	20,201	16,098	16,594	8,251	11,894	120,443

TABLE 2.1–5 POPULATION DISTRIBUTION WITHIN 10 MILES OF MILLSTONE 2000 PROJECTED

Sector	Distance to Plant										Total
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	
N	18	778	932	845	126	230	582	225	578	1,852	6,166
NNE	14	387	1,234	2,131	2,006	1,749	1,796	2,415	2,366	3,389	17,487
NE	179	489	905	4,191	11,441	7,359	8,802	8,765	983	2,115	46,203
ENE	24	492	314	5,352	1,045	7,746	4,041	3,285	1,087	2,870	26,256
E	0	685	444	1,944	208	597	0	68	1,546	975	6,467
ESE	0	154	39	0	0	0	0	0	125	233	551
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	14	0	0	0	0	0	0	0	14
WSW	0	0	528	98	92	336	509	169	0	78	1,810
W	0	192	1,144	1,093	473	821	513	606	950	436	6,228
WNW	0	514	1,255	2,118	373	258	227	1,783	548	448	7,524
NW	0	684	940	1,285	1,229	695	646	108	226	88	5,901
NNW	158	304	961	564	696	990	238	462	491	339	5,239
Total	393	4,715	8,710	19,621	17,663	21,781	17,354	17,886	8,900	12,823	129,846

TABLE 2.1–6 POPULATION DISTRIBUTION WITHIN 10 MILES OF MILLSTONE 2010 PROJECTED

Sector	Distance to Plant										Total
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	
N	18	803	961	871	129	237	600	230	595	1,908	6,352
NNE	14	399	1,272	2,197	2,068	1,804	1,853	2,492	2,437	3,495	18,301
NE	184	504	930	4,321	11,767	8,617	9,074	9,036	1,013	2,180	47,626
ENE	25	507	324	5,518	1,078	7,988	4,166	3,387	1,119	2,960	27,072
E	0	707	458	2,005	215	616	0	70	1,593	1,005	6,669
ESE	0	159	41	0	0	0	0	0	138	255	593
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	15	0	0	0	0	0	0	0	15
WSW	0	0	54	102	95	346	525	175	0	79	1,867
W	0	198	1,179	1,126	440	488	847	530	625	443	6,417
WNW	0	529	1,294	2,184	385	266	234	1,838	566	461	7,757
NW	0	705	969	1,325	1,267	716	666	111	232	90	6,081
NNW	163	350	992	582	718	1,021	245	476	506	350	5,403
Total	404	4,861	8,980	20,231	18,210	22,458	17,893	18,440	9,180	13,226	133,883

TABLE 2.1–7 POPULATION DISTRIBUTION WITHIN 10 MILES OF MILLSTONE 2020 PROJECTED

Sector	Distance to Plant										Total
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	
N	19	828	990	899	133	243	620	236	613	1,968	6,549
NNE	14	411	1,310	2,264	2,132	1,860	1,909	2,569	2,513	3,602	18,584
NE	188	519	960	4,455	12,134	8,885	9,355	9,318	1,044	2,247	49,105
ENE	25	523	333	5,689	1,220	8,236	4,296	3,492	1,151	3,052	27,907
E	0	728	472	2,067	222	635	0	72	1,642	1,036	6,874
ESE	0	162	41	0	0	0	0	0	144	268	615
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	15	0	0	0	0	0	0	0	15
WSW	0	0	562	105	98	356	541	180	0	80	1,922
W	0	205	1,216	1,161	504	874	546	644	1,011	450	6,611
WNW	0	544	1,226	2,252	398	274	242	1,895	583	476	8,000
NW	0	727	998	1,365	1,308	738	687	114	239	93	6,269
NNW	168	361	1,023	600	738	1,053	253	491	523	362	5,572
Total	414	5,008	9,256	20,857	18,777	23,154	18,449	19,011	9,463	13,634	138,023

TABLE 2.1–8 POPULATION DISTRIBUTION WITHIN 10 MILES OF MILLSTONE 2030 PROJECTED

Sector	Distance to Plant										Total
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	
N	19	855	1,021	927	136	250	638	242	631	2,027	6,746
NNE	14	425	1,351	2,334	2,196	1,916	1,968	2,650	2,590	3,712	19,156
NE	193	535	990	4,592	12,510	9,160	9,644	9,606	1,075	2,315	50,620
ENE	26	539	343	5,866	1,145	8,492	4,428	3,598	1,188	3,147	28,772
E	0	751	487	2,132	229	655	0	73	1,692	1,068	7,087
ESE	0	167	43	0	0	0	0	0	151	281	642
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	15	0	0	0	0	0	0	0	15
WSW	0	0	580	108	101	366	558	185	0	81	1,979
W	0	212	1,254	1,197	520	901	561	663	1,043	458	6,809
WNW	0	560	1,377	2,323	409	281	249	1,956	602	490	8,247
NW	0	748	1,029	1,407	1,349	761	708	116	246	95	6,459
NNW	174	371	1,055	618	761	1,085	261	507	539	374	5,745
Total	426	5,163	9,545	21,504	19,356	23,867	19,015	19,596	9,757	14,048	142,277

TABLE 2.1–9 POPULATION DISTRIBUTION WITHIN 50 MILES OF MILLSTONE - 1990 CENSUS

Sector	Distance to Plant					Total
	0-10	10-20	20-30	30-40	40-50	
N	5,721	22,283	26,357	32,610	18,658	105,629
NNE	16,221	34,824	23,730	27,465	35,598	137,838
NE	42,848	9,444	11,334	29,987	199,334	292,947
ENE	24,354	23,914	16,498	43,001	99,721	207,488
E	5,999	10,712	7,992	10,920	0	35,623
ESE	508	0	0	836	0	1,344
SE	0	0	807	0	0	807
SSE	0	0	2,420	0	0	2,420
S	0	1,614	13,541	0	0	15,155
SSW	0	2,443	12,569	14,807	4,498	34,317
SW	14	938	22,042	8,252	143,933	175,179
WSW	1,682	2,471	0	0	20,389	24,542
W	5,782	27,956	34,384	184,723	267,465	520,310
WNW	6,981	12,474	27,895	148,259	259,824	455,433
NW	5,473	6,215	31,331	191,767	365,578	600,364
NNW	4,860	8,809	17,850	115,424	78,820	225,762
Total	120,443	164,097	248,750	808,051	1,493,818	2,835,159

TABLE 2.1–10 POPULATION DISTRIBUTION WITHIN 50 MILES OF MILLSTONE - 2000 PROJECTED

Sector	Distance to Plant					Total
	0-10	10-20	20-30	30-40	40-50	
N	6,166	24,028	28,707	35,404	20,273	114,578
NNE	17,487	37,551	25,721	29,926	38,135	148,820
NE	46,203	10,183	12,196	31,611	206,940	307,133
ENE	26,256	25,744	17,633	45,998	105,848	221,509
E	6,467	11,497	8,553	11,687	0	38,204
ESE	551	0	0	895	0	1,446
SE	0	0	878	0	0	878
SSE	0	0	2,635	0	0	2,635
S	0	1,759	14,742	0	0	16,501
SSW	0	2,660	13,688	16,122	4,897	37,367
SW	14	1,022	24,000	8,985	156,725	190,746
WSW	1,810	2,641	0	0	22,201	26,652
W	6,228	29,887	36,343	195,006	281,709	549,173
WNW	7,524	13,340	29,762	156,623	273,153	480,402
NW	5,901	6,660	33,435	200,205	380,339	626,540
NNW	5,239	9,492	19,194	121,620	83,732	239,277
Total	129,846	176,464	267,517	854,082	1,573,952	3,001,861

TABLE 2.1–11 POPULATION DISTRIBUTION WITHIN 50 MILES OF MILLSTONE - 2010 PROJECTED

Sector	Distance to Plant					Total
	0-10	10-20	20-30	30-40	40-50	
N	6,352	24,773	300,056	36,785	21,101	119,067
NNE	18,031	38,716	26,730	31,421	39,720	154,618
NE	47,626	10,499	12,626	32,221	210,368	313,340
ENE	27,072	26,652	18,530	48,258	109,494	230,006
E	6.669	11,986	8,981	12,272	0	39,908
ESE	593	0	0	940	0	1,533
SE	0	0	920	0	0	920
SSE	0	0	2,761	0	0	2,761
S	0	1,847	15,445	0	0	17,292
SSW	0	2,788	14,344	16,896	5,132	39,160
SW	15	1,073	25,151	9,416	164,248	199,903
WSW	1,867	2,689	0	0	23,267	27,823
W	6,417	30,426	37,096	199,100	286,889	559,928
WNW	7,757	13,590	30,311	159,776	278,156	489,590
NW	6,081	6,807	34,052	202,762	384,902	634,604
NNW	5,403	9,778	19,778	123,964	85,735	244,658
Total	133,883	181,624	276,781	873,811	1,609,012	3,075,111

TABLE 2.1–12 POPULATION DISTRIBUTION WITHIN 50 MILES OF MILLSTONE - 2020 PROJECTED

Sector	Distance to Plant					Total
	0-10	10-20	20-30	30-40	40-50	
N	6,549	24,541	31,470	38,219	21,963	123,742
NNE	18,584	39,916	27,784	32,989	41,349	160,622
NE	49,105	10,825	13,051	32,748	213,221	318,950
ENE	27,907	27,557	19,336	50,343	112,285	234,428
E	6,874	12,452	9,376	12,811	0	41,513
ESE	615	0	0	981	0	1,596
SE	0	0	965	0	0	965
SSE	0	0	2,894	0	0	2,894
S	0	1,939	16,184	0	0	18,123
SSW	0	2,922	15,033	17,707	5,379	41,041
SW	15	1,127	26,355	9,869	172,131	209,497
WSW	1,922	2,737	0	0	24,383	29,042
W	6,611	30,974	37,863	203,283	292,190	570,921
WNW	8,000	13,844	30,871	162,992	283,254	498,961
NW	6,269	6,957	37,678	205,354	389,518	642,776
NNW	5,572	10,070	20,382	126,369	87,794	250,187
Total	138,023	186,861	286,242	893,665	1,643,467	3,148,258

TABLE 2.1–13 POPULATION DISTRIBUTION WITHIN 50 MILES OF MILLSTONE - 2030 PROJECTED

Sector	Distance to Plant					Total
	0-10	10-20	20-30	30-40	40-50	
N	6,746	26,332	32,953	39,716	22,860	128,607
NNE	19,156	41,155	28,879	34,637	43,058	166,885
NE	50,620	11,159	13,494	33,286	219,112	324,671
ENE	28,772	28,495	20,176	52,519	115,158	245,120
E	7,087	12,937	9,789	13,375	0	43,188
ESE	642	0	0	1,024	0	1,666
SE	0	0	1,011	0	0	1,011
SSE	0	0	3,033	0	0	3,033
S	0	2,036	16,957	0	0	18,993
SSW	0	3,062	15,755	18,558	5,637	43,012
SW	15	1,183	27,619	10,342	180,394	219,553
WSW	1,979	2,787	0	0	25,554	30,320
W	6,809	31,532	38,647	207,551	297,607	582,146
WNW	8,247	14,102	31,441	166,276	288,449	508,515
NW	6,459	7,110	35,317	207,981	394,192	651,059
NNW	5,745	10,373	21,003	128,835	89,919	255,875
Total	142,277	192,263	296,074	914,100	1,678,940	3,223,654

TABLE 2.1–14 TRANSIENT POPULATION WITHIN 10 MILES OF MILLSTONE 1991-1992 SCHOOL ENROLLMENT

Sector	Distance to Plant										Total
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	
N	0	310	0	0	0	0	0	74	0	413	797
NNE	0	0	0	374	897	2,073	174	0	0	444	3,962
NE	0	0	636	210	697	1,352	1,542	534	0	0	4,971
ENE	0	0	0	2,501	0	888	0	1,043	1,609	266	6,307
E	0	181	0	0	0	1,330	0	0	183	0	1,805
ESE	0	0	0	0	0	0	0	0	0	0	68
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	0	0	0	0	0	0	0	0	0
WSW	0	0	0	0	0	0	0	0	0	0	0
W	0	0	0	0	0	0	263	0	864	0	1,127
WNW	0	0	345	0	0	0	0	0	0	0	345
NW	0	0	0	843	0	0	0	0	0	0	843
NNW	0	0	0	298	1,250	0	0	0	0	0	1,548
TOTAL	0	602	981	4,226	2,844	5,643	1,979	1,651	2,656	1,191	21,773

Note:

Includes student enrollment only.

Sources:

Connecticut Department of Education listing of schools: Telephone survey conducted in March 1992.

TABLE 2.1–15 TRANSIENT POPULATION WITHIN 10 MILES OF MILLSTONE - EMPLOYMENT

Sector	Distance to Plant										Total
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	
N	0	0	0	300	0	0	0	0	0	200	500
NNE	0	0	0	0	0	0	375	375	109	277	1,134
NE	0	0	375	80	831	0	375	375	0	0	2,036
ENE	0	0	0	0	8,800	5,500	820	0	0	0	15,120
E	0	0	0	0	0	0	0	0	0	0	0
ESE	0	0	0	0	0	0	0	0	0	0	68
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	256	0	256
SW	0	0	0	0	0	0	0	0	0	0	0
WSW	0	0	0	0	0	0	0	0	0	0	0
W	0	0	0	0	0	0	0	0	0	0	0
WNW	0	0	0	0	0	0	125	125	0	0	250
NW	0	500	0	843	0	0	125	125	0	0	750
NNW	0	0	0	0	0	0	0	0	0	0	0
TOTAL	0	500	375	380	9,631	5,500	1,820	1,000	363	477	20,046

Note: Firms with 50 employees or more. Excludes plant employee population.

Sources: Telephone suvey conducted in March 1992.

TABLE 2.1–16 POPULATION DISTRIBUTION WITHIN 50 MILES OF MILLSTONE - 2030 PROJECTED

FACILITY	LOCATION	TOTAL ANNUAL ATTENDANCE	SUMMER DAILY ATTENDANCE
State Parks:			
Bluff Point	ENE/E 6-8	97,641	490 *
Fort Griswold	ENE 5-6	58,965	200 *
Haley Farm	ENE/E 7-9	11,675	60 *
Harkness Memorial	E 2-3	157,962	790 *
Rocky Neck	W 3-5	412,495	2,360 **
State Forests:			
Nehantic	WNW/NNW 7-10	81,146	400 *

Notes:

* Daily summer attendance based on 90% of yearly attendance from April through September.

** Includes campers from April 15 to September 15.

Source:

State of Connecticut DEP - Office of Parks and Forests, 1990 Park Attendance.

TABLE 2.1–17 LOW POPULATION ZONE PERMANENT POPULATION DISTRIBUTIONS

DIRECTION	1990 CENSUS	2030 PROJECTED
N	1,298	1,536
NNE	903	1,065
NE	1,144	1,351
ENE	768	909
E	760	899
ESE	179	212
SE	0	0
SSE	0	0
S	0	0
SSW	0	0
SW	3	3
WSW	429	506
W	1,025	1,211
WNW	1,046	1,233
NW	1,167	1,377
NNW	1,124	1,327
TOTAL LPZ	9,846	11,629

Sources:

1990 Census of Population and Housing

Connecticut Office of Policy and Management, Interim Population Projections Series 91.1, 4/91

TABLE 2.1–18 LOW POPULATION ZONE SCHOOL ENROLLMENT AND EMPLOYMENT

DIRECTION	SCHOOL	EMPLOYMENT
N	310	0
NNE	0	0
NE	0	75
ENE	0	0
E	292	0
ESE	0	0
SE	0	0
SSE	0	0
S	0	0
SSW	0	0
SW	0	0
WSW	0	0
W	0	0
WNW	345	0
NW	0	500
NNW	0	0
TOTAL LPZ	947	0

Notes:

1991-1992 Student Enrollment

Firms with 50 employees or more.

Sources:

Telephone survey conducted in March 1992; Connecticut Department of Education school listing.

TABLE 2.1–19 METROPOLITAN AREAS WITHIN 50 MILES OF MILLSTONE 1990 CENSUS POPULATION

AREA	1990 POPULATION
Bridgeport - Milford, CT PMSA	443,722
Bristol, CT PMSA	79,488
Fall River, MA-RI PMSA	157,272
Hartford, CT PMSA	767,899
New Haven - Meriden, CT MSA	530,240
Nassau - Suffolk, NY PMSA	2,609,212
New Britain, CT PMSA	148,188
New London - Norwich, CT-RI MSA	266,819
Providence, RI PMSA	654,869
Waterbury, CT MAS	221,629
Middletown, CT PMSA	90,320

Notes:

PMSA - Primary Metropolitan Statistical Area.

MSA - Metropolitan Statistical Area.

Total population of metropolitan areas completely or only partially within 50 miles of the site.

TABLE 2.1–20 POPULATION CENTERS WITHIN 50 MILES OF MILLSTONE

STATE	MUNICIPALITY	1990 POPULATION
Connecticut	Branford	27,603
	Bristol	60,640
	Cheshire	25,684
	East Hartford	50,452
	East Haven	26,144
	Enfield	45,532
	Glastonbury	27,901
	Groton	45,144
	Hamden	52,434
	Hartford	139,739
	Manchester	51,618
	Meriden	59,479
	Middletown	42,762
	Milford	49,938
	Naugatuck	30,625
	New Britain	75,491
	New Haven	130,474
	New London	28,540
	Newington	29,208
	Norwich	37,371
	Shelton	35,418
	Southington	38,518
	Stratford	49,389
	Vernon	29,841
	Wallingford	40,822
	Waterbury	108,961
	West Hartford	60,110
	West Haven	54,021
	Wethersfield	25,651

TABLE 2.1–20 POPULATION CENTERS WITHIN 50 MILES OF MILLSTONE

STATE	MUNICIPALITY	1990 POPULATION
	Windsor	27,817
Rhode Island	Coventry	31,083
	Cranston	76,060
	Johnston	26,542
	Newport	28,227
	Warwick	85,427
	West Warwick	29,268
New York	Brookhaven	407,779
	Southampton	44,976

Notes: Municipalities with 25,000 people or more. Municipalities completely or only partially within 50 miles.

Source: 1990 U.S. Census of Population and Housing.

TABLE 2.1–21 POPULATION DENSITY WITHIN 10 MILES OF MILLSTONE 1990 (PEOPLE PER SQUARE MILE)

Sector	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	Average
N	82	1,226	883	571	66	99	212	71	161	460	292
NNE	66	610	1,168	1,440	1,054	751	653	762	657	843	827
NE	842	772	855	2,830	5,993	3,591	3,200	2,761	273	526	2,183
ENE	112	772	298	3,612	550	3,328	1,469	1,035	302	714	1,241
E	0	1,080	421	1,313	109	256	0	21	430	242	306
ESE	0	243	37	0	0	0	0	0	34	57	26
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	14	0	0	0	0	0	0	0	1
WSW	0	0	498	66	49	145	185	54	0	20	86
W	0	302	1,082	738	249	353	186	191	264	109	295
WNW	0	808	1,118	1,429	196	111	83	562	153	112	356
NW	0	1,076	890	868	646	298	235	34	63	22	279
NNW	755	533	909	380	366	425	87	146	137	84	248
Average	116	464	515	828	580	585	394	352	155	199	384

Source: 1990 Census of Population

TABLE 2.1–22 POPULATION DENSITY WITHIN 10 MILES OF MILLSTONE 2030 (PEOPLE PER SQUARE MILE)

Sector	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	Average
N	97	1,452	1,041	675	77	116	250	82	189	544	344
NNE	71	722	1,377	1,700	1,243	887	771	900	776	995	976
NE	985	908	1,009	3,345	7,084	4,243	3,780	3,263	322	621	2,579
ENE	133	915	350	4,272	648	3,933	1,736	1,222	356	844	1,466
E	0	1,275	496	1,553	130	303	0	25	507	286	361
ESE	0	284	44	0	0	0	0	0	45	75	33
SE	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	0	0	0	0
SSW	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	15	0	0	0	0	0	0	0	1
WSW	0	0	591	79	57	170	219	63	0	22	101
W	0	360	1,278	872	294	417	220	225	313	123	347
WNW	0	951	1,404	1,692	232	130	98	664	180	131	420
NW	0	1,270	1,049	1,025	764	352	278	39	74	25	329
NNW	888	630	1,075	450	431	503	102	172	162	100	293
Average	136	548	608	979	685	691	466	416	183	235	453

Source: CT Office of Policy and Management, Interim Population Projections Series 91.1, 4/91.

TABLE 2.1–23 POPULATION DENSITY WITHIN 50 MILES OF MILLSTONE 1990 (PEOPLE PER SQUARE MILE)

Sector	0-10	10-20	20-30	30-40	40-50	Average
N	292	378	269	237	106	215
NNE	827	591	242	200	202	281
NE	2,183	160	116	218	1,129	597
ENE	1,241	406	168	313	564	423
E	306	182	81	79	0	73
ESE	26	0	0	6	0	3
SE	0	0	8	0	0	2
SSE	0	0	25	0	0	5
S	0	27	138	0	0	31
SSW	0	41	128	108	25	70
SW	1	16	225	60	815	357
WSW	86	42	0	0	115	50
W	295	475	350	1,345	1,514	1,061
WNW	356	212	284	1,079	1,471	928
NW	279	106	319	1,396	2,070	1,224
NNW	248	150	182	840	446	460
Average	384	174	158	368	528	361

Source: 1990 Census of Population and Housing.

TABLE 2.1–24 POPULATION DENSITY WITHIN 50 MILES OF MILLSTONE 2030 (PEOPLE PER SQUARE MILE)

Sector	0-10	10-20	20-30	30-40	40-50	Average
N	344	447	336	289	129	262
NNE	976	699	294	252	244	340
NE	2,579	190	138	242	1,224	662
ENE	1,466	484	206	382	652	499
E	361	220	100	97	0	88
ESE	33	0	0	7	0	3
SE	0	0	10	0	0	2
SSE	0	0	31	0	0	6
S	0	35	173	0	0	39
SSW	0	52	161	135	32	88
SW	1	20	81	75	1,021	447
WSW	101	47	0	0	145	62
W	347	536	394	1,511	1,685	1,187
WNW	420	240	320	1,210	1,633	1,036
NW	329	121	360	1,514	2,232	1,327
NNW	293	176	214	938	509	522
Average	453	204	189	416	594	410

Source: CT Office of Management, interim Population projections, Series 91.1, 4/91.

TABLE 2.1–25 CUMULATIVE POPULATION DENSITY WITHIN 50 MILES OF MILLSTONE 1990 (PEOPLE PER SQUARE MILE)

Sector	0-10	10-20	20-30	30-40	40-50	Average
N	292	378	269	237	106	215
NNE	827	591	242	200	202	281
NE	2,183	160	116	218	1,129	597
ENE	1,241	406	168	313	564	423
E	306	182	81	79	0	73
ESE	26	0	0	6	0	3
SE	0	0	8	0	0	2
SSE	0	0	25	0	0	5
S	0	27	138	0	0	31
SSW	0	41	128	108	25	70
SW	1	16	225	60	815	357
WSW	86	42	0	0	115	50
W	295	475	350	1,345	1,514	1,061
WNW	356	212	284	1,079	1,471	928
NW	279	106	319	1,396	2,070	1,224
NNW	248	150	182	840	446	460
Average	384	174	158	368	528	361

Source: 1990 Census of Population and Housing.

TABLE 2.1–26 CUMULATIVE POPULATION DENSITY WITHIN 50 MILES OF MILLSTONE 2030 (PEOPLE PER SQUARE MILE)

Sector	0-10	0-20	0-30	0-40	0-50
N	344	421	374	337	262
NNE	976	768	505	394	340
NE	2,579	787	426	346	662
ENE	1,466	730	438	414	499
E	361	255	169	138	88
ESE	33	8	4	5	3
SE	0	0	6	3	2
SSE	0	0	17	10	6
S	0	26	108	60	39
SSW	0	39	107	119	88
SW	1	15	163	125	447
WSW	101	61	27	15	62
W	347	488	436	906	1,187
WNW	420	285	305	701	1,036
NW	329	173	277	818	1,327
NNW	293	205	210	529	522
Average	453	226	223	307	410

TABLE 2.1–27 DESCRIPTION OF FACILITIES

Facility	Location	Approx. No. Persons Employed or Stationed	Approximate Distance From Site Miles	Sector
Industrial				
1. Dow Chemical Corp	Ledyard	115	10+	NNE
2. Pfizer Corporation	Groton	3,000	4.9	ENE
3. Electric Boat (Division of General Dynamics)	Groton	12,000	5	ENE
Transportation				
4. Groton/New London Airport (Trumbull)	Groton	153	6	ENE
5. New London Transportation Center	New London	20	4	NE
Military				
6. U.S. Navy Submarine Base	Groton	10,300	7	NE
7. U.S. Cost Guard Academy	New London	1,260	5.6	NE
8. Camp Rowland	East Lyme	16	2	NW
9. Stone's Ranch Military Reservation	East Lyme	14	7	NW
Industrial Related Facilities				
10.Hess Oil Corporation	Groton	14	5	ENE
11. Hendel Petroleum Co.	Waterford	75	2.5	NE
12. Montville Station Electric Generation Plant	Montville	67	10	NNE

TABLE 2.1–28 LIST OF HAZARDOUS MATERIALS POTENTIALLY CAPABLE OF PRODUCING SIGNIFICANT MISSILES

Hazardous Material	Avg. No. of Cars per Train Containing Hazardous Materials	Approx. No. of Cars per Year
1. Propane	2.20	44
2. Anhydrous Ammonia	0.266	5
Total	2.466	49

2.2 METEOROLOGY

Information regarding meteorology is presented in Section 2.3 of the Millstone Unit 3 Final Safety Analysis Report (Reference 2.2-1). With the exceptions given below, that information is incorporated herein by reference.

2.2.1 REGIONAL CLIMATOLOGY

(See Section 2.3.1 of the Millstone 3 Final Safety Analysis Report of Reference 2.2-1).

2.2.2 LOCAL METEOROLOGY

(See Section 2.3.1 of the Millstone 3 Final Safety Analysis Report of Reference 2.2-1).

2.2.2.1 Potential Influence of the Plant and Its Facilities on Local Meteorology

Millstone Unit Number 1 used a once-through cooling water system, discharging its cooling water into an existing quarry into which Units 2 and 3 also discharge and thence into Long Island Sound. Thin wisps of steam fog occasionally form over the quarry and less frequently over the discharge plume during the winter months, depending on tidal conditions and temperature differences between air and water. This fog dissipates rapidly as it moves away from the warm water area. Because the maximum discharge plume (defined by the 1.5°F isotherm of temperature differential when all three Millstone units were at full power) is approximately an ellipse of 1500 meter by 800 meters, the extent of the steam fog is negligible. With the permanent shutdown of Millstone Unit Number 1, this maximum discharge plume size is further reduced.

2.2.2.2 Local Meteorological Conditions for Design and Operating Bases.

2.2.2.2.1 Design Basis Tornado

The specifications for the Millstone Unit Number 1 design basis tornado are:

Rotational velocity	300 mph
Translational velocity	60 mph
Total pressure drop	2.25 psi
Rate of pressure drop	1.2 psi/sec

2.2.3 ON SITE METEOROLOGICAL MEASUREMENTS PROGRAM

The Millstone Site is served by a common meteorological tower, located south of Millstone Unit Number 1. The meteorological tower is capable of measuring wind speed, direction, and air temperature at various heights. For details regarding the capability of the On Site Meteorological Measurements program, see Section 2.3.3 of the Millstone 3 Final Safety Analyses Report Reference 2.2-1, with the exception that Millstone Unit Number 1 no longer has the data recording systems and data recording capability to display parameters transmitted by modem/

phone line from the instrument shack at the base of the tower to the control room area for display on the plant process computer described in Section 2.3.3.3.

2.2.4 SHORT TERM (ACCIDENT) DIFFUSION ESTIMATES

2.2.4.1 Objective

Accidents could result in short-term releases of radioactivity from several possible venting points. Atmospheric diffusion factors (χ/Q) based on site meteorological data are calculated at the exclusion area boundary (EAB) and low population zone (LPZ) for each downwind sector for each release point. The diffusion factors are calculated for different release time periods depending on the length of the release. These diffusion factors are used in the calculation of radiological consequences of the releases.

2.2.4.2 Calculations

2.2.4.2.1 Venting Point and Receptor Locations

The LPZ is taken to be 3860 m in all sectors from any release point.

2.2.4.2.2 Models

Accident χ/Q's were calculated using the basic methods of Regulatory Guide 1.145. χ/Q values for the Millstone Unit 2 Control Room due to ground level releases were calculated using the methods of Murphy and Campe. (Reference 2.2-2).

2.2.4.3 Results

The calculated χ/Q's used in design basis accident (DBA) radiological consequence calculations are presented with the list of assumptions in Chapter 5.

2.2.5 LONG-TERM (ROUTINE) DIFFUSION ESTIMATES

2.2.5.1 Objective

Low levels of radioactivity are routinely released on a continuous basis from the Unit Number 1 BOP exhaust point and the SFPI ventilation exhaust point. Atmospheric diffusion factors (χ/Q) based on site meteorological data are calculated for various downwind receptor locations of interest. The meteorological data is used to calculate the dose consequences to the public from routine airborne effluents. The calculated doses are submitted periodically to the Nuclear Regulatory Commission (NRC).

2.2.5.2 Calculations

2.2.5.2.1 Venting Point and Receptor Locations

Routine releases of radioactivity in gaseous effluents are vented from the Unit Number 1 BOP exhaust point and the SFPI ventilation exhaust point.

2.2.5.2.2 Database

Calculations are performed on a periodic basis using the actual meteorology for this period.

2.2.5.2.3 Models

χ/Q values are ground level dispersion factors, and releases are modeled using a conventional Gaussian plume model.

2.2.6 REFERENCES

2.2-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.3-Meterorology.

2.2-2 Murphy, K. G., and Campe, K. M. Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, 13th AEC Air Cleaning Conference, 1973.

2.3 HYDROLOGIC ENGINEERING

Information regarding hydrologic engineering is presented in Section 2.4 of the Millstone 3 Final Safety Analysis Report (Reference 2.3-1). With the exceptions given below, that information is incorporated herein by reference.

2.3.1 HYDROLOGIC DESCRIPTION

(See Section 2.4.1 of the Millstone 3 Final Safety Analysis Report, Reference 2.3-1).

2.3.2 SITE AND FACILITIES

Millstone Point is located on the north shore of Long Island Sound. To the west of the site is Niantic Bay and to the east is Jordon Cove. Figure 2.3-1 shows the general topography of the Millstone area. The site grade elevation for Millstone Unit Number 1 varies from 14 feet to above 15 feet mean sea level (MSL). Section 2.3.3.2 discusses the probable maximum hurricane used to calculate maximum water levels.

2.3.3 FLOODS

This section reviews the flood history in the vicinity of Millstone Point, flood design considerations, and the effects of local intense precipitation.

2.3.3.1 Flood History

Flooding near the site has historically been caused by hurricanes. The maximum historical flooding was the result of a hurricane on September 21, 1938, which produced a flood level of 9.7 feet MSL at New London, Connecticut.

The only sources of flooding that could affect Millstone Unit Number 1 are direct rainfall and storm surges.

2.3.3.2 Flood Design Considerations

The controlling event for flooding at the Millstone site is a storm surge resulting from the occurrence of a probable maximum hurricane (PMH) (see Section 2.3.6). The maximum stillwater level is +18.11 feet MSL, and the associated wave run up is +22.3 feet MSL.

Chapter 3 describes the flooding protective features at Millstone Unit Number 1.

2.3.3.3 Effect of Local Intense Precipitation

A discussion on the development of the probable maximum precipitation (PMP) for the site may be found in Section 2.3.2 of Reference 2.3-1.

A study was performed to determine the impact of the PMP intensity on the plant roof structures. The radwaste disposal building, intake structure, radwaste/control building, and southwest corner of the reactor building roofs can support the loads resulting from a PMP without crediting the roof drains.

The turbine building, reactor building, warehouse, and heating/ventilation area roofs credit scuppers to assure that the loads due to a PMP will remain below the roof design live loads.

PMP studies show that the area east of Millstone Unit Number 1, north of the radwaste truck bay, including the semi-enclosed area just east of the Unit 2 Control Room would have maximum ponding on the order of 15.5 to 16.2 feet. MSL. Further, these studies show that areas west of Millstone Unit Number 1 and 2, south of Millstone Unit Number 1, extending around the gas turbine building, to the east side of Millstone Unit Number 1 north of the radwaste truck bay would experience less ponding on the order of 14.6 to 14.9 feet. MSL. Ponding at the intake structure would be negligible since runoff would flow directly to the adjoining Niantic Bay.

During a PMP scenario, in-leakage through door openings could occur once the flood depths exceed door sill elevations. Secured external and internal doors will have a tendency to limit or control the amount of in-leakage.

2.3.4 PROBABLE MAXIMUM FLOOD (PMF) ON STREAMS AND RIVERS

(See Section 2.4.3 of Reference 2.3-1, the Millstone Unit 3 Final Safety Analysis Report).

2.3.5 POTENTIAL DAM FAILURE, SEISMICALLY INDUCED

(See Section 2.4.4 of Reference 2.3-1, the Millstone Unit 3 Final Safety Analysis Report).

2.3.6 PROBABLE MAXIMUM SURGE AND SEICHE FLOODING

2.3.6.1 Probable Maximum Winds and Associated Meteorological Parameters

The meteorologic characteristics used to calculate the probable maximum storm surge at the Millstone Point site are those associated with the PMH as reported by the U.S. National Oceanic and Atmospheric Administration (NOAA) in their unpublished report HUR 7-97. HUR 7-97 described the PMH as "…a hypothetical hurricane having that combination of characteristics which will make it the most severe that can probably occur in the particular region involved. The hurricane should approach the point under study along a critical path and at an optimum rate of movement." Actually, nine different PMH storm patterns can be constructed using wind speed, storm size and forward speed parameters given in HUR 7-97 in various combinations. The storm, which would cause the maximum surge buildup at the entrance to Long Island Sound is one with a large radius to maximum wind and a slow speed of translation. Pertinent parameters are tabulated below:

<u>Central Pressure Index</u>

The minimum surface atmospheric pressure in the eye of the hurricanes.

Radius to Maximum Wind

(R) at 48 nautical miles. This is the distance from the eye of the storm to the locus of maximum wind.

Forward Speed

(T) 15 knots. This is the rate of forward movement of the hurricane center.

Maximum Wind

(Vx) 115.5 mph. This is the absolute highest surface wind speed in the belt of maximum winds.

Peripheral Pressure

(Pn) 30.56 inches. This is the surface atmospheric pressure at the outer edge of the hurricane where the hurricane circulation ends.

Although other parametric combinations give a higher wind speed, this particular combination yields the highest surge.

2.3.6.2 Surge and Seiche Water Levels

Although frontal storms and squall lines cause tidal flooding in the Millstone Point area, by far the most severe flooding has resulted from hurricanes. For this reason, the PMH as defined in Section 2.3.6.1 was used to compute the design storm surge level at the site. The calculated total surge height or still water level considers the wind setup, the water level rise due to barometric pressure drop, the astronomical tide and forerunner or initial rise.

The maximum still water level is +18.11 feet, and the associated wave run up elevation is +22.3 feet MSL.

2.3.6.3 Wave Action

Wave characteristics are dependent upon wind speed and duration, fetch length, and water depth. Millstone Point is sheltered from the direct onslaught of open ocean waves by Long Island.

At the time of the peak surge, the wind is from the southeast direction and the wave attack would be along the large axis of the point. Thus the intake structure, and the southeast portions of the Reactor and Turbine Generator Buildings are primarily involved.

2.3.6.4 Resonance

(See Section 2.4.5 of the Millstone Unit 3 Final Safety Analysis Report Reference 2.3.1).

2.3.6.5 Protective Structures

At the time of the peak surge, the wind is from the southeast direction and the wave attack would be along the large axis of the point. Thus, the southeast portions of the Reactor Building would be primarily involved.

2.3.6.6 Probable Maximum Tsunami Flooding

(See Section 2.4.6 of the Millstone Unit 3 Final Safety Analysis Report Reference 2.3-1).

2.3.7 ICE EFFECTS

There is no available history of ice or ice jams in Niantic Bay.

2.3.8 COOLING WATER CANALS AND RESERVOIRS

(See Section 2.4.8 of the Millstone Unit 3 Final Safety Analysis Report Reference 2.3-1.)

2.3.9 CHANNEL DIVERSIONS

(See Section 2.4.9 of the Millstone Unit 3 Final Safety Analysis Report Reference 2.3-1.)

2.3.10 FLOODING PROTECTION REQUIREMENTS

None.

2.3.11 LOW WATER CONSIDERATIONS

2.3.11.1 Low Flow in Rivers and Streams

Since Millstone Unit Number 1 does not depend on either rivers or streams as a source of cooling water, this section is not applicable.

2.3.11.2 Low Water Resulting from Surges, Seiches, or Tsunamis

No effect at Millstone Unit 1.

2.3.12 DISPERSION, DILUTION, AND TRAVEL TIMES OF ACCIDENTAL RELEASES OF LIQUID EFFLUENTS SURFACE WATERS.

(See Section 2.4.12 of the Millstone Unit 3 Final Safety Analysis Report Reference 2.3-1.)

2.3.13 GROUNDWATER

(See Section 2.4.13 of the Millstone Unit 3 Final Safety Analysis Report Reference 2.3-1.)

2.3.14 TECHNICAL SPECIFICATION AND EMERGENCY OPERATION REQUIREMENTS

Station Procedures address necessary precautions and actions to take in the event of anticipated hurricane, tornado, or flood conditions.

2.3.15 REFERENCES

2.3-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.4 - Hydrologic Engineering.

2.3-2 Letter from J. J. Shea to W. G. Counsil, “Millstone Nuclear Power Station, Unit 1 - Safety Evaluation Report on Hydrology SEP Topics II-3.A, II-3.B, II-3.B.1, and II-3.C,” dated June 30, 1982.

2.4 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

2.4.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

Information regarding the geologic and seismic qualities of the Millstone site is presented in Section 2.5.1 of the Millstone Unit Number 3 Final Safety Analysis Report (Reference 2.4-1). That information is incorporated herein by reference.

2.4.2 VIBRATORY GROUND MOTION

Information regarding vibratory ground motion at the Millstone site is presented in Section 2.52 of Reference 2.4-1. With the exceptions given below, that information is incorporated herein by reference.

2.4.2.1 Safe Fuel Storage Earthquake

The design of the plant is such that spent fuel pool remain intact during a ground motion of 0.17 g.

2.4.3 SURFACE FAULTING

2.4.3.1 Geologic conditions of the Site

Section 2.5.1.2 of Reference 2.4-1 discusses the stratigraphy, structural geology, and geologic history of the site are in detail.

2.4.3.2 Evidence of Fault Offset

The published geologic maps which include the site area do not indicate the presence of faulting. A discussion of faults discovered during excavation of the Millstone Unit Number 3 site can be found in Section 2.5.3.2 of Reference 2.4-1. This discussion can be considered typical for the entire Millstone site.

2.4.3.3 Earthquakes Associated with Capable Faults

There is no evidence of capable faults within the five-mile radius of the site. The majority of the significant seismic activity has been associated with the White Mountain Plutonic Province. Some activity has been associated with the Ramapo fault system (Reference 2.4-2); however, the fault is not considered capable (Reference 2.4-3).

2.4.3.4 Investigation of Capable Faults

There are no capable faults within the site area. The faults uncovered in the excavation are discussed in Section 2.5.3.2 of Reference 2.4-1.

2.4.3.5 Correlation of Epicenters with Capable Faults

There has been no spatial correlation between earthquakes and folds in the site region. Some correlation has been suggested with the Ramapo fault in New York and New Jersey. However, the Ramapo is not considered capable (Reference 2.4-3).

2.4.3.6 Description of Capable Faults

There are no capable faults within five miles of the site.

2.4.3.7 Zone Requiring Detailed Faulting Investigation

Eleven incapable fault zones were uncovered during excavation at the Millstone Unit Number 3 site. These faults have been mapped in detail and are discussed in Section 2.5.3.2 of Reference 2.4-1.

2.4.3.8 Results of Faulting Investigation

There is no evidence of capable faulting within the five mile radius of the site. The faults at the site are related to the rifting associated with the Triassic-Jurassic Period or older, with the last activity occurring approximately 142 million years ago.

2.4.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

No information on the stability of subsurface materials and foundations is available from the excavation for Millstone Unit Number 1. A discussion of this subject for the Millstone Unit Number 3 excavation can be found in Reference 2.4-1. This information can be considered typical for the Millstone site.

2.4.5 STABILITY OF SLOPES

The stability of slopes at the Millstone site was evaluated in Reference 2.4-4, wherein it was concluded that there are no natural or man-made slopes at the site that could be or become unstable such as to affect safety related structures, systems or components.

2.4.6 EMBANKMENTS AND DAMS

No embankments or dams have been constructed at the Millstone site.

2.4.7 REFERENCES

2.4-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.5, Geology, Seismology, and Geotechnical Engineering.

2.4-2 Aggarwal, Y.P. and Sykes, L.R. 1978. Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey. Science, Vol. 200, Number 430, pages 425-429.

2.4-3 Nuclear Regulatory Commission 1977. (6 NRC 547 (1977) Atomic Safety and Licensing Appeal Board Hearings on Indian Point, Units 1, 2, and 3 (Dockets Number 50-3, 50-247, and 50-285) ALAB-436.

2.4-4 Nuclear Regulatory Commission. Letter from J. Shea to W. G. Counsil dated June 30, 1982, "SEP Review Topic II-4, D, Stability of Slopes, Millstone Nuclear Power Station Unit 1."

CHAPTER 3 – FACILITY DESIGN AND OPERATION

3.1 DESIGN CRITERIA

3.1.1 CONFORMANCE WITH 10 CFR 50 APPENDIX A GENERAL DESIGN CRITERIA

3.1.1.1 Summary Discussion

The General Design Criteria (GDC) for Nuclear Power Plants as listed in Appendix A to 10 CFR 50 were effective May 21, 1971 and subsequently amended July 7, 1971.

Millstone Unit Number 1, was issued a provisional operating license (POL) on October 7, 1970, and is not obligated to comply with the GDC (Reference 3.1-3). Therefore, Millstone Unit Number 1, is not required to seek exemptions for those areas where it does not comply with the GDC. An evaluation of the design bases of the Millstone Nuclear Unit Number 1, as compared to the GDCs, was performed in support of the application for a full term operating license (FTOL), see Reference 3.1-1. It was concluded therein that Millstone Unit Number 1 satisfies and is in compliance with the intent of the GDCs. Nevertheless, it should be noted that this comparison and conclusion was not a commitment to meet all of the current GDCs or even to meet the intent of the current GDCs. Instead, the Reference 3.1-1 comparison determined the degree of compliance with the GDCs at that time. Also, compliance is demonstrated based upon those interpretations in effect at the time the specific licensing question, or issue, was being addressed.

3.1.1.2 Systematic Evaluation Program and Three Mile Island Evaluations of General Design Criteria

During the systematic evaluation program (SEP) initiated by the NRC in 1977, a large number of generic and plant specific safety concerns were addressed and resolved (Reference 3.1-2). Many of these SEP issues, and later issues which arose from the Three Mile Island (TMI) accident, involved a consideration of the NRC GDC affected by a specific issue and how the plant design compared to the criteria. A compilation of this more recent evaluation of specific safety concerns and the affected GDC are listed in Table 3.1-1.

3.1.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1.2.1 Seismic Classification

The Code of Federal Regulations requires that structures, systems, and components important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions. 10 CFR 100, Appendix A further defines a safe shutdown earthquake (SSE) and the structures, systems and components required to remain functional, as those plant features necessary to ensure:

(1) The integrity of the reactor coolant pressure boundary,

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, and

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to the guideline exposures of 10 CFR 100.

Regulatory Guide 1.29, Revision 3, describes an acceptable method for identifying and classifying those plant features that should be designed to withstand the effects of an SSE.

The plant structures and equipment, including their foundations and supports, are divided into two structural safety categories:

Seismic Class I: structures and equipment whose failure could cause significant release of radioactivity or which are vital to the removal of decay heat.

Seismic Class II: structures and equipment which are not essential to the containment of radioactivity or removal of decay heat.

Only the Reactor Building at and below elevation 108 feet 6 inches , the fuel pool liner and the spent fuel racks remain Seismic Class I in the permanently defueled condition. The Reactor Building houses, protects and supports the spent fuel pool. It supports maintenance of the fuel configuration in the fuel pool, provides protection from external hazards and supports maintenance of water in the fuel pool to a depth necessary to ensure the irradiated fuel is always immersed. The fuel storage racks are designed to assure subcriticality in the fuel pool and are designed to withstand the anticipated earthquake loadings as Class I structures.

The Reactor Building structure above elevation 108 feet 6 inches (enclosure) is classified as seismic Class II in the permanently defueled plant condition. The Reactor Building above elevation 108 feet 6 inches provides a weather enclosure for the spent fuel pool and supports the reactor building overhead crane. However, it has no structural function in providing support for the spent fuel pool. Since the enclosure is no longer credited to provide secondary containment (DSAR Section 3.1.2.2) and since its failure during a seismic event could adversely affect the spent fuel pool and its contents or adjacent safety related SSCs, the seismic design of the enclosure is categorized as Seismic II/I and is further discussed in Section 3.1.6.

Dismantlement of Seismically Designed Structures, Systems and Components

For SSCs designated as Seismic Class I in the permanently defueled condition, the following criteria apply prior to performing dismantlement operations:

(1) Downgrading seismic classification of components shall be performed in accordance with appropriate engineering and design procedures and processes.

(2) When downgrading seismic classification of an SSC, a 10 CFR 50.59 evaluation shall be performed if:

a. the seismic classification is described in the DSAR, or

b. it's failure in a seismic event could affect a Seismic Class I component described in the DSAR in such a manner as to cause an unanalyzed incident or an accident with offsite doses exceeding the doses from the design basis accident.

(3) When downgrading seismic classification of an SSC, a 10 CFR 50.54 evaluation shall be performed if the structure classification is described in the Quality Assurance Program (QAP).

(4) When downgrading seismic classification of an SSC, a 10 CFR 50.59 evaluation shall be performed if, during a seismic event, its failure has the potential to drain the fuel pool water level lower than 9 feet above the active fuel.

3.1.2.2 Safety Related Classification

Nuclear plant SSC have traditionally been classified as "safety related" in accordance with 10 CFR 50.2 and in 10 CFR 100, Appendix A, Section III, if they are relied upon to remain functional during and following design basis events to assure:

- The integrity of the reactor coolant pressure boundary,

- The capability to shut down the reactor and maintain it in a safe shutdown condition,

  or

- The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11.

Clearly, the first two parts of the safety related definition (reactor coolant pressure boundary, and capability to achieve and maintain safe shutdown) do not apply to a permanently defueled plant, given the license restrictions of 10 CFR 50.82. The third part of the safety related definition (accident consequences comparable to 10 CFR 100 guidelines) is dependent on the results of new design basis accident analysis assumptions and results developed to address the existing defueled plant condition. SSC that are required to protect workers and the public from the consequences of design basis events may need to remain classified as safety related.

Types and consequences of potential accidents were reanalyzed and it was concluded that the only remaining accident is a fuel handling accident. This accident was analyzed assuming no secondary containment or standby gas treatment system in operation, with a puff ground level release. The resulting off site radiological exposure was determined to be significantly less than the guideline exposures set forth in 10 CFR 50.34(a)(1) and 10 CFR 100.11. Therefore, no SSC is required to be safety related to prevent or mitigate the consequences of the only remaining accident, except to account for assumptions inherent in the analysis.

The only damage assumed in the analysis is damage to a certain number of fuel bundles. Assumptions inherent to this conclusion are that the fuel configuration, other than the direct impact damage, did not change and that the fuel pool water remained inplace. This implies that there is no failure of either the fuel pool structure or the fuel racks. Since these are passive structural items, assumption of failure is not required as long as the items are safety related and designed to withstand these loads. Therefore, the fuel pool and supporting structure, fuel pool liner, and the fuel racks must be considered as safety related to support the assumptions made in the accident analysis. No other components, systems or structures meet this criterion.

3.1.2.3 Non-Safety Related Plant Functions Maintained in the Defueled Condition

In addition to the Safety Related criterion above, other non-safety related plant functions must be maintained in the defueled condition. The following criteria were used to determine which SSC were still required:

Criterion 1 Is the SSC associated with storage, control or maintenance of nuclear fuel in a safe condition; or handling of radioactive waste?

Criterion 2 Is the SSC program associated with radiological safety?

Criterion 3 Is the SSC associated with an outstanding commitment to the regulators which remains applicable to storage, control, or maintenance of nuclear fuel in a safe condition; or handling of radioactive waste or radiological safety?

Criterion 4 Does the SSC satisfy a requirement based on regulations governing management of nuclear fuel or radioactive materials, including any SSC which is independently required by the License or Technical Specifications?

These criteria were applied to all Millstone Unit Number 1 SSC. A positive response to any criterion, including the Safety Related criterion, results in the group of those SSC which must remain functioning.

3.1.2.4 SSCs Important to the Defueled Condition

All SSC that must remain functional will be maintained in accordance with applicable Millstone Unit Number 1 procedures or quality processes. Commitments exist for augmented quality related to Fire Protection (FPQA), and Radwaste (RWQA). These requirements would apply to the appropriate portions of the SSC which meet the criteria above.

Non-nuclear safety standards apply to other SSC that do not fall under existing quality processes. However, to provide added assurance of adequate reliability for non-safety related SSC that are important to safeguarding the heath and safety of the public and workers, another augmented quality classification is being developed. This classification, Important to the Defueled Condition (ITDC), implements management expectations but does not satisfy any regulatory requirement, and will apply to selected systems and components which perform the following functions:

- Storage, control, maintenance or handling of nuclear fuel
- Storage, control, maintenance or handling of radioactive waste, if not already RWQA
- Radiological safety

Systems and components were reviewed for these functions. Note that application of these functions differs from the 4 criteria in that requirements apply only to the primary SSC and are not extended to supporting systems, equipment or structures. The intent of the ITDC augmented quality is to increase reliable operation of the system(s) primarily responsible for performing each function. Acceptable performance of the supporting SSC are demonstrated during routine operation and/or periodic testing of the ITDC SSC.

Additionally, certain regulatory requirements to which the licensee made a licensing commitment may go beyond the functional scope of an SSC (e.g., Emergency Plan, Security Plan, Quality Assurance Program, etc.). These commitments and legal requirements were also considered in the reclassification process.

<u>Authorizations, Restrictions and Limitations on use of the SSC reclassification criteria</u>

The SSC reclassification criteria is used as a basis to change various Millstone Unit Number 1 procedures and programs, provided that the change involves an SSC that is non-ITDC and, provided that plant procedures contain an acceptable method for approving the change. The following kinds of “soft” changes associated with non-ITDC SSCs are allowed:

- SSC classifications,
- drawings,
- calculations,
- procedures,
- nonconforming items and corrective actions,
- external industry operating experience reports,
- commitments,

- open work orders (in process at the time the decision was made to decommission the plant)
- the application of 10 CFR 50 Appendix B criteria provided it does not represent a reduction in commitment.

Use of these criteria does not authorize:

a. Activities creating new hazards or initiators not already recognized as part of the current license basis (e.g., decontamination or decommissioning of major components defined in 1 CFR 50.82)

b. The physical removal/disassembly of existing SSCs, or the installation of new SSCs. However, it may provide the basis for initiating such a change.

c. Changes to Technical Specification requirements.

d. Changes to regulations, license conditions, rules, and permits until such time that relief is granted by the regulating authority. However, it may provide the basis for requesting relief from the regulations, license conditions, rules, and permits.

e. Changes to commitments. Application of the commitment change process is required to change commitments.

f. Changes to the QAP. However, it may provide the basis for initiating a change to the QAP.

g. Changes to the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM). However, it may provide the basis for initiating a change to the REMODCM.

h. Changes to the Emergency Plan. However, it may provide the basis for initiating a change to the Emergency Plan.

i. Changes to the Security Plan. However, it may provide the basis for initiating a change to the Security Plan.

j. Changes to the Fire Protection Plan. However, it may provide the basis for initiating a change to the Fire Protection Plan.

k. Changes to the Radiation Protection Program. However, it may provide the basis for initiating a change to the Radiation Protection Program.

<u>Boundaries and Interfaces for ITDC SSCs</u>

SSCs identified as ITDC that require “availability” shall be maintained in a state such that the necessary functional capability is maintained.

<u>Engineered Requirements for ITDC SSCs</u>

A higher level of engineered quality is maintained for ITDC SSCs to assure that the capability exists to reliably meet performance expectations and requirements. However, ITDC SSCs are not safety related and are not required to satisfy 10 CFR 50 Appendix B requirements. Although not required by regulation, the following criteria is developed and applied, as specified by engineering, to ITDC SSCs to assure continued reliability:

a. Design Control: Measures will be invoked to assure applicable regulatory requirements, license basis, and design basis information is correctly translated into specifications, drawings, procedures and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Design changes, including field changes will be subjected to engineered design control measures commensurate with the importance of the SSC.

b. Procurement Document Control: Measures will be invoked to assure that applicable regulatory requirements, design basis, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, services.

c. Instruction, Procedures, and Drawings: Activities affecting SSCs will be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and will be accomplished in accordance with these instructions, procedures, and drawings. Instructions procedures, and drawings will include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

d. Control of Purchased Material, Equipment, and Services: Measures will be invoked to assure that material, equipment, and services conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished, inspection at the source, and examination upon delivery.

e. Inspection: Inspection of activities affecting quality will be invoked and executed to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity.

f. Handling, Storage, and Shipping: Measures will be invoked to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

g. Test Control: Surveillance testing will be established for SSCs to ensure that the SSCs perform satisfactorily commensurate with the importance of their intended function.

h. Measuring and Test Equipment: Appropriate controls will be invoked to assure that measuring and test devices used on SSCs are properly controlled, calibrated and adjusted at specified periods to maintain accuracy within necessary limits.

i. Corrective Action: Measures will be invoked to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures will assure that the cause of the condition is determined and corrective action is taken to preclude repetition.

3.1.3 WIND AND TORNADO LOADINGS

10 CFR 50 (General Design Criteria 2), as implemented by Standard Review Plan (SRP) Sections 3.3.1 and 3.3.2 and Regulatory Guides (RG) 1.76 and 1.117, requires that the plant be designed to withstand the effects of natural phenomena such as wind and tornadoes.

The Millstone Unit Number 1 capability to withstand wind and tornado loadings was evaluated in the Systematic Evaluation Program (SEP) (Reference 3.1-4) as Topic III-2. Several submittals were made to the U.S. Nuclear Regulatory Commission (NRC) to address issues raised under that topic (References 3.1-6, 3.1-7, 3.1-8, and 3.1-9). In an evaluation dated November 25, 1985 (Reference 3.1-5), the NRC concluded that the proposal will provide adequate protection against tornado events.

3.1.4 WATER LEVEL DESIGN

The original design basis water level at Millstone Unit Number 1 is the probable maximum flood (PMF) level of 19.0 feet above mean sea level (MSL). In the defueled condition, flooding of Unit 1 structures is acceptable. The spent fuel is stored in the upper elevations of the Reactor Building, and as such is adequately protected from the PMF. The intake structure itself which was originally designed as seismic Class 1, is designed to withstand a water level of elevation 32.4 feet MSL. This level accounts for an assumed 13.4 feet MSL still water level and for non-breaking waves above this level as they strike the structure.

3.1.5 MISSILE PROTECTION

Systems and components have been examined to identify and classify potential missiles.

3.1.5.1 Internally Generated Missiles

Two broad categories of systems and components are reviewed to determine the potential for generating missiles; pressurized components and high speed rotating machinery. Only designs where a single failure could lead to a missile ejection were considered.

It was determined there are no highly pressurized components or high speed rotating machines capable of generating significant missile hazards in the permanently defueled condition. Therefore, no internally generated missiles are postulated.

3.1.5.2 Missiles Generated by Natural Phenomena

The effects of missiles generated by natural phenomena has been evaluated References 3.1-13, 3.1-14, 3.1-15, 3.1-16, and 3.1-17). On the basis of those evaluations, Millstone Unit Number 1 is adequately protected against such missiles inhibiting the ability to maintain safe storage of new and irradiated fuel.

3.1.5.3 Missiles Generated by Events Near the Site

The objective of this assessment (Reference 3.1-19) is to assure that the integrity of the safety related structures, systems, and components will not be impaired and that they will perform their safety functions in the event of a site proximity missile.

The potential for hazardous activities in the vicinity of the Millstone site are addressed in Chapter 2. The licensee concludes that the generation of missiles at these facilities does not pose a credible threat to the Millstone site. Therefore, no specific protection is required other than that described for tornado-generated missiles.

Therefore, Millstone Unit Number 1 does not present an undue risk to the health and safety of the public as a result of proximity missile hazards.

3.1.5.4 Aircraft Hazards

There is presently one small commercial airport approximately 6 miles east-northeast of the site. Groton/New London Airport handles regularly scheduled commercial passenger flights but is inadequate for handling large jets. The licensee has determined that the probability of an aircraft striking safety related structures of Millstone Unit Number 1 is sufficiently low that it does not constitute a significant hazard.

3.1.6 SEISMIC DESIGN

The Millstone Unit Number 1 plant was designed for an earthquake (equivalent to the operating basis earthquake or OBE) with a horizontal peak ground acceleration (HPGA) of 0.07g and reviewed for an earthquake (equivalent to the safe shutdown earthquake or SSE) with a PGA of 0.17 g. A smoothed design response spectrum recommended by John Blume and Associates and the north 69° west component of the 1952 Taft earthquake record normalized to the specified HPGAs were used as seismic input for the analyses and design. The vertical component of ground motion was assumed to be two-thirds of the horizontal components. For the dynamic analyses of seismic Class I structures, the buildings (or structures) were modeled as lumped mass-spring systems with fixed base to simulate the rock founded foundations.

The dynamic responses of the Reactor Building and Radwaste Building/Control were analyzed by time-history approach.

Two methods were used for the analysis of safety related equipment:

(1) the response spectrum analysis approach with smoothed response spectrum recommended by John Blume and Associates as input, and

(2) the equivalent static method using peak structural responses as input.

Chapter 4 of the NRC NUREG/CR-2024 report, "Seismic Review of the Millstone 1 Nuclear Power Plant prepared for the NRC as part of the Systematic Evaluation Program" (Reference 3.1-21), summarizes the details of the original analysis and design.

To assure that the Reactor Building enclosure (structure above elevation 108 feet 6) is capable of withstanding an SSE with a peak ground acceleration of 0.17g without adversely affecting nearby safety related SSCs (Seismic II/I criterion), this portion of the structure is analyzed for the realistic, median-centered in-structure accelerations developed by Vectra Technologies (Reference 3.1-32) for use in the USI A-46 (SQUG) program evaluations of equipment in the Category I portion of the Reactor Building. These floor accelerations and spectra are considered more realistic since they incorporate the variabilities of the input motion at a rock site and the structural parameters (mass and stiffness). The SSE floor accelerations at elevation 82 feet 9 inches (highest elevation evaluated in the Vectra report) are approximately 80% of the corresponding floor accelerations obtained from the EDS Report (Reference 3.1-23). Therefore the floor accelerations at the operating floor and at the roof level are conservatively taken as 80% of the corresponding accelerations from Reference 3.1-23.

3.1.6.1 Comparison of Measured and Predicted Responses

Plant procedures have been developed for abnormal operational events such as earthquakes. If ground motion is detected, plant walkdowns are initiated to determine plant capability.

3.1.7 DESIGN OF CLASS I AND CLASS II STRUCTURES

3.1.7.1 Design Criteria, Applicable Codes, Standards and Specifications

The design of all structures and facilities (Class I and II) conformed to the applicable general codes or specifications in effect at time of design.

3.1.7.2 Loads and Loading Combinations

General requirements for the design of all structures and equipment include provisions for resisting the stresses resulting from dead loads, live loads and wind or seismic loads with impact loads considered as part of the live load. The treatment of equipment stresses are generally limited to those produced by non-operating loads such as the effect of building motion due to earthquake on the anchorage or support for a piece of equipment. However, the loads resulting from operating pressures or temperatures on equipment are considered where they would increase the stresses.

Thermal gradients in the foundation were not considered in the design.

Selection of materials to resist the expected loads is based on standard practice in the power plant field. The use of these materials is governed by local building codes and the experience and knowledge of the designers and builders.

The loadings of concern are the following:

D = Dead load of structure and equipment plus any other permanent loads contributing stress, such as soil, hydrostatic, temperature loads or operating pressures and live loads expected to be present when the plant is operating.

E = Design earthquake load.

E' = Maximum earthquake load.

W = Wind load.

The criteria which have been followed for all Class I structures with respect to stress levels and load combinations for the postulated events are noted below:

Class I portions of Reactor Building and Radwaste Building

(a) D + E — Normal allowable code stresses are used (AISC for structural steel, ACI for reinforced concrete). The customary increase in design stresses, when earthquake loads are considered, is not permitted.

(b) D + E' — Stresses are limited to the minimum yield point as a general case. However, in a few cases, stresses may exceed yield point. In this case, an analysis, using the Limit-Design approach, will be made to determine that the energy absorption capacity exceeded the energy input. This method has been discussed in the AEC publication TID-7024, "Nuclear Reactor and Earthquakes," Section 5.7. The resulting distortion is limited to assure no loss of function and adequate factor of safety against collapse.

(c) D + W — Normal allowable code stresses (AISC for structural steel, ACI for reinforced concrete) with the customary increases in stresses when wind loads are considered.

The maximum allowable stresses used for various loading conditions are given for Class I structures in Table 3.1-2.

Floor live loads were established based upon equipment and operating loads and applied to the basic building code, which is recommended to the boroughs by the State of Connecticut. Roof live loads are a minimum of 60 psf for Class I buildings and 40 psf for Class II buildings.

All Class I structures will withstand the maximum potential loadings resulting from a wind velocity of 115 miles per hour with gusts up to 140 miles per hour. Although some damage to these structures could occur, this damage would under no circumstances impair the functions for the capability of safe storage of irradiated fuel.

Accidental torsion on the structures was not considered in the analyses. The Reactor Building is a box-like structure with heavy columns and thick walls which give it a high torsional rigidity. Accidental eccentricity would therefore produce negligible stresses and has been ignored. Although a lack of symmetry applies to the arrangement of Class I structures, it is felt that because the buildings are not generally structurally connected, torsional effects are likely to be of little consequence.

In the analysis of concrete structures, the design modulus of $3x10^6$ psi is in accordance with the ACI building code requirements for reinforced concrete (ACI 318-63), Section 1102, which is standard design practice. However, it is recognized that the modulus of elasticity of concrete increases with age following the 28 day period, but it is difficult to evaluate the amount of increase. The following factors affect the strength of concrete.

(1) Curing temperature

(2) Initial temperature

(3) Variations in mixes

(4) Amount of hydration

The elastic modulus is not directly proportional to the strength of concrete: nevertheless, the effect of increasing the strength causes an increase in the modulus. However, the increase in the modulus due to age is not believed to be significant in the light of all the uncertainties affecting the modulus of concrete.

Whatever the small change in the modulus may be, this effect is partially accounted for by cracks in the concrete structure due to shrinkage and temperature. Such cracks tend to make the structure more flexible, which tends to compensate for the increased modulus. Also, the percent change in the modulus is small compared to other inputs in the analysis such as dimensions, areas, cross sections, mass grouping, etc. Hence, the effect of a small modulus change on the validity of the dynamic analysis is considered to be negligible.

3.1.7.3 Structural Criteria for Class II Structures

Class II structures and equipment are designed following the normal practice for the design of power plants in the State of Connecticut, but as a minimum, this was not less than given in the "Uniform Building Code" for Zone 2. The usual practice of determining the stress due to earthquakes by applying a static load based on a specified seismic coefficient was followed. The design of the Class II portion of the Reactor Building is addressed in Section 3.1.6.

Allowable stresses for building materials in Class II structures are as specified in the Basic Building Code, which is recommended to the boroughs by the State of Connecticut. A one-third increase is allowed for combinations including seismic or wind loads.

3.1.7.4 Seismic Class I and II Structures

3.1.7.4.1 Reactor Building

Function

The functions of the Reactor Building are to enclose the spent fuel pool and associated equipment and protect it from the weather. It supports maintenance of the fuel configuration in the fuel pool, provides protection from external hazards and supports maintenance of water in the fuel pool to a depth necessary to ensure the irradiated fuel is always immersed.

The Reactor Building at and below elevation 108 feet 6 inches is retained as seismic Class I in the permanently defueled condition. Above elevation 108 feet 6 inches, the building enclosure is revised to seismic Class II with the requirement that the enclosure is capable of sustaining an SSE without collapse (Seismic II/I criterion).

Description

The Reactor Building completely encloses the spent fuel pool. This building is a cast-in-place reinforced concrete structure. At the 108 foot 6 inch elevation, internal steel frame lateral bracing has been placed to support the crane and the roof of the Reactor Building. The Reactor Building is founded on rock with adequate strength at an elevation of minus 32 feet 0 inches, with a foundation of reinforced concrete 142 feet 6 inches square.

The new fuel storage vault and the spent fuel storage pool are located in the Reactor Building. The reactor service and refueling area is serviced by an overhead bridge crane. A refueling service platform with necessary handling and grappling fixtures services the spent fuel storage pool.

Fuel storage pool is a reinforced concrete structure, completely lined with seam-welded stainless steel plate which is welded to reinforcing members embedded in concrete.

The pool is located between elevations 65 feet 9 inches and 108 feet 6 inches. The fuel pool sits on a 5 feet 4 inch thick reinforced concrete slab which is supported by the reactor building perimeter and the primary containment drywell wall. The pool stainless steel liner prevents leakage even in the unlikely event the concrete develops cracks.

The liner was designed considering thermal stress, and the welds were dye penetrant inspected to ensure leak tight integrity. Construction materials used in the construction of the spent fuel storage facility includes 4000 psi, 28 day strength concrete, 40 ksi deformed bar reinforcing steel, and ASTM, A-167, Type 304 stainless steel.

Reactor Building Seismic Design and Analysis

Based on the recommended earthquake design criteria established for the station, envelopes of maximum acceleration, displacement, shear and overturning moment versus height have been developed and are presented for the two assumed earthquake directions. See Figures 3.1-1 through 3.1-5. Based on the data developed by John A. Blume and Associates, engineers, the design criteria have been established as follows for computerized analysis: the mathematical model was subjected to an excursion through the north 69° west component of the 1952 Taft earthquake with an applied factor of 7/17. The resulting maximum shears, moments and displacements were used for design.

The maximum envelopes of building design shears, moments and displacements are presented graphically in Figures 3.1-3 through 3.1-5, respectively. These curves have been used in the seismic design of the Reactor Building. Loads and shears from reactor pressure vessel and associated piping and equipment are transferred to the drywell structure and pedestal and then to the foundation mat. Careful grouting between the drywell and mat ensures direct transfer of compressive loads to the mat. Shears are transferred to the mat by friction and bearing.

The Reactor Building was designed to resist the seismic shears and moments presented herein without the usual increase in stress for short-term loadings. In addition, the structure was reviewed to assure that it can resist 2.4 times the postulated seismic shears and moments without causing injury to the structure. In addition to the horizontal accelerations, a vertical building (and equipment) acceleration was used for design.

The Reactor Building enclosure structure (above elevation 108 feet 6 inches) is analyzed for a realistic median-centered SSE, as described in Section 3.1.6, and is shown to resist the resulting inertial loads from the accelerations with no loss of structural integrity.

3.1.7.4.2 Control Room and Radwaste Treatment Building

Description

The waste treatment facility is north of and adjacent to the reactor building. The building includes equipment and tankage space below grade with the plant control room above grade. The area below grade is of reinforced concrete construction with shielded compartments provided for the various pieces of radwaste equipment. The control room above grade is of reinforced concrete walls with a two foot thick reinforced concrete roof. The control room and radwaste facility are considered seismic Class II. The analytical model used in the seismic analysis of the control room and radwaste building is shown in Figure 3.1-6 and is similar to those for the Reactor Building. The Radwaste Building is seismically analyzed consistent with Regulatory Guide 1.143.

General Structural Features

The building substructure is founded on rock. The maximum bearing pressure on the rock is 10 tons per square foot. The exterior walls are of cast-in-place concrete and designed for an earth pressure per square foot at any depth equal to the depth in feet times 90 pounds. The exterior walls and the base slab were originally designed to resist hydrostatic pressure and uplift due to exterior flooding to elevation 19 feet 0 inches. In the defueled condition, the below grade elevations are allowed to flood such that the uplift force is reduced.

The interior walls of the substructure are of cast-in-place concrete and those for the superstructure are either cast in place or made of concrete masonry units. With minor exceptions, all floors are poured-in-place concrete slabs.

The east half grade floor at elevation 14 feet 6 inches, including the concrete shielding plugs which close hatchways over equipment in the substructure, is designed for a uniform live load of 200 psf.

All tanks are made of ductile metal and all sump pits are lined so that these containers can be subjected to substantial distortion without rupture.

The substructure is massive reinforced concrete, not subject to fracturing. Even in the event fracturing occurred, seepage would be into the building rather than out, since the water table is above basement level.

3.1.7.4.3 Intake Structure

The intake structure is a reinforced concrete frame supported on a reinforced concrete substructure which is founded on rock. The building has a flat roof consisting of 10 gauge steel with concrete slab covered with insulation and a tar and felt roofing membrane. Hatches are provided in the roof for removal of major pieces of equipment. The front wall of the intake structure is designed to resist the standing wave. Seismic stress levels were calculated using coefficients of 0.07 g at grade and 0.12 g at the roof level for design earthquake and 2.4 times these values for the maximum earthquake. The structure is capable of withstanding 300 mph wind but not the tornado internal pressure of 2.5 psi. However, the large number of hatches in the roof will release this pressure. Although originally design as seismic Class I, the intake is considered seismic Class II in the permanently defueled condition.

The intake structure is located west of the main plant and has five 11 foot 2 inch wide bays. Each bay is provided with manually raked trash racks and stop log guides.

Provision for service and cooling water strainers is made in a separate covered pit adjacent to the intake.

3.1.7.4.4 Turbine Building

The Turbine Building is a Class II structure. The Turbine Building foundation consists of a reinforced concrete mat supported on rolled structural steel H section bearing piles. All piles were driven to rock or to refusal in the dense strata immediately above rock. Reinforced concrete shield walls are provided up to the operating deck at elevation 54 feet 6 inches.

The remaining portions of the building have steel framing and metal siding. The Turbine Building ground floor consists of a reinforced concrete slab supported on sand fill over the foundation mats. The turbine generator pedestal is a massive reinforced concrete pedestal designed to support the turbine generator. It is supported on a six foot thick mat which forms an integral part of the remaining building mat foundations. The roof is covered with metal decking, insulation and roofing material flashed at the parapet walls. An overhead rolling door at the west end of the building provides rail car access into the building.

3.1.8 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

None of the plant process instrumentation provides safety related functions in conjunction with the storage and handling of irradiated fuel or radioactive waste, or is credited with any function in the safety evaluations performed to ensure that no undue risk to the health and safety of the public exists. No plant instrumentation or electrical systems are required for mitigation of the design basis fuel handling accident. Seismic qualification of plant instrumentation and electrical equipment is not required.

3.1.9 ENVIRONMENTAL DESIGN OF ELECTRICAL EQUIPMENT

This section is related to qualification of the electrical portion of the engineered safety features to perform their intended functions in the combined normal, accident and post accident environments. There are no non-structural engineered safety features related to the safe storage and handling of the irradiated fuel or radioactive waste, or credited in the safety evaluations performed to ensure that no undue risk to the health and safety of the public exists. No non-structural engineered safety features are credited in accident analysis to prevent or mitigate the consequences of the current design basis fuel handling accident.

3.1.10 REFERENCES

3.1-1 Millstone Nuclear Power Station Unit Number 1 Application for Full Term Operating License, September 1, 1972.

3.1-2 NUREG-0824, Integrated Plant Safety Assessment, Systematic Evaluation Program, Millstone Nuclear Power Station, Unit Number 1, February 1983.

3.1-3 Samual J. Chilk (Nuclear Regulatory Commission) memo to J. M. Taylor (Nuclear Regulator Commission), “SECY-92-233 Resolution of Deviations Identified during the Systematic Evaluation Program” dated September 18, 1992.

3.1-4 Integrated Plant Safety Assessment, Systematic Evaluation Program, Millstone Nuclear Power Station, Unit Number 1, NUREG-0834, Supplement Number 1, November 1985.

3.1-5 Letter, C.I. Grimes (NRC) to J.F. Opeka, subject: IPSAR Sections 4.4 Wind and Tornado Loadings and 4.7 Tornado Missiles.

3.1-6 Letter, February 2, 1984, from W.G. Counsil to D. M. Crutchfield (NRC), Subject: Millstone Nuclear Power Station Unit Number 1, SEP Topics II-3.B Flooding Potential and Protection Requirements, III-2 Wind and Tornado Loadings, III-3.A Effects of High Water Level on Structures, III-7.B Design Codes, Design Criteria and Load Combinations.

3.1-7 Letter, March 16, 1984, from W. G. Counsil to D. M. Crutchfield (NRC), Subject: Millstone Nuclear Power Station Unit Number 1, SEP Topics II-3.B Flooding Potential and protection Requirements, II-4.F Settlement of Foundations and Buried Equipment, III-2 Wind and Tornado Loadings, III-3.A Effects of High Water Level on Structures, III-6 Seismic Design Considerations.

3.1-8 Letter, October 7, 1983, from W. G. Counsil to D. M. Crutchfield (NRC), Subject: Millstone Nuclear Power Station Unit Number 1, SEP Topic III-2 Wind and Tornado Loadings.

3.1-9 Letter, December 3, 1982, from W. G. Counsil to D. M. Crutchfield (NRC), Subject: Millstone Nuclear Power Station Unit Number 1, SEP Topic III-2 Wind and Tornado Loadings.

3.1-10 Letter, February 4, 1986, from J. F. Opeka to C.I. Grimes (NRC), Subject: Millstone Nuclear Power Station Unit Number 1 ISAP Topic 1.19, Integrated Structural Analysis.

3.1-11 Letter, W. G. Counsil to D. M. Crutchfield (NRC), dated March 16, 1984, Millstone Nuclear Power Station, Unit Number 1, SEP Topic II-3.B, Flooding Potential and protection Requirements, SEP Topic II-4.F, Settlement of Foundations and Buried Equipment, SEP Topic III-2, Wind and Tornado Loadings, SEP Topic III-3.A, Effects of High Water Level on Structures SEP Topic III-6, Seismic Design Considerations.

3.1-12 10 CFR 50, Appendix A, General Design Criterion 4.

3.1-13 Letter of June 29, 1982, W.G. Counsil to D. M. Crutchfield,: Millstone Nuclear Power Station Unit Number 1, SEP Topic III-4.A, Tornado Missiles."

3.1-14 Letter of March 9, 1982, W.G. Counsil to D. M. Crutchfield,: Millstone Nuclear Power Station Unit Number 1, SEP Topic III-4.A, Tornado Missiles."

3.1-15 Letter of November 19, 1981, W.G. Counsil to D. M. Crutchfield: Millstone Nuclear Power Station Unit Number 1, SEP Topic III-4.A, Tornado Missiles."

3.1-16 Letter of August 31, 1981, W.G. Counsil to D. M. Crutchfield,: Millstone Nuclear Power Station Unit Number 1, SEP Topic III-4.A, Tornado Missiles."

3.1-17 Letter of October 16, 1985, J. F. Opeka to C. I. Grimes, Millstone Nuclear Power Station Unit Number 1, "Integrated Safety Assessment Program."

3.1-18 Letter of November 25, 1985, C. I. Grimes to J. F. Opeka, Integrated Plant Safety Assessment Report, Section 4.4, Wind and Tornado Loadings, Section 4.7, Tornado Missiles - Millstone Unit Number 1.

3.1-19 Letter of April 29, 1981, W. G. Counsil to D. M. Crutchfield, “Millstone Nuclear Power Station Unit Number 1, SEP Topic III-4.D, Site Proximity Missiles.”

3.1-20 Letter of September 17, 1981, W. G. Counsil to D. M. Crutchfield, “Millstone Nuclear Power Station Unit Number 1, SEP Topic III-4.D, Site Proximity Missiles.”

3.1-21 NRC NUREG/CR-2024 Report, “Seismic Review of the Millstone-1 Nuclear Power Plant,” July 1981.

3.1-22 SEP Safety Topics III-6, Seismic Design Considerations and III-II, Component Integrity - Millstone Nuclear Power Station Unit Number 1, SAR dated 6/30/82.

3.1-23 EDS Report Number 02-0240-1094, “Generation of In-Structure Seismic Response Spectra Millstone Unit Number 1,” dated June 1982.

3.1-24 NRC letter,”Site Specific Ground Response Spectra for SEP Plants Located in the Eastern United States,” June 17, 1981.

3.1-25 NRC NUREG/CR-1582 Report, “Seismic Hazard Analysis,” Vols. 2-4, October 1981.

3.1-26 Letter J. F. Opeka to C.I. Grimes, “Millstone Nuclear Station, Unit Number 1 ISAP Topic 1-19, Integrated Structural Analysis,” dated January 6,1986.

3.1-27 Letter from D. G. Eisenhut, NRC, to W. G. Counsil, dated January 1, 1980.

3.1-28 Letter from D. M. Crutchfield, NRC, to W. G. Counsil, dated July 28, 1980.

3.1-29 Letter from W. G. Counsil to D. M. Crutchfield, NRC, dated October 16, 1985.

3.1-30 Millstone Unit 2 Final Safety Analysis Report Section 5.8.6.

3.1-31 Millstone Unit 3 Final Safety Analysis Report Section 3.7.4.2.

3.1-32 Vectra Technologies Report Number 0024-00099-RB-1, Rev. 1, “Reactor Building A-46 Spectra,” dated June 10, 1996.

TABLE 3.1–1 COMPARISON WITH NRC GENERAL DESIGN CRITERIA

GENERAL DESIGN CRITERIA		SEP AND TMI SAFETY ISSUES WHICH LISTED THE SPECIFIED GDC AS PART OF CONCERN	AFFECTED REGULATORY GUIDE
I OVERALL REQUIREMENTS			
1	QUALITY STANDARDS AND RECORDS	SEP II-3.A, II-3.B, II-3.C, III-3.A AND III-7.B	1.27, 1.59
2	DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA	SEP II-2.A, II-3.A, II-3.B, II-3.C, II-4.E, II-4.F, II-4.3, III-19 III-2, III-3.A, III-3.B, III-3.C, III-6, III-7.B, III-8.C, III-11, VIII-3.A, VIII-3.B, TMI II.B.1	1,27, 1.,32, 1.59, 1.60, 1.61, 1.68, 1.75, 1.76, 1.92, 1.102, 1.117, 1.120, 122, 1.127, 1.129, 1.132
3	FIRE PROTECTION	(SEE DSAR Section 3.2.9)	
4	ENVIRONMENTAL AND MISSILE DESIGN BASES	SEP II-1.C, II-3.A, II-3.B, II-3.C, III-1, III-4.B, III-5.A, III-5.B, III-7,B, III-11, V-5, VIII-3.A, VIII-3.B, TMI II.B.2, II.B.3, 2.1.6.A, 2.1.8.A, III-4,A	1.3, 1.4, 1.7, 1.20, 1.27, 1.29, 1.32, 1.35, 1.45, 1.46, 1,59, 1.68, 1.75, 1.115, 1.12,
5	SHARING OF STRUCTURES, SYSTEMS AND COMPONENTS	SEP III-1, VIII-3.A AND VIII-3.B	1.32, 1.75, 1.129
II PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS			
10	REACTOR DESIGN	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
11	REACTOR INHERENT PROTECTION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
12	SUPPRESSION OF REACTOR POWER OSCILLATIONS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
13	INSTRUMENTATION AND CONTROL	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
14	REACTOR COOLANT PRESSURE BOUNDRY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
15	REACTOR COOLANT SYSTEM DESIGN	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
16	CONTAINMENT DESIGN	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
17	ELECTRIC POWER SYSTEMS	SEP III-1, VII-7, VIII-2, VIII-3.A VIII-3.B, TMI II.E.3.1, II.G.1	1.6, 1.9, 1.32, 1.75, 1.129
18	INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
19	CONTROL ROOM	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION

TABLE 3.1–1 COMPARISON WITH NRC GENERAL DESIGN CRITERIA

GENERAL DESIGN CRITERIA		SEP AND TMI SAFETY ISSUES WHICH LISTED THE SPECIFIED GDC AS PART OF CONCERN	AFFECTED REGULATORY GUIDE
III PROTECTION AND REACTIVITY CONTROL SYSTEMS			
20	PROTECTION SYSTEM FUNCTIONS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
21	PROTECTION SYSTEM RELIABILITY AND TESTABILITY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
22	PROTECTION SYSTEM INDEPENDENCE	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
23	PROTECTION SYSTEM FAILURE MODES	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
24	SEPARATION OF PROTECTION AND CONTROL SYSTEMS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
25	PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
26	REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
27	COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
28	REACTIVITY LIMITS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
29	PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
IV FLUID SYSTEMS			
30	QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
31	FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
32	INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
33	REACTOR COOLANT MAKEUP	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION

TABLE 3.1–1 COMPARISON WITH NRC GENERAL DESIGN CRITERIA

GENERAL DESIGN CRITERIA		SEP AND TMI SAFETY ISSUES WHICH LISTED THE SPECIFIED GDC AS PART OF CONCERN	AFFECTED REGULATORY GUIDE
34	RESIDUAL HEAT REMOVAL	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
35	EMERGENCY CORE COOLING	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
36	INSPECTION OF EMERGENCY CORE COOLING SYSTEM	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
37	TESTING OF EMERGENCY CORE COOLING SYSTEM	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
38	CONTAINMENT HEAT REMOVAL	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
39	INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
40	TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
41	CONTAINMENT ATMOSPHERE CLEANUP	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
42	INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
43	TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
44	COOLING WATER	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
45	INSPECTION OF COOLING WATER SYSTEM	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
46	TESTING OF COOLING WATER SYSTEM	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
V REACTOR CONTAINMENT			
50	CONTAINMENT DESIGN BASIS	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
51	FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION

TABLE 3.1–1 COMPARISON WITH NRC GENERAL DESIGN CRITERIA

GENERAL DESIGN CRITERIA		SEP AND TMI SAFETY ISSUES WHICH LISTED THE SPECIFIED GDC AS PART OF CONCERN	AFFECTED REGULATORY GUIDE
52	CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
53	PROVISIONS FOR CONTAINMENT INSPECTION AND TESTING	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
54	SYSTEMS PENETRATING CONTAINMENT	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
55	REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
56	PRIMARY CONTAINMENT ISOLATION	NOT APPLICABLE TO THE PERMANENTLY DEFUELEDCONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
57	CLOSED SYSTEMS ISOLATION VALVES	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION	NOT APPLICABLE TO THE PERMANENTLY DEFUELED CONDITION
VI FUEL AND RADIOACTIVITY CONTROL			
60	CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT	SEP II.2.C, XI-1, XI-2, TMI II.B.2, II.B.3, 2.1.6.A, 2.1.8.A	1.3, 1.4
61	FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL	SEP XI-1, XI-2	
62	PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING		
63	MONITORING FUEL AND WASTE STORAGE	SEP XI-1, XI-2	
64	MONITORING RADIOACTIVITY RELEASES	SEP II-2.C, XI-1, XI-2; TMI II.B.2, II.B.3, 2.1.6.A, 2.1.8.A	

TABLE 3.1–2 ALLOWABLE STRESSES FOR CLASS I STRUCTURES

	LOADING CONDITIONS	REINFORCING STEEL MAX. ALLOWABLE STRESS	CONCRETE MAX. ALLOWABLE COMPRESSION STRESS	CONCRETE MAX. ALLOWABLE SHEAR STRESS	CONCRETE MAX. ALLOWABLE BEARING	STRUCTURAL STEEL TENSION ON THE NET SECTION	STRUCTURAL STEEL SHEAR ON GROSS SECTION	STRUCTURAL STEEL COMPRESSION ON GROSS SECTION	STRUCTURAL STEEL BENDING
(1)	DEAD LOADS PLUS LIVE LOADS,* PLUS OPERATING LOAD PLUS SEISMIC LOADS (0.07g)	0.5 Fy	0.45 f ' c	1.1 f ' c	0.25 f ' c	0.60 Fy	0.40 Fy	VARIES WITH SLENDERNESS RATIO	0.66 Fy TO 0.60 Fy
(2)	DEAD LOADS PLUS LOADS, * PLUS OPERATING LOADS PLUS WIND LOADS	0.667 Fy	0.60 f ' c	1.467 f ' c	0.333 f ' c	0.80 Fy	0.53 Fy	VARIES WITH SLENDERNESS RATIO	0.88 Fy TO 0.80 Fy
(3)	DEAD LOADS PLUS LIVE LOADS,* PLUS OPERATING LOADS, PLUS SEISMIC LOADS 0.17g			GROSS STRUCTURAL INTEGRITY CAN BE MAINTAINED (SEE NOTE 1 BELOW)					

25% of live loads were considered concurrent with the seismic loads

Fy = Minimum Yield Point of the Material.

f ' c = Compressive Strength of Concrete.

NOTE 1: The structure was analyzed to assure that gross structural integrity can be maintained during ground motion having 17/7 the intensity of the operating basis earthquake described in SECTION 3.1.6, even though stresses in some of the materials may exceed the yield point.

3.2 SYSTEMS

3.2.1 FUEL STORAGE AND HANDLING

3.2.1.1 New Fuel Storage

Since Millstone Unit 1 is a de-commissioned unit, new fuel will no longer be received.

3.2.1.2 Spent Fuel Storage

3.2.1.2.1 Design Bases

The design bases for the storage of spent fuel are as follows:

a. A fuel storage pool for the underwater storage of 2959 fuel assemblies.

b. Maintain a k_{eff} of less than 0.95 at all times, including postulated criticality accidents. Assumed are worst case results, considering maximum variation in the position of the fuel assemblies within the storage rack, neutron absorber variation (where credited), seismic induced deflections and calculation uncertainty. Boraflex is not credited.

c. The concrete shielding walls are designed as part of the Class 1 portion of the Reactor Building structure. The thickness of the walls and the standards of design are such as to preclude structural damage or loss of function of the walls.

d. Structural design of the fuel storage and equipment storage facilities meets all requirements for Class I structures.

e. The fuel storage racks for the fuel are designed to assure subcriticality in the fuel pool. The storage racks are an interconnected honeycomb array of square stainless steel boxes forming individual cells for fuel storage. 1045 storage cells contain Boraflex sheets (not credited) on four sides, and 2184 storage cells contain B_4C plates for neutron absorption. Of the 1045 storage cells with Boraflex, only 775 cells are allowed to contain fuel.

f. Criticality Accident Requirements. Millstone 1 has chosen to comply with 10 CFR 50.68(b).

3.2.1.2.2 Facilities Description

The fuel pool contains water which is not borated. The fuel storage pool is a reinforced concrete structure, completely lined with seam-welded, stainless steel plate (11 gauge) which is welded to reinforcing members (channels, I-beams, etc.) embedded in concrete. The liner is reinforced by increased thickness and suitable insert strips in areas subject to heavy loading such as the cask handling area The concrete shielding walls are two or more feet thick and are designed as part of the Class I portion of the Reactor Building structure.

Interconnecting drainage paths are provided behind the liner welds to:

(1) Prevent pressure buildup behind the liner plates,

(2) To control the loss of pool water and

(3) To provide liner leak detection and measurement capability.

The drainage paths are suitably grouped to indicate the area of leakage. To avoid unintentional draining of the pool, there are no penetrations that would permit the pool to be drained below approximately nine feet above the top of the active fuel, and all lines extending below this level are equipped with suitable valving to prevent backflow. The passage between the fuel storage pool and the refueling cavity above the reactor vessel is provided with two gates. The refueling cavity is maintained in a drained down state. The gate adjacent to the refueling cavity is welded to the passage liner forming a permanent pressure boundary for the fuel storage pool. The double sealed gate adjacent to the fuel storage pool is removable but normally maintained in the closed position. A normally open drain line between the gates permits detection of leaks from the gate adjacent to the fuel storage pool. The drain line may be isolated and the volume between the gates flooded to support removal of the gate for repairs in the event of such leakage.

In response to the NRC I.E. Bulletin 84-03, augmented leak detection capability has been installed in the spent fuel pool to indicate high/low level in the pool.

The water in the pool is cooled and filtered as required by the spent fuel pool cooling and in-pool cleanup system described in Subsection 3.2.1.3.

The storage pool is designed to hold 20 fuel channels.

An area of approximately seven feet by seven and one half feet is reserved for loading a spent fuel shipping cask.

Canisters containing irradiated reactor vessel internals and other materials classified per 10 CFR 61 as greater than class “C” (GTCC) waste are stored in the fuel storage pool adjacent to the fuel shipping cask area.

3.2.1.2.3 Safety Evaluation

The spacing of fuel bundles in the spent fuel storage pool, the presence of neutron absorbing poisons (where credited) in the fuel storage racks, not placing fuel in prohibited locations identified in the Technical Specifications, and the design of the fuel bundles maintains k_{eff} less than or equal to 0.95. This is assured by limiting the fuel assemblies in the pool to those that have a maximum K_∞ of 1.24 in the normal reactor configuration at cold conditions, and an average U-235 enrichment of 3.8 weight percent or less. The criticality analysis confirms acceptable results regardless of the spent fuel pool temperature.

Irradiated fuel being moved in the fuel storage pool is covered by an eight foot minimum of water above the top of active fuel, which is sufficient for radiation shielding. Radiation monitors in the fuel storage pool work area monitor the radiation level and alarm upon excessive levels.

Limit switches on the refueling platform hoists interrupt power to the hoists when raising fuel, at a point that ensures a minimum of eight feet of water above the top of active fuel. The brakes on the refueling platform equipment lock upon loss of power.

The fuel storage racks are analyzed to withstand the impact of a dropped fuel assembly and handling tool with a combined dry weight of 1675 pounds from the maximum lift height of the refueling platform telescoping mast. The analyses performed (References 3.2-9 to 3.2-12) demonstrate that the spent fuel racks remain functional and that the spent fuel remains in a subcritical, submerged and coolable condition.

A liquid level transmitter, monitoring pool water level, is provided to detect loss of water from the pool. A level transmitter, monitoring the skimmer surge tank, is provided to permit water loss detection by initiating a low level alarm and provide level indication in the Millstone Unit 2 Control Room.

A safety evaluation of spent fuel can be found in References 3.2-1, 3.2-2, 3.2-3., 3.2-4, 3.2-5, and 3.2-8.

3.2.1.3 Spent Fuel Pool Cooling System

The Spent Fuel Pool Cooling System has been analyzed to remove the maximum heat load from the spent fuel pool.

3.2.1.3.1 Design Bases

The Fuel Pool structure, pool liner, fuel racks, and external cooling system have been designed for a temperature of approximately 150°F. However, all of these structures and components have been demonstrated to be structurally adequate for abnormal temperature excursions to 212°F. With a complete loss of external cooling and a closed airspace above the pool, it would take approximately 10 days for the pool temperature to rise to 212°F from an initial SFP bulk water temperature of 100°F, or approximately 7.5 days to rise to 212°F if starting from the TRM upper temperature limit of 140°F. The spent fuel pool cooling system and secondary DHR cooling system have been qualified for satisfactory operation with pool temperatures as high as 170°F. This is greater than the maximum anticipated pool water temperature, following loss of cooling, provided that natural ventilation within the reactor building is established within approximately 5 days if starting from an initial SFP bulk water temperature of 100°F, or 2.5 days if starting from the TRM upper temperature limit of 140°F.

Multiple methods are available to add water to the pool and adequate time is available to repair, manually reinstate or line up the system used for pool water cooling. Most significantly, if this system is not used to cool the pool water, no fuel damage would result and the potential off site exposure would not approach the guidelines established in 10 CFR 50.34(a) or 10 CFR 100.11

provided makeup is initiated at a rate equal to or greater than the maximum evaporation rate at any time prior to fuel uncovery. Water above the fuel provides shielding and heat sink functions.

For the permanently defueled condition, the design bases for the fuel pool cooling system is:

a. To maintain the bulk water temperature for the spent fuel pool at a temperature less than or equal to 140°F and greater than 68°F.

b. To provide high clarity water to the fuel pool using the in-pool cleanup system.

c. To remove radioactivity released to the pool water using the in-pool cleanup system.

3.2.1.3.2 Spent Fuel Pool Heat Load

Millstone Unit Number 1 has permanently ceased power operation and all irradiated fuel has been permanently removed from the reactor vessel. There are 2885 irradiated fuel assemblies in the spent fuel pool including one segmented bundle, consisting of 19 fuel rods. A decay heat load calculation was performed utilizing the computer program ORIGEN2, an industry standard for such analysis (Reference 3.2-13). The results show that total heat load in the pool was 1.781 MBtu/hr on 1/1/99. The spent fuel pool secondary cooling system (DHR) has been sized to remove the spent fuel decay heat load of approximately 1.5 Mbtu/hr, projected to exist on 6/1/00.

3.2.1.3.3 Loss of Fuel Pool Cooling

With the spent fuel pool heat load established, a second calculation (Reference 3.2-14) was performed to determine the transient and steady state spent fuel pool and reactor building temperatures without active cooling to the spent fuel pool. Several cases were analyzed with different ventilation configurations such as forced ventilation, natural ventilation and no ventilation through the building. Steady state and transient calculations were performed to establish maximum pool and building temperatures and evaporation rates, as well as time frames for potential operator actions. All analyses were performed using the GOTHIC computer program.

The limiting case evaluated was during summer conditions (92°F, 50% Relative Humidity) following the loss of active spent fuel pool cooling and without the reactor building HVAC system in operation. In this case the time to reach 212°F in the spent fuel pool is approximately 7.5 days if starting from the TRM upper temperature limit of 140°F. This calculation also establishes a maximum evaporative loss of 3.8 gpm under the above conditions. If natural ventilation is established, by opening the reactor building truck bay doors, equipment hatch garage doors and the tornado dampers on the reactor building roof, the maximum calculated pool temperature is 163°F and the maximum evaporation rate is 3.0 gpm.

3.2.1.3.4 System Description

The spent fuel pool cooling system cools water in the fuel pool on an as needed basis to maintain water temperature. An in-pool demineralizer and filter maintain purity and water quality. Water is circulated by either one or two pumps which take suction from the skimmer surge tanks. The adjustable spent fuel pool weir gates maintain pool level and skim water from the surface of the fuel pool. System lineups may vary due to decreasing heat removal needs. The flow diagram for the spent fuel pool cooling system is shown in Figures 3.2-1 through 3.2-3.

The bulk temperature monitoring system consists of a single RTD (combined temperature transmitter and level transmitter) with the sensing element located approximately 9' below the normal water level of the fuel storage pool, and a local temperature indicator. The transmitter output is monitored in the Millstone Unit 2 Control Room via the Programmable Logic Controller (PLC) which provides both indication of bulk temperature and notification of a high and low water temperature conditions within the fuel storage pool.

The in-pool fuel pool demineralizer operates on an as needed basis to maintain pool water chemistry. The in-pool filter operates on an as needed basis to maintain pool water clarity. The skimmer surge tanks are shielded with concrete.

The fuel pool cooling system is controlled and operated locally and from the Millstone Unit 2 Control Room. The system is provided with indicators and alarms for system flow, water level, and temperature, skimmer surge tank level, and component operating status.

3.2.1.3.5 Safety Evaluation

The fuel pool water acts passively to transfer decay heat from the fuel and will protect the fuel from damage without human intervention as long as the fuel is completely immersed in water. If external cooling is stopped, the pool water temperature would gradually increase, resulting in no fuel damage. In the most severe case of a closed airspace, with the current decay heat load in the Millstone Unit Number 1 Fuel Pool and no external cooling, the pool temperature would only reach equilibrium (stop rising) when the pool water boils, which is the natural limit of water temperature in a space at atmospheric pressure. The fuel pool structure, pool liner, fuel racks, and external cooling system have been demonstrated to be adequate for abnormal temperature excursions to 212°F. With a complete loss of external cooling and a closed airspace above the pool, it would take approximately 10 days for the pool temperature to rise to 212°F from an initial SFP bulk water temperature of 100°F, or approximately 7.5 days to rise to 212°F if starting from the TRM upper temperature limit of 140°F. This is significantly longer than required to reinstate external cooling of the water. If natural ventilation is established, by opening the reactor building truck bay doors, equipment hatch garage doors and the tornado dampers on the reactor building roof, the maximum calculated pool temperature is 163°F.

3.2.1.4 Fuel Handling System

3.2.1.4.1 Design Bases

The design bases for the fuel handling system are as follows:

a. No release of contamination or exposure of personnel to radiation will exceed the 10 CFR 20 limits.

b. Limited work on irradiated components will be possible at any time.

3.2.1.4.2 System Description

The fuel handling system handles irradiated fuel.

A refueling platform, equipped with a refueling grapple and two one-half ton auxiliary hoists is provided for servicing the fuel storage pool. The operating floor is serviced by the Reactor Building crane, which is equipped with a 110 ton main hoist and a seven-ton auxiliary hoist. These hoists can reach any major equipment storage area on the operating floor.

3.2.1.4.3 Safety Evaluation

The refueling bridge and other fuel handling equipment are required for movement of fuel and other items stored in the fuel pool into storage/shipping containers. The reactor building crane is required to move storage and shipping casks in the reactor building. These functions are required in the permanently defueled condition, but are not safety related.

3.2.2 MONITORING AND CONTROL FUNCTIONS

The Millstone Unit 2 Control Room serves as the control room for Millstone Unit 1, and is continuously manned. It is described in Section 7.6 of the Millstone Unit 2 Final Safety Analysis Report. Millstone Unit 2 Operations personnel are responsible for the monitoring and control of the Unit 1 spent fuel pool island (SFPI) and auxiliary systems via a computer console located in the Millstone Unit 2 Control Room. The computer console in the Millstone Unit 2 Control Room interfaces with a Programmable Logic Controller (PLC) for data acquisition and trending. The PLC is located in the Millstone Unit 1 Central Monitoring Station (CMS). The CMS is located within the Maintenance Shop.

The Millstone Unit 1 CMS is not manned. It contains two computer consoles that may only be used as monitors, because they are normally in a locked supervisory mode.

There are no monitoring stations in the original Unit 1 Control Room. The original Unit 1 Control Room no longer performs any Unit 1 function.

3.2.3 DECAY HEAT REMOVAL (DHR) SYSTEM

3.2.3.1 Design Bases

The DHR system is designed to provide cooling to the spent fuel pool cooling system. The system design bases are:

Design Temperature:	170°F
Design Flow Rate (maximum)	625 gpm per pump

Design Pressure: 200 psig

The DHR system is normally in service to supply spent fuel pool cooling system cooling loads as needed. System lineups vary during the permanently defueled condition due to reduced heat removal needs.

3.2.3.2 System Description

The DHR system provides a supply of cooling water to the shell side of the spent fuel pool heat exchangers. Water is circulated in a closed loop by the DHR pumps. Heat is removed from the system by the four DHR air-water heat exchangers located outside on the roof above the H&V area. System configuration may vary depending on heat load. The remainder of the system consists of a cooling water expansion tank, an air separator, piping and valves, and controls and instrumentation. A demineralizer maintains system activity below established limits. The flow diagram for the system is shown in Figure 3.2-4.

3.2.3.3 Safety Evaluation

The DHR system supplies cooling water to the fuel pool heat exchangers. Fuel pool cooling is a function that is required for the permanently defueled condition, but is not safety related. Therefore, this function of the DHR system is not safety related.

3.2.3.4 Testing and Inspection

The system components and instrumentation are tested periodically as necessary to ensure operational readiness.

3.2.3.5 Instrumentation

DHR system instrumentation and controls are located locally and in the Millstone Unit 2 Control Room.

3.2.4 MAKEUP WATER SYSTEM

3.2.4.1 Demineralized Water

3.2.4.1.1 System Description

The spent fuel pool makeup system will supply and store demineralized water to makeup for evaporation and leakage in the pool. The primary source will be from the Unit 2 Primary Makeup System which is supplied from the onsite water treatment facility. A 5,000 gallon storage tank and transfer pump are installed in the reactor building to provide makeup water to the spent fuel pool during period when the normal makeup from Unit 2 is unavailable. A connection to the pool makeup line is also provided near the reactor building truck bay door to allow makeup to be provided by a tanker truck or fire water if necessary.

The piping, tanks and other equipment of the spent fuel pool water storage and makeup system and makeup system are of corrosion resistant metals which prevent contamination of the makeup water with foreign material.

The flow diagram for the system is shown in Figure 3.2-5.

3.2.4.1.2 Safety Evaluation

The spent fuel pool makeup water system provides demineralized makeup water to the spent fuel pool and spent fuel pool cooling system. This function supports fuel pool cooling, but is not safety related.

3.2.4.1.3 Testing and Inspection

Operation of the makeup system is on demand at intermittent intervals to replenish water in the spent fuel pool makeup water storage tank and the skimmer surge tanks. The equipment is visually inspected periodically. Sampling of the makeup water storage tank is a standard monitoring procedure.

3.2.4.1.4 Instrumentation

The motor control switch for the makeup water transfer pump is located locally at the pump. Local makeup storage tank level indication is also provided.

3.2.5 INTENTIONALLY DELETED

3.2.6 PROCESS SAMPLING SYSTEM

3.2.6.1 Design Bases

The reason for sampling process gases is to provide representative samples for testing to obtain data from which the performance of the plant equipment and systems are determined.

3.2.6.2 System Description

The Unit Number 1 BOP ventilation exhaust flow is continuously sampled for radioactive particulates. The sample is taken from the exhaust duct which runs along the north exterior wall of the Reactor Building. A single point sample nozzle is positioned to obtain a representative sample of the well mixed exhaust air. The sample passes through a particulate filter and is then expelled back into the exhaust duct.

The SFPI ventilation exhaust flow is continuously monitored for gaseous radiation and particulates. The sample is taken from the exhaust duct near the reactor building exhaust plenum. A single point sample nozzle is positioned to obtain a representative sample of the turbulent and well mixed exhaust air. The sample passes through a particulate filter and a gas monitor and is then expelled back into the exhaust duct.

Grab samples can be taken from the BOP and SFPI ventilation exhaust ducts and analyzed for radioactive content.

3.2.6.3 Safety Evaluation

The BOP and SFPI ventilation systems are not safety related.

3.2.6.4 Testing and Inspection

Functional tests were performed after installation. Routine use substitutes for subsequent periodic testing, with the exception of calibration and maintenance.

3.2.7 ELECTRICAL SYSTEMS

3.2.7.1 Introduction

The station electrical systems include the equipment and facilities which provide power to desired plant equipment, instrumentation and controls. The system is designed to provide reliable power for the permanently defueled condition. The power system is designed with a sufficient source, relay protection, control, and necessary switching.

3.2.7.2 Off Site Source

The off site source is through the emergency station service transformer (ESST), which steps down a 23 kV source from the Waterford Substation 36F2 circuit to 4160V.

The off site power system is designed to provide a reliable source of power to the on site AC power distribution system.

3.2.7.3 Intentionally Deleted.

3.2.7.4 On Site Electric System

3.2.7.4.1 Introduction

Sufficient time is available to operators following a loss of offsite power to assure the continued safe storage of fuel without reliance on emergency sources of power.

AC power is provided through the emergency station service transformer. The emergency station service transformer has adequate capacity to supply all normal auxiliaries required to support the permanently defueled condition. Power for the SFPI and other decommissioning related activities is from the ESST via Bus 14H.

SFPI and decommissioning related 125V DC power is obtained from rectified AC power at the point of use, and a separate 125V DC source consisting of a 125V DC battery, a battery charger, disconnect switch and distribution panel.

3.2.7.4.2 4160 Volt System

The emergency station service transformer (15G-31S) steps down 23 kV to 4160 volts for the auxiliary buses. In the permanently shutdown condition, the plant will normally be operated with auxiliary electrical loads supplied from the emergency station service transformer.

The circuit breakers located on 4160 V bus 14H are operated locally at the switchgear. Breakers will trip automatically when over-current conditions exist. The control power to the 4160 volt bus circuit breaker is from the decommissioning 125 volt DC system.

The major component of the 4160 volt power system is described below.

(1) Emergency Station Service Transformer

The emergency station service transformer is an outdoor, 27,750-4160 volt three phase, 60 Hz., 200 kV BIL, 10.7/12.5 MVA OA/FA 55°C, and 14 MVA, FA 65°C, transformer.

3.2.7.4.3 480 Volt System

Power from 4160 volt bus 14H is stepped down through transformers energizing the 480 volt buses SFPI-B1 and FAC-B2.

Bus and MCC supply breakers are opened and closed locally. All breakers will trip automatically when overload conditions exist.

3.2.7.4.4 120 Volt Systems

The SFPI electrical system utilizes its own dedicated 120V AC power derived from the SFPI AC power system.

The SFPI instrument AC system is provided by the SFPI 120V AC distribution system and backed up by point of use UPS equipment. The SFPI PLC system has an integral 24V DC power supply.

3.2.7.4.5 AC Power System Design Criteria

(1) Interrupting Capacity - The switchgear, load centers, motor control centers, and distribution panels are sized for interrupting capacity based on maximum short circuit availability at their location. Low voltage metal enclosed breakers at load centers and molded case breakers at motor control centers are adequately sized for these maximum available short circuit currents.

(2) Electrical System Protection - Electrical system protection is provided by protective devices or relays which monitor the electrical characteristics of the equipment and/or power system to assure operation consistent with design parameters, as follows:

(a) Initiate removal from service any piece of equipment which has sustained a fault.

(b) Provide automatic supervision of manual and/or automatic operations which could jeopardize the safe operation of the plant.

3.2.7.4.6 125 V DC System

SFPI related 125V DC utilizes rectified AC power. The rectifiers are located at the SFPI 480V AC switchgear bus. In addition, the decommissioning 125V DC system consists of a 125V DC battery, charger, disconnect switch and distribution panel.

3.2.7.4.7 Intentionally Deleted

3.2.7.4.8 Safety Evaluation

In the permanently defueled condition portions of the electrical systems are required for power and/or control of required non-safety related equipment in other systems. Since none of the equipment powered by these systems is safety related (Class 1E), all of the electrical systems are non-safety related. Although single failure criteria still apples to the unit, it need not be applied to systems and equipment that are non-safety related. Since none of the electrical systems or equipment is safety related or required for Regulatory Guide 1.97 (post accident monitoring) commitments, the EEQ program need not be applied. General Design Criteria Number 17 (Electric Power Systems) includes certain requirements for availability of offsite power to support critical functions. Since the reactor cannot be made critical under allowed plant conditions in the permanently defueled condition, no power source is required to be operable or available.

3.2.8 AIR CONDITIONING, HEATING, COOLING AND VENTILATION SYSTEMS

3.2.8.1 Reactor Building and SFPI Heating and Ventilation System

3.2.8.1.1 Design Bases

The Reactor Building and SFPI heating and ventilation systems are operated to maintain a temperature above freezing within the areas of that building.

The systems also maintain a slightly negative pressure when compared to the outside atmosphere. This is performed to ensure that there will be no inadvertent unmonitored release to the site area from the reactor building.

Water vapor from quiescent evaporation of liquid waste may be released into the ventilation system. The process allows only the distillate vapor into the ventilation system, assuring positive control over the species and concentration of radionuclides released with Reactor Building exhaust air.

Ventilating air flow is routed to areas of progressively greater radioactive contamination prior to final exhaust. Back-draft dampers are provided to prevent reverse flow between areas of different contamination potential.

Filtering of supply air is provided to reduce the presence of dust particles.

Reactor Building main supply and exhaust units consist of fan, motor, and their associated controls.

The SFPI system includes supply and exhaust fans installed in modular units.

3.2.8.1.2 System Description

The Reactor Building and SFPI HVAC systems provide for the protection of personnel and equipment from airborne radioactive contaminants and excessive thermal conditions. Air flow is directed to areas of progressively greater radioactive contamination prior to exhaust.

The Reactor Building is provided with supply and exhaust ventilation to ensure proper air flow direction and remove heat generated from equipment.

The SFPI system includes variable speed supply and exhaust fans to maintain space temperature within acceptable limits while also maintaining a negative pressure within the SFPI envelope relative to the outside and to Reactor Building areas outside the SFPI envelope.

A flow diagram of the Reactor Building HVAC system is given in Figure 3.2-12. The SFPI HVAC system is shown in Figure 3.2-6.

Reactor Building HVAC

The supply segment of the system provides fresh air to all levels in the Reactor Building outside the SFPI envelope. Outside air passes through fixed louvers, a damper, filters, and electric heating coils. One fan is available to deliver air flow. Electric unit heaters are provided inside the drywell for freeze protection. Exhaust air flow combines in a common duct and continues on to the main exhaust fan plenum.

System components, in addition to those mentioned above, include screens, filters, ductwork with dampers, supply outlets, return and exhaust intakes, heating coils, and instrumentation and controls. Control actuation, indication, and alarm instrumentation are incorporated in a central HVAC master control panel.

SFPI HVAC System

The supply segment of the system provides fresh air to the operating floor of the Reactor Building, portions of the 82 feet 9 inches elevation and the spent fuel pool pump area. Outside air passes through fixed louvers in the side of the reactor building wall, filters, and electric heating coils. A single variable speed 100% capacity fan is available to deliver air flow.

A single variable speed exhaust fan discharges air from the SFPI envelope through a HEPA filter, and fixed louver in the reactor building wall. The exhaust fan is operated in conjunction with the supply fan to maintain space temperatures within acceptable limits while also maintaining a slight negative pressure in the SFPI envelope relative to the outside and to Reactor Building areas outside the SFPI envelope.

Electric unit heaters are installed in all SFPI areas to maintain acceptable space temperatures.

System components, in addition to those mentioned above, include ductwork with dampers, supply outlets, return and exhaust intakes, and instrumentation and controls. Control actuation, indication, and alarm instrumentation are incorporated in a local control panel. Indication and alarm functions are provided in the Millstone Unit 2 Control Room.

Natural ventilation cooling capability is also provided by opening the Reactor Building truck bay doors, equipment hatch garage doors and the tornado dampers located on the Reactor Building roof. This path would be used following an extended loss of all spent fuel pool cooling capability.

3.2.8.1.3 Safety Evaluation

The Reactor Building and SFPI heating and ventilation systems maintain environmental conditions in building spaces (to support personnel comfort or operation of equipment located on those spaces), direct ventilation air from areas of low radioactive contamination to areas of progressively greater contamination (to minimize the spread of contamination), and vent potentially contaminated exhaust air. Natural ventilation cooling capability is also provided for spent fuel pool cooling following an extended loss of all active pool cooling capability. The Reactor Building and SFPI heating and ventilation systems are not safety related, but are required in the permanently defueled condition because they house SSCs that are associated with the safe storage and handling of irradiated fuel or radioactive waste.

3.2.8.2 Radwaste Building Ventilation System

3.2.8.2.1 Design Bases

The Radwaste Building ventilation system operates to supply filtered air to this building's areas.

Supply air is filtered. The presence of dust particles potentially increases the spread of radioactive contamination.

This system also filters the exhaust air prior to its discharge, to limit the release of any radioactive contaminants to the environment.

Ventilating air flow is routed to areas of progressively greater radioactive contamination potential prior to final exhaust. Back-draft dampers are provided to prevent reverse flow between areas of different contamination potential.

3.2.8.2.2 System Description

Figure 3.2-13 shows the ventilating flow through the Radwaste Building. The ventilating system is designed to provide a passive supply of filtered air and exhaust it after filtration. Air is drawn through the building by the main exhaust fan. An exhaust filter unit is provided.

Outside air is drawn into the system through two inlets above the roof of the building and protected by bird screening. The air is drawn through a filter designed to remove dust. A header conveys fresh air to various areas of the building.

The fresh air supply is located in the clean areas of the building while the inlets to the exhaust ducts are located where the rate of contamination is the highest.

The exhaust air is passed through the filtering system before discharge through the main exhaust fan.

3.2.8.2.3 Safety Evaluation

The Radwaste Building ventilation directs ventilation air from areas of low radioactive contamination to areas of progressively greater contamination (to minimize the spread of contamination), and vents potentially contaminated exhaust air. The Radwaste Building ventilation system is only required, in the permanently defueled condition, to support personnel access to the space.

3.2.8.3 Intentionally Deleted

3.2.8.4 Turbine Building Heating and Ventilation

3.2.8.4.1 Design Bases

The Turbine Building ventilation system is operated to maintain a slight negative pressure in the building to prevent any radioactive out-leakage, as well as, to provide fresh air to support personnel access.

3.2.8.4.2 System Description

Fresh air is supplied to the Turbine Building through louvers in the walls and roof.

The ventilation system is arranged with one supplementary transfer fan and connecting ductwork to induce flow to the north end of elevation 14 feet 6 inches.

The Turbine Building exhaust system collects air from various areas into an exhaust air header then discharges it into a plenum which also receives air from the Reactor Building and Liquid Radwaste Building. One exhaust fan is furnished to handle the combined exhaust from these three buildings. This fan discharges into a duct which runs along the north wall of the Reactor Building before releasing the exhaust air to the environment. Potentially contaminated areas in the Turbine

Building are maintained at a negative pressure by exhausting from these areas. The exhaust air is drawn from adjacent spaces. This arrangement controls the air flow pattern and prevents out leakage.

The Turbine Building ventilation air is normally discharged to the atmosphere without treatment.

A flow diagram of the Turbine Building area ventilation system is shown in Figure 3.2-7.

3.2.8.4.3 Safety Evaluation

The Turbine Building ventilation system directs ventilation air from areas of low radioactive contamination to areas of progressively greater contamination (to minimize the spread of contamination), and vents potentially contaminated exhaust air. The Turbine Building ventilation system is only required, in the permanently defueled condition, to support personnel access to the space.

3.2.9 FIRE PROTECTION SYSTEMS

The licensee's Nuclear Plant Fire Protection Program has been developed to ensure that any single fire will not cause an unacceptable risk to public health and safety, and will not significantly increase the risk of radioactive release to the environment.

A Fire Protection Program has been established at Millstone Unit Number 1. This program establishes the fire protection policy for the protection of structures, systems, and components important to the defueled condition of the unit and the procedures, equipment, and personnel required to implement the program.

3.2.9.1 Design Bases

To achieve and maintain a high level of confidence for the Fire Protection Program, it has been organized and is administered using the defense-in-depth concept. The defense-in-depth concept assures that if any level of fire protection fails, another level is available to provide the required defense. In fire protection terms, this defense-in-depth concept consists of the following levels;

- Preventing fires from starting,
- Early detection of fires that do start, and
- Controlling and/or extinguishing them quickly so as to limit their damage.

None of these levels can be perfect or complete, but strengthening any one level can compensate in some measure for weaknesses, known or unknown, in the others.

3.2.9.2 System Description

3.2.9.2.1 Site Water Supply System

The underground fire protection water supply system consists of a 12 inch cast and ductile iron, cement-lined pipe extending around Millstone Unit Number 1, 2, and 3 in a loop arrangement.

The supply system services individually valved lines feeding fixed pipe water suppression systems (sprinklers, waterspray, and standpipes) throughout the plant and hydrants located around the exterior of the plant.

The Millstone Unit Number 2 and 3 fire pumphouses contain three, 2,000 gpm at 100 psi, fire pumps which supply the yard loops; two with electric-motor drives and one with diesel engine drive. The Millstone Unit Number 3 pumphouse contains one electric driven pump (M7-8), fed from Millstone Unit Number 3 power, and the diesel-driven fire pump (M7-7). The Millstone Unit 2 pumphouse contains one electric driven pump (P-82) fed from Unit 2 power. All three pumps have individual connections to the underground supply system. Maximum system flow and pressure requirements can be met with any one of the three pumps out of service.

System operation is such that a 50 gpm electric jockey pump (M7-11) maintains system pressure by automatically starting when line pressure drops to 105 psig and will run until pressure reaches 120 psig as indicated by a line pressure switch. A hydro-pneumatic tank is provided in the system to prevent short cycling of the jockey pump. At pressures below 105 psig, the MP2 P-82 electric pump first starts at 98 psig to maintain system pressure and flow. The Millstone Unit Number 3 M7-8 electric pump then will start at 85 psig and it is fed 480 VAC from MCC-CD-6 (MCC number 22A-2 Compartment number 1A). This pump is auto-started by a pressure switch set at 85 psig decreasing, while the M7-7 diesel-driven fire pump is auto-started by a separate pressure switch set at 75 psig decreasing. The diesel pump is started by its own self-contained battery system. A battery charger is provided for recharging. Both Millstone Unit Number 3 electric and diesel-driven fire pumps deliver 2000 gpm at 100 psi discharge pressure and remain in operation until they are manually shut down. Electrical interlocks stop the jockey pump when either of the two Millstone Unit Number 3 fire pumps start.

The fire pumps are supplied from two 250,000 gallon ground level tanks. The tanks are automatically filled through a water line fed from city water.

If a major fire in any location of the MP-1 site should occur, the combined water tank and makeup water capacity would provide an adequate water supply for MP-1. The necessary pressure and flow would be maintained through the use of any two simultaneously operating 2,000 gpm rated pumps.

3.2.9.2.2 Fixed Suppression Systems

The fire protection features for the Unit 2 Control Room are discussed in Section 9.10 of the Millstone Unit 2 Final Safety Analysis Report.

(1) Sprinkler and Waterspray Systems

The fixed water suppression systems for the “cold and dark” stage of the decommissioned unit are designed as follows:

- Wet Pipe Automatic Sprinkler System (Maintenance Shop/Central Monitoring Station (CMS) Sprinkler System)
- Automatic Deluge Waterspray System (ESST Deluge System)
- Dry Pipe Manual Sprinkler Systems (Condenser Bay, Turbine Building Truck Unloading Area, and Reactor Building Rail Airlock Sprinkler Systems)

The design concept for the fixed fire water suppression systems will use automatic operating systems for the heated plant area (Maintenance Shop/CMS) and the ESST located outside the east wall of the Maintenance Shop. For the unheated plant areas, a manual actuation concept will be used. The design will be to operate with “dry pipes in the unheated areas (Turbine, Reactor, and Radwaste Buildings) and “flood up” the piping systems to activate the suppression system by opening a single isolation valve in the Maintenance Shop (Valve 1-Fire-37). This valve will be accessible to the plant operators or responding fire department members outside the fire areas being protected by the dry pipes.

The sprinkler systems and deluge waterspray system have been designed using the guidance of the National Fire Protection Association (NFPA) Standard Number 13 for the “Installation of Sprinkler Systems” or NFPA Standard Number 15 for “Waterspray Fixed Systems.” The dry manual operating concept is not in conformance with NFPA but has been determined to be acceptable for the hazards of the decommissioned plant.

(2) Wet Pipe Automatic Operating Sprinkler System

An automatic, closed head, wet pipe design sprinkler system has been provided for the Maintenance Shop/Central Monitoring Station (CMS) area. This system has an alarm check valve which actuates an electric pressure switch to transmit a waterflow signal to the PLC. The system is provided with an outside screw and yoke (OS&Y) isolation valve between the supply connection and the system distribution piping. Sprinkler heads are closed, heat actuated type sprinkler heads.

(3) Automatic Operating Deluge Waterspray Systems

An automatic, open head, deluge type waterspray system has been provided for the Emergency Station Services Transformer (ESST). This system has a deluge valve that actuates upon an input from a heat detection circuit located around the transformer. Upon actuation, an electric alarm switch actuates and transmits an actuation signal to the PLC and water flows into the distribution piping and discharges from all open spray heads. The system has an OS&Y isolation valve located between the supply header and the distribution piping. Automatic operation is initiated by a single zone heat detection circuit installed in the hazard area.

Manual operation of the automatic deluge system is provided via a mechanical pushbutton operator located on the deluge valve in the Maintenance Shop Welding area.

(4) Dry Pipe Manual Sprinkler Systems

Three sprinkler systems are provided in the unheated portion of the facility. These systems protect the Condenser Bay, the Turbine Building Truck Unloading Area, and the Reactor Building Rail Airlock. Sprinkler systems in the unheated portion of the plant are operated as dry pipe manual sprinkler systems. Each system has an isolation valve that separates the system from the supply header. The systems have closed fusible type sprinkler heads. There is no waterflow alarm provided. System piping has been arranged to facilitate complete draining during cold weather conditions. These systems would be charged with water by manually opening isolation valve 1-Fire-37 located in the Maintenance Shop Welding Area as part of a fire fighting strategy for the facility.

3.2.9.2.3 Portable Suppression Capabilities

(1) Hose Stream Coverage

Hose stream coverage is available to all fire areas of the plant from stand pipe connections to fixed 1.5 inch hose stations or by use of 2.5 inch diameter hose with gated wye connections available from outside hose houses.

The hose stations in the Maintenance Shop/CMS area are fed by the “wet” header piping and are available for immediate fire suppression use. The hose stations in the Turbine Building, Reactor Building, and Liquid Radwaste Building are fed off of the “dry” fire water header and will be available for fire fighting following the flood-up of the header following the opening of valve 1-Fire-37 in the Maintenance Shop. Hose stations in the Solid Radwaste Building are fed directly off a connection to the yard fire main and are maintained wet with heat tracing on the piping and valves to prevent freezing in this unheated area.

Hose station locations are shown in the FHA (Reference 3.2-19).

(2) Portable Extinguishers

Selection and placement of portable fire extinguishers are in accordance with the intent of the guidelines of NFPA Standard Number 10, “Standard for Portable Fire Extinguishers”. All extinguishers utilized are Underwriters Laboratories (UL) listed.

3.2.9.2.4 Fire Detection and Alarm Systems

The fire detection and alarm systems installed in the plant are designed in general compliance with NFPA Standard Number 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protective Signaling Systems," and with NFPA Standard Number 72, "National Fire Alarm Code."

Fire detection systems are used for early warning detection and in some cases may have the capability to actuate fixed fire suppression systems.

Detection devices consist of fixed temperature detectors and smoke detectors. Smoke detectors are of the spot type, employing the ionization principle. Specific application of these detectors in each fire area is detailed in the FHA (Reference 3.2-19).

In general, the installation of detector units is in accordance with the intent of the guidelines set forth in NFPA Standard Number 72E, "Standard on Automatic Fire Detectors".

Fire/smoke detectors, as with waterflow indicators, and valve tamper devices are arranged to transmit signals to local alarm panels and a fixed suppression system control panel, if applicable. Actuation signals are also transmitted through the local alarm panels to control panels in the Central Monitoring Station (CMS). A Fire Alarm panel located in the CMS monitors those areas necessary to support the Spent Fuel Pool Island. Trouble signals for these devices are transmitted in a similar manner. A general alarm is provided in the Unit 2 Control Room. Identification of the exact alarm or trouble signals must be performed locally in the Unit 1 CMS.

The alarm system also monitors other miscellaneous fire protection system features.

3.2.9.2.5 Ventilation Systems and Smoke Removal

Removal of the products of combustion from any specific plant area requires the use of the normal plant ventilation system, which is designed to handle the expected normal environment within a given area or the use of portable exhaust fans by the fire brigade. There are no cable tunnels, culverts, or other unventilated areas that pose any special venting problems. Removal of gaseous radioactive waste either from plant processes or airborne particulates requires the use of charcoal filters.

The ventilation and filtration systems of potential radiation release areas are discussed in detail for the Reactor, Turbine, Radwaste, Radwaste storage, and Screenhouse Buildings in the FHA, Reference 3.2-19.

3.2.9.3 Safety Evaluation and Fire Hazards Analysis

3.2.9.3.1 Evaluation Criteria

An evaluation of the overall Fire Protection Program as indicated by the FHA, (Reference 3.2-19), found that the program does provide reasonable assurance that a fire will not cause an unacceptable risk to the public health and safety. The fire protection program accomplishes this by assuring a fire will not significantly increase the risk of radioactive release to the environment. Therefore, the Fire Protection Program meets the basic requirements of General Design Criteria 3 and 5 as applicable to a permanently defueled facility. Branch Technical Position (BTP) APCSB 9.5.1, “Guidelines for Fire Protection for Nuclear Power Plants," provides the implementing criteria for GDC 3 and gives the general guidelines used to review Millstone Unit Number 1. BTP APCSB 9.5.1 provides the guidelines acceptable to the NRC staff for implementing the following criteria:

a. General Design Criterion 3 (10 CFR 50, Appendix A) - Fire Protection.

b. Defense-in-Depth Criterion: For each fire hazard, a suitable combination of fire prevention, fire detection and suppression capability, and ability to withstand safely the effects of a fire is provided. Both equipment and procedural aspects of each are considered.

c. Single-Failure Criterion: No single active failure shall result in complete loss of protection of both the primary (fix installed systems) and backup fire suppression capability (standpipe/extinguishers).

d. Fire Suppression System Capacity and Capability: Fire suppression capability is provided, with capacity adequate to extinguish any fire that can credibly occur and have adverse effects on equipment and components important to safety.

e. Backup Fire Suppression Capability: Total reliance for fire protection is not placed on a single automatic fire suppression system. Appropriate backup fire suppression capability is provided in the form of portable fire extinguishers or hose stations.

In addition to the specific guidance of the BTP, the evaluation considered the adequacy of the Fire Protection Program on the effects of potential fire hazards throughout the plant based on sound fire protection engineering practices and judgments.

3.2.9.3.2 Fire Hazard Analysis Methodology

Fire Protection was evaluated by conducting a fire hazard analysis of individual fire areas and fire zones within the plant. The analysis methodology is described in the Fire Hazards Analysis (Reference 3.2-19).

3.2.9.3.3 Fire Hazard Analyses Results

The fire hazards analysis results for each fire area are contained in the FHA (Reference 3.2-19).

3.2.9.4 Inspection and Testing

Administrative controls are provided through existing Plant Administrative Procedures, Operating Procedures and the Quality Assurance Program to ensure that the Fire Protection Program and equipment is properly maintained. This includes QA audits of the program implementation, conduct of periodic test inspections, and remedial actions for systems and barriers out of service.

The technical requirements found in Millstone Unit Number 1 Technical Requirements Manual describe the limiting condition for operation and surveillance requirements for the fire protection system. These technical requirements ensure the fire protection system is properly maintained and operated.

All fire protection equipment and systems are subject to periodic inspections and tests in accordance with the intent of National Fire Codes and the Fire Protection Program.

The following fire protection features will be subjected to periodic tests and inspections:

(1) Fire alarm and detection systems

(2) Wet pipe automatic sprinkler systems

(3) Water spray systems

(4) Interior fire water supply headers

(5) Fire pumps

(6) Fire barriers (walls, fire doors, penetration seals, fire dampers)

(7) Manual suppression (fire hoses, hydrants, extinguishers)

Equipment out of service including fire suppression, detection, and barriers will be controlled through the administrative program and appropriate remedial actions taken. The program requires all impairments to fire protection systems to be identified and appropriate notification given to the Site Fire Marshal for evaluation.

As conditions warrant, remedial actions would include compensatory measures to ensure an adequate level of fire protection in addition to timely efforts to effect repairs and restore equipment to service.

3.2.9.5 Personnel Qualification and Testing

3.2.9.5.1 Fire Protection Organization

The officer responsible for the Fire Protection Program at Millstone Unit Number 1 is defined in the QAP. Formulation, and assessment of the effectiveness of the program are delegated as indicated in Reference 3.2-20, the Fire Protection Program Manual.

3.2.9.5.2 Fire Brigade and Training

The Site Fire Brigade and Nuclear Training are a site (Units 1, 2, and 3) organizations. The Millstone Site Fire Brigade consists of a minimum of a Shift Leader and four Fire Brigade personnel. MP-2 supplies an advisor, who is at a minimum a fully qualified Unit 1 Plant Equipment Operator, to the Fire Brigade Shift Leader. The advisor will provide direction and support concerning plant operations and priorities.

Members of the Fire Brigade are trained by the Nuclear Training Department.

Site Fire Brigade personnel are responsible for responding to all fires, fire alarms, and fire drills. To ensure availability, a minimum of a Shift Leader and four Fire Brigade personnel remain in the owner controlled area and do not engage in any activity which would require a relief in order to respond to a fire (e.g., continuous fire watch).

If assistance is needed to fight a fire, additional equipment and manpower is supplied by the off site local fire departments. Within a 5 mile radius of the plant there are numerous local volunteer fire companies. Letters of commitment to supply public fire department assistance have been obtained from these fire companies.

The Shift Leader coordinates the Site Fire Brigade activities, and ensures proper communications and coordination of support for the local fire department chief or officer in charge once on site, and other on site activities (e.g., Chemistry, Health Physics, and Security).

Nuclear Training coordinates with the Site Fire Marshal and periodically familiarizes local fire department personnel with the Station’s layout and fire fighting equipment. The Site Fire Marshal coordinates with the Site Fire Brigade Personnel and all Unit Shift Managers, informing them of the status of the site fire protection equipment, should equipment become inoperable or unavailable.

Fire Protection drills are planned and critiqued by Nuclear Training and members of the management staff responsible for plant fire protection. Performance deficiencies of the Fire Brigade or of individual Fire Brigade personnel are remedied by scheduling additional training for the Site Fire Brigade or individuals.

3.2.9.5.3 Quality Assurance

The QA Program has been applied via the Fire Protection Program Manual to the FPSs which provide a function for the operating units.

3.2.10 REFERENCES

3.2-1 Docket Number 50-245, LS05-82-03-060, J. Shea to W.G. Counsil, 'SEP Topic IX-1, Fuel Storage (Millstone 1)," March 9, 1982.

3.2-2 Docket Number 50-245, B10301, W.G. Counsil to D.M. Crutchfield, 'Millstone Nuclear Power Station, Unit Number 1, SEP Topic IX-1, Fuel Storage,' August 31, 1981.

3.2-3 Docket Number 50-245, B10346, W.G. Counsil to D.M. Crutchfield, 'Millstone Nuclear Power Station, Unit Number 1, SEP Topic IX-1, Fuel Storage,' December 14, 1981.

3.2-4 Docket Number 50-245, B12961, M.L. Boyle to E.J. Mroczka, 'Millstone Nuclear Power Station, Unit Number 1, Issuance of Amendment Number 40 (TAC No. 68157),' November 27, 1989.

3.2-5 Docket Number 50-245, A08680, M.L. Boyle to E.J. Mroczka, 'Millstone Nuclear Power Station, Unit Number 1, Issuance of Amendment Number 43 (TAC No. 72183)," March 30, 1990.

3.2-6 Docket Number 50-245, J.W. Andersen to J.F. Opeka, 'Millstone Nuclear Power Station, Unit Number 1, Issuance of Amendment Number 89 (TAC No. M93080)," November 9, 1995.

3.2-7 Reference deleted.

3.2-8 J.A. Price (Dominion) letter to U.S. NRC, “Millstone Power Station, Unit Number 1, Docket Number 50-245, Fuel Storage Requirements, Technical Specification 4.2”, Letter Number B18972, dated Sept. 18, 2003.

3.2-9 Holtec Report Number H1-971914, Revision 1, “Analysis Of 1675 Pound Fuel Assembly System Drop Onto The Irradiated Fuel Assembly.”

3.2-10 Holtec Report Number AH1-971691, Revision 0, “Criticality Safety Analysis Of The MP1 Racks With A Dropped Fuel Assembly.”

3.2-11 Holtec Report Number H1-971698, Revision 0, “Flow And Temperature Field Analysis Of Localized Cell Blockage In The Millstone Unit Number 1 Spent Fuel Pool.”

3.2-12 Holtec Report Number H1-971675, Revision 1, “Analysis Of Tetrabor And Boraflex Racks Under 1675 Pound Fuel Assembly System Impact.”

3.2-13 Holtec Report Number H1-98210, Revision 1, “Decay Heat Load Calculation for the Millstone Unit 1 Spent Fuel Pool.”

3.2-14 Holtec Report Number H1-992125, Revision 0, “Steady State Temperature of Millstone Unit 1 SFP and RB with No Active SFP Cooling.”

3.2-15 Docket Number 50-245, B10291, W.G. Counsil to D.M. Crutchfield, “Millstone Nuclear Power Station, Unit Number 1, SEP Topic IX-3, Station Service and Cooling Water Systems,” November 24, 1981.

3.2-16 Docket Number 50-245, NUREG-0824, Integrated Plant Safety Assessment, Systematic Evaluation Program, Millstone Nuclear Power Station, Unit Number 1, February 1983, Topic IX-3, "Station Service and Cooling Water Systems.”

3.2-17 Docket Number 50-245, B10292, W.G. Counsil to D.M. Crutchfield “Millstone Nuclear Power Station, Unit Number 1, SEP Topic IX-5, Ventilation Systems”, November 19, 1981.

3.2-18 Docket Number 50-245, LS05-82-09-043, J. Shea to W.G. Counsil, SEP Topic IX-5, Ventilation Systems, Millstone Nuclear Power Station, Unit Number 1, September 14, 1982.

3.2-19 Fire Hazard Analysis Millstone Unit Number 1, Revision 6, July 2000.

3.2-20 Millstone Nuclear Power Station Fire Protection Program Manual.

CHAPTER 4 – RADIOACTIVE WASTE MANAGEMENT

4.1 SOURCE TERMS

With the permanent defueled condition of Unit 1, fission, corrosion, and activation products from
operation are no longer produced. The radioactive inventory that remains is primarily attributable
to activated reactor components and structural materials and residual radioactivity. The
accumulation of small amounts of solid waste as evaporator bottoms or contaminated materials
may easily be controlled. Planned liquid effluent releases will be evaluated prior to release, and 11-1
appropriate controls (e.g., monitoring) will be established. The Radiological Effluent Monitoring
and Offsite Dose Calculation Manual ensures that Unit 1 complies with 10 CFR 50, Appendix I.

4.2 RADIATION PROTECTION DESIGN FEATURES

4.2.1 FACILITY DESIGN FEATURES

Radiation shielding was provided to restrict radiation emanating from various sources throughout the plant. The primary objective of radiation shielding is to minimize the radiation exposure of plant personnel and the general public.

Millstone Unit Number 1 is permanently shutdown and many installed components which are provided with shielding, are no longer required to safely store irradiated fuel. However, many of these installed components continue to contain radioactive material or remain radioactive themselves. Shielding that was originally designed to shield these components while they supported reactor operation, continues to provide shielding from the residual activity in the permanently shutdown condition.

With the vessel in a drained down condition, a concrete shielding package is installed over the reactor vessel head and reactor cavity floor to provide shielding from activated reactor vessel internals.

4.2.1.1 Design Basis

Normal operating conditions determined the major portion of the original plant shielding design requirements. Two exceptions to this were the Control Room where shielding was determined by radiation levels produced during the loss-of-coolant accident and the shutdown cooling system where shielding was determined by shutdown conditions. Although these conditions are no longer applicable, these were the bases for the unit shielding.

4.2.1.2 Ventilation

Information on ventilation systems is contained in Chapter 3.

4.2.2 RADIATION PROTECTION PROGRAM

4.2.2.1 Organization

The radiation protection program is established to provide an effective means of radiation protection for permanent and temporary employees and for visitors at the station. The radiation protection program is developed and implemented through the applicable guidance of Regulatory Guides 8.2, Revision 0; 8.8, Revision 3; and 8.10 Revision 1.

The radiation protection department and line function management implement and enforce the radiation protection program.

The officer responsible for implementing the radiation protection program is defined in the QAP.

The radiation protection manager meets or exceeds the qualifications for radiation protection manager in Regulatory Guide 1.8, Revision 1. Radiation protection technicians meet or exceed the qualifications specified in ANSI N18.1-1971.

4.3 ALARA PROGRAM

4.3.1 POLICY CONSIDERATIONS

It is the policy of the licensee to maintain individual and plant personnel total radiation exposure ALARA. The licensee's ALARA policy complies with 10 CFR 20 and 10 CFR 50.

4.3.1.1 Design Considerations

The basic objective of facility radiation shielding is to reduce external dose to plant personnel in conjunction with a program of radiologically controlled personnel access and occupancy in radiation areas to levels which are both ALARA and within the regulations defined in 10 CFR 20. With the reactor shutdown and all fuel stored in the spent fuel pool, the number and magnitude of potential radiation sources have been reduced substantially from the original bases for the radiation protection design features.

4.3.1.2 Operational Considerations

Radiation surveys have been performed and will continue to be performed to ensure that plant areas are correctly posted and barricaded.

4.4 LIQUID WASTE MANAGEMENT SYSTEMS

Liquid waste from the Unit 1 Reactor Building Floor Drain (RBFD) System is collected in two (2) 11-1
active RBFD sumps. There are three (3) active sumps in the Unit 1 Radwaste Building that pump to the “A” RBFD sump. In addition, water collected in the Unit 1 Turbine Building sumps, Unit 1 Ventilation Exhaust Duct (abandoned), Site Stack sump, and similar miscellaneous waste water received from Units 2 and 3 are collected in the RBFD sumps.

The primary method for disposing of this waste water will be using the Waste Water Processing System (WWPS) located in the Unit 1 Reactor Building. The Waste Water Processing System consists of four (4) 10,000 gallon Sample Tanks, recirculation pump, demineralizer, filters and associated piping. The “A” RBFD sump will pump to the WWPS sample tanks, where the water will be batch recirculated and sampled before subsequent discharge. Radiological monitoring will be conducted using an in-line Liquid Effluent Monitor (RE-MG-110). Prior to discharge through DSN-001A (Emergency Service Water discharge piping to discharging canal), dilution flow requirements will be established by crediting Unit 2 Circulating Water Flow to the common discharge canal. In the future, the WWPS will be used to process, sample and discharge Unit 1 Spent Fuel Pool water after all spent fuel assemblies are removed from the Spent Fuel Pool.

The alternative method for processing waste water will be using an eight (8) gallon per hour atmospheric evaporator. Waste water collected in the “A” RBFD sumps will be pumped to a staging tank. Collected liquids may be surveyed for activity and pumped to the evaporator. The distillate vapor will be diluted in the Balance of Plant (BOP) Reactor Building Exhaust flow and released as a ground level release. Radiological monitoring will be conducted by a particulate monitor in the BOP ventilation exhaust or by screening a grab sample of the process liquid. Concentrates in the bottom of the atmospheric evaporator will be collected as required, and disposed as Low Specific Activity (LSA) trash.

If neither of these liquid process methods are available, the RBFD sumps can be pumped to containers which would permit the collected liquids to be processed at a later date, or sent offsite for processing.

4.5 SOLID WASTE MANAGEMENT

The plant has no capability for processing concentrated waste solutions to a solidified product.

Dry Activated Waste (DAW) is processed and stored in appropriate containers to allow for offsite shipment.

Interim on site storage facilities accept waste from Millstone Units 1, 2 and 3. Information regarding facility design criteria is presented in Section 11.4 of the Millstone Unit 3 Final Safety Analysis Report.

4.5.1 DESIGN BASES

The design basis objective of solid waste management is to provide for processing, packaging and handling solid dry wastes, and to allow for radioactive decay and/or temporary storage prior to shipment off site and subsequent disposal.

Solid radwaste handling at Millstone Unit 1 ensures compliance with the following regulations and Regulatory Guides:

(1) 10 CFR 20, Standards for Protection Against Radiation

(2) 10 CFR 50, Appendix I

(3) 10 CFR 61.55, Classification of Waste for Near Surface Disposal

(4) 10 CF 61.56, Waste Characteristics

(5) 10 CFR 71, Quality Assurance Criteria for Shipping Packages of Radioactive Material

(6) Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures and Components

(7) Regulatory Guide 8.8, ALARA Provisions

4.5.2 SYSTEM DESCRIPTION

The solid waste management process is designed to accommodate the following radioactive wastes, which are typical for BWR power plants:

Dry active wastes, which consist of contaminated clothing, tools and small pieces of equipment that cannot be economically decontaminated; miscellaneous paper, rags, etc., from contaminated areas; air filters from radioactive ventilation systems; used reactor equipment such as control rod blades, temporary control curtains, fuel channels and in-core ion chambers - Radioactivity levels of most DAW are low enough to permit handling by contact, it is processed and stored in appropriate containers to allow for off site shipment. Used radioactive equipment may be stored for sufficient time to permit decay before removal for interim storage or off site shipment. Equipment too large to be handled as described above is handled on a case-by-case basis with special procedures. Concentrated bottoms from the waste evaporator may be shipped as low specific activity (LSA) dry active waste.

Summaries of solid waste shipments, types, volumes, and radionuclide composition are given in Reference 4.5-1.

4.5.3 REFERENCES

4.5-1 Millstone Nuclear Power Station Unit Number 1, Docket Number 50-245, Annual Radioactive Effluents Report.

4.6 EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING

4.6.1 DESIGN

4.6.1.1 Design Basis

The effluent radiation monitoring system (RMS) provides nonsafety related functions. The system provides the means for compliance with Nuclear Regulatory Commission (NRC) regulations 10 CFR 20, 10 CFR 50 Appendix A General Design Criteria (GDC) 60, 63 and 64, 10 CFR 50, Appendix I and Regulatory Guides (RG) 1.21, 4.15 and 8.8.

4.6.1.2 System Design Description

SFPI Ventilation Exhaust Monitor

The SFPI ventilation exhaust radiation monitor is designed with the capability to monitor, indicate and record the discharge of gaseous radioactivity. Capability for sampling of particulate activity is provided. Annunciation in the Millstone Unit 2 Control Room occurs if setpoints are exceeded.

Although the monitor cannot determine the individual activity level of the radionuclides in the effluent gas, it provides the overall level and a basis for correlation with laboratory analyses of filter and grab sample activities.

The SFPI gas sample is taken from the exhaust duct near the reactor building exhaust plenum. A single point sample nozzle is positioned to obtain a representative sample of the turbulent and well mixed exhaust air. The monitor is located in a heated enclosure on the 65 foot elevation of the Reactor Building directly below the exhaust duct. The sample passes through a particulate filter and a shielded detection chamber (fixed volume) and is then expelled back into the exhaust duct. The particulate filters are periodically removed for detailed radiological quantitative analysis.

The detector readout is sent to the PLC for display and recording. The range of indication is 1×10^{-6} µci/cc to 1×10^{0} µci/cc (Kr-85).

BOP Ventilation Exhaust Monitor

The Unit Number 1 BOP ventilation exhaust flow is continuously sampled for radioactive particulates. The sample is taken from the exhaust duct which runs along the north exterior wall of the Reactor Building. A single point sample nozzle is positioned to obtain a representative sample of well mixed exhaust air. The particulate sample skid is located in an insulated enclosure on the 65 foot elevation, north wall, of the Reactor Building. The sample passes through a particulate filter and is then expelled back into the exhaust duct. The particulate filter is periodically removed for detailed radiological quantitative analysis.

4.6.2 AREA RADIATION MONITORING INSTRUMENTATION

4.6.2.1 Design Bases

The purpose of the ARM system is to warn of abnormal radiation levels in the SFPI where radioactive material may be present, stored, handled, or inadvertently introduced. The system also provides information concerning radiation at selected locations in the SFPI.

4.6.2.2 System Description

The area radiation monitoring system detects, measures, and indicates ambient gamma radiation dose rates at selected locations in the SFPI. It provides audible and visual alarms in the Millstone Unit 2 Control Room (locally at some locations) when radiation levels exceed pre-selected values or when a monitor has operational failure. Table 4.6-2 lists the area radiation monitor locations and ranges.

<u>Refueling Floor Area Radiation Monitor</u>

The refueling floor ARM is a 3 channel digital unit. Each detector is a gama sensitive GM tube located as described in Table 4.6-2. Each channel is provided with a failsafe High, Warn and Failure alarm relay as well as an analog output. The alarms and analog output are sent to the PLC for recording and alarm. Each unit has a built in check source and local audible and visual alarm indication.

4.6.3 REFERENCE

4.6-1 Letter from W.G. Counsil to D.G. Eisenhut dated July 1, 1981, “Haddam Neck Plant, Millstone Nuclear Power Station, Unit Numbers 1 and 2, Post TMI Requirements - Response to NUREG-0737,” Docket Numbers 50-213, 50-245, 50-336.

TABLE 4.6–1 EFFLUENT RADIATION MONITORS

Monitor	Detector	Range	Trip Function
SFPI ventilation exhaust	(1) Beta Sinctillator	10^{-6} to 10^{0} µci/cc	None

TABLE 4.6–2 AREA RADIATION MONITORING SYSTEM SENSOR AND CONVERTER LOCATIONS FOR MILLSTONE UNIT NO. 1

REACTOR BUILDING		
Station Number	**SENSOR AND CONVERTER LOCATION**	**Range mR/hr**
RM-SFPI-01 CH1	West Refuel Floor	$0.01\text{-}10^2$
RM-SFPI-01 CH2	East Refuel Floor	$0.01\text{-}10^2$
RM-SFPI-01-CH3	West Refuel Floor Hi Range	$10.0\text{-}10^6$

CHAPTER 5 – ACCIDENT ANALYSIS

5.1 INTRODUCTION

In July of 1998, the licensee certified to the NRC that Millstone Unit Number 1 had both permanently ceased operations and that all fuel had been removed from the reactor vessel and placed in the spent fuel pool (Reference 5.1-1). Since Millstone Unit Number 1 will never again enter any operational mode, reactor related accidents are no longer a possibility.

The remaining analyzed accident that is in this chapter is the fuel handling accident. Conservatism in equipment design, conformance to high standards of material and construction, the control of mechanical and pressure loads, and strict administrative control over plant operations all serve to assure the integrity of the fuel in the spent fuel pool.

New hazards, new initiators, and new accidents that may challenge offsite guideline exposures, may be introduced as a result of certain decommissioning activities. These issues will be evaluated when the scope and type of the decommissioning activities are finalized.

5.1.1 ACCIDENT EVENT EVALUATION

5.1.1.1 Unacceptable Results for Design Basis Accidents (DBAs)

The following are considered to be unacceptable safety results for DBAs:

(1) Radioactive material release that results in dose levels that exceed the guideline values of 10 CFR 100.

(2) Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

(3) Radiation exposure to plant operations personnel in the Millstone Unit 2 Control Room in excess of 5 REM whole body, 30 REM inhalation, and 75 REM skin.

5.1.1.2 Fuel Handling Accident Assumptions

Fuel handling accident analysis assumptions are listed on Table 5.2-1.

5.1.1.3 Results

The results of the Fuel Handling Accident analytical evaluation are provided in Section 5.2.

5.1.1.4 Radiological Consequences

Consequences of radioactivity release during a fuel handling accident are presented in Section 5.2.

5.1.2 REFERENCES

5.1-1 "Millstone Nuclear Power Station, Unit Number 1 Certification of Permanent Cessation of Power Operations and that Fuel has been Permanently Removed from the Reactor," July 21, 1998.

5.2 FUEL HANDLING ACCIDENT

As the bounding accident analysis, an inadvertent release of radioactivity, as a result of a fuel handling accident in the spent fuel pool, was evaluated and is discussed below.

With the permanent cessation of operations of Millstone Unit Number 1, the prior limiting fuel handling accident, i.e., a fuel assembly drop onto the top of the core during fuel-handling operations, was no longer part of the plant's design and licensing basis. Several fuel handling accident scenarios are still possible in the spent fuel pool. These scenarios are identified later in this Section.

he radiological consequences of a fuel handling accident in the spent fuel pool are described in this section. For conservatism, a bounding analysis was made to calculate the radiological release from a failure of all fuel rods in four (4) fuel assemblies in the spent fuel pool. Other assumptions taken into consideration are described later in this Section. The off site radiological consequences of this release, i.e., from 4 failed fuel assemblies or, for example, 248 fuel rods for 8x8 fuel assemblies, are substantially less than the 10 CFR Part 100 limits and are tabulated in this section.

5.2.1 FUEL HANDLING ACCIDENT SCENARIOS IN THE SPENT FUEL POOL

The consequences of the following postulated fuel handling drop events were evaluated:

- Spent fuel pool gate (1200 lbs.) drop onto irradiated fuel and fuel storage racks in the spent fuel pool.
- New fuel assembly drop (600 lbs.) onto irradiated fuel and fuel storage racks in the spent fuel pool.
- Lifting of a Tri-Nuc Filter skid (965 lbs.) into the spent fuel pool and potential drop onto irradiated fuel and fuel storage racks.
- Postulated drop of items (pumps, boxes, filters, stellite containers and tables) temporarily stored on the spent fuel pool equipment rail onto irradiated fuel and fuel storage racks.
- Drop of an irradiated fuel assembly onto other irradiated fuel in the spent fuel pool.

These analyses utilized two sophisticated elasto-plastic finite-element models. The first represents the fuel assembly components, the second represents the rack with its pedestals, liner and underlying reinforced concrete structure. The LS-DYNA3D computer code (Reference 5.2-1) was used. Conservative assumptions and restrictive inputs were utilized to result in an upper bound estimate of the calculated damage for the postulated drop event.

The following assumptions were utilized in the analysis:

Regarding the impactor movement and the target:

- Both the impactor and the target are submerged.
- The target is in a stationary position prior to impact.
- The trajectory of the impactor is vertical.
- The form drag force opposed to the impactor movement is proportional to it’s velocity squared.
- The friction drag force is conservatively neglected.

Regarding the impact mechanism transmission:

- The impactor makes first contact with the fuel assembly handle which is located above the rack elevation. Furthermore, the handle is conservatively considered as a prefect rigid body, without deformability or energy absorption capacity.

Regarding failure criteria:

- Failure of an individual fuel rod is assumed to occur when the irradiated zircaloy material reaches its postulated failure stress (strain). For additional conservatism, the entire length of each fuel rod is assumed irradiated to the state where the brittle material behavior is active.
- Overstress of the lower guide ends (between the lower end of the fuel rod and the bottom fitting) is not considered as a failure of the supported rod.

The analysis of these additional accident scenarios has determined that the limiting event is the drop of the spent fuel pool gate, which can result in extensive damage of the fuel assemblies, showing a total of 54 ruptured fuel rods. The drop of the new fuel assembly resulted in damage to the targeted fuel assemblies, but no ruptured fuel rods were recorded for either the impactor or the target. Drop of an irradiated fuel assembly results in failure of all 64 guide ends, but no rupture of fuel rods occurs. These results bounded all fuel types stored within the Millstone Unit Number 1 spent fuel pool for the analyses performed to date.

5.2.2 RADIOLOGICAL CONSEQUENCES

Since the licensee has certified to the NRC that there is a permanent cessation of operations of Millstone Unit Number 1 and that fuel has been permanently removed from the reactor vessel, a calculation evaluating the radiological consequences of a fuel handling accident in the spent fuel pool was performed and eventually chosen as the new bounding accident (Reference 5.2-2). Taking into account the actual source term of the fuel in the spent fuel pool (i.e., appropriate decay time of fuel), the reanalysis assumed four fuel assemblies (e.g., 248 rods in an 8x8 assembly) failed in the spent fuel pool and resulted in an unfiltered, i.e., no Standby Gas

Treatment System (SGTS) available and secondary containment not set, puff release. Additional assumptions and input parameters are given in Table 5.2-1. This reanalysis was performed using the guidelines of Standard Review Plan 15.7.4 Rev. 1 and Regulatory Guide 1.25. Doses were calculated using the TACT-III, ORIGEN-2, and ELISA computer codes.

The results of this dose assessment for 4 failed fuel assemblies revealed the following radiological dose data:

Thyroid dose at the exclusion area boundary 5.44E-04 REM

Thyroid dose at the low-population zone 1.69E-05 REM

Whole-body dose (calculated as TEDE) at the exclusion area boundary 1.03E-03 REM

Whole-body dose (calculated as TEDE) at the low-population zone 3.20E-05 REM

These doses are well within the limits of 10 CFR 100, and are therefore acceptable.

Doses were also calculated to the Millstone Unit Number 2 Control Room. The results of this dose assessment is as follows:

Thyroid dose to the Millstone Unit Number 2 Control Room	7.65E-02 REM
Whole-body dose to (calculated as TEDE) the Millstone Unit Number 2 Control Room	8.67E-02 REM
Beta skin dose to the Millstone Unit Number 2 Control Room	2.19E+01 REM

These doses are less than the limits specified in GDC 19. Doses were not calculated for the Millstone Unit Number 3 control room since the atmospheric dispersion factor (χ/Q) is approximately 50 times less that the (χ/Q) to the Millstone Unit Number 2 control room.

Therefore, the dose to the Millstone Unit Number 3 control room would be approximately 50 times less than the Millstone Unit Number 2 control room dose.

5.2.3 REFERENCES

5.2-1 LS-DYNA3D, Version 932, Livermore Software Technology Corporation, May 1, 1995.

5.2-2 Calculation Package NUC-197, "MP1 Defueled State - Radiological Analysis of a Fuel Handling Accident," Duke Engineering and Services, October 11, 1999.

TABLE 5.2–1 ASSUMPTIONS AND INPUT CONDITIONS FOR FUEL HANDLING ACCIDENT AT MILLSTONE UNIT NO. 1

Assumption	Basis
1. Core Power Level During Irradiation = 2011 MWt	Technical Specifications
2. Varied to identify conservative results based on actual burnup.	Regulatory Guide 1.25 See Ref. 5.2-3.
3. Varied to identify conservative results based on actual burnup	Regulatory Guide 1.25 See Ref. 5.2-3.
4. Pool Scrubbing Factor = 60	Extrapolation of Regulatory Guide 1.25 DF to MP1 conditions. See Ref. 5.2-3.
5. Chemical form of Iodine above pool:	Regulatory Guide 1.25 See Ref. 5.2-3.
• 85 percent Elemental	
• 15 percent Organic	
6. Number of Assemblies in Core: 580	Technical Specifications
7. For radiological dose assessment: Number of fuel assemblies assumed to fail = 4	DSAR Section 5.2.2
8. Release fractions from fuel rods:	Regulatory Guide 1.25 & conservative assumption
• 30 percent Noble Gases	
• 12 percent Iodines	
9. No credit taken for secondary containment	Technical Specifications
• SGTS not in operation	
• Puff release is an unfiltered ground release	
10. Breathing rate = 3.47×10^{-4} m^3/sec	Regulatory Guide 1.25
11. Ground level dispersion factor (χ/Q): EAB (0-2 hr.) = 6.10×10^{-4} sec/m^3 LPZ (0-4 hr.) = 1.90×10^{-5} sec/m^3	SEP Topic 11-2.c, Docket Number 50-245
12. Decay Time for fuel = 3.8 years	Based on the MP1 shutdown on November 4, 1995.

CHAPTER 6 – CONDUCT OF OPERATIONS

6.1 ORGANIZATIONAL STRUCTURE

Information regarding the organizational structure is presented in Section 1.0 of the Quality Assurance Program Description Topical Report (Reference 6.1-1). With the exception given below, that information is incorporated herein by reference.

The owner, holding 100 percent of the Millstone Unit Number 1 nuclear plant, is Dominion Nuclear Connecticut, Inc..

6.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

Information regarding the management and technical support organization is presented in Section 1.0 of Reference 6.1-1. That information is incorporated herein by reference.

6.1.1.1 Technical Support for Operations

Information regarding the technical support for operations is presented in Section 1.0 of Reference 6.1-1. That information is incorporated herein by reference.

6.1.1.2 Organizational Arrangement

Information regarding the organizational arrangement is presented in Section 1.0 of Reference 6.1-1. That information is incorporated herein by reference.

6.1.2 OPERATING ORGANIZATION

6.1.2.1 Plant Organization

The plant organization is as shown in Reference 6.1-1.

6.1.2.2 Plant Personnel Responsibilities and Authorities

Information regarding the plant personnel responsibilities and authorities is presented in Section 1.0 of Reference 6.1-1. That information is incorporated herein by reference.

6.1.2.3 Operating Shift Crews

The minimum shift crew composition is contained in the Administrative Controls section of the Millstone Unit Number 1 Technical Specifications.

6.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

6.1.3.1 Qualification Requirements

Qualifications of plant managerial and supervisory personnel are established by the American National Standards Institute (ANSI) N18.1 (Reference 6.1-2) except for the following:

a. The Operations Manager or Assistant Operations Manager shall be a Certified Fuel Handler.

b. The Radiation Protection Manager shall meet or exceed the qualifications of Regulatory Guide 1.8, Rev. 1.

6.1.4 REFERENCES

6.1-1 Quality Assurance Program Description Topical Report.

6.1-2 American National Standards Institute, ANSI N 18.1-1971, Selection and Training of Nuclear Power Plant Personnel.

6.2 TECHNICAL SPECIFICATIONS

Technical Specifications set forth the limits, operating conditions and other requirements for the protection of the health and safety of the public. These specifications have been written in fulfillment of 10 CFR 50.36 and are controlled pursuant to 10 CFR 50.90, 50.91, and 50.92. Technical Specifications are maintained as Appendix A to the operating license.

The Technical Requirements Manual (TRM) contains clarifications for certain technical specifications and a central location for other documents which place operating limits on the plant. Changes to the TRM are controlled pursuant to the 10 CFR 50.59 process.

6.3 PROGRAMS

6.3.1 TRAINING

Programs are credited to train plant personnel. Key technical operating personnel receive onsite classroom or guided self study and on-the-job training. Appropriate plant personnel receive instruction in emergency plan and radiation protection procedures. Specialized training in specific areas conducted by the equipment manufacturers or other vendors is utilized as necessary. Training on a continuing basis is used to maintain a high level of proficiency in the staff.

6.3.2 EMERGENCY PLAN

The staff approved Millstone Nuclear Power Station Emergency Plan (Reference 6.3-1) addresses the criteria set forth in NUREG-0654, FEMA-REP-1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, Revision 1, November 1980 and NUREG-0737, Supplement 1. As such, the Emergency Plan provides for an acceptable state of emergency preparedness and meets the requirements of 10 CFR Part 50 and Appendix E thereto.

6.3.3 PHYSICAL SECURITY PLANS

The security plan (Reference 6.3-2) states the security measures to be employed by the licensee for the protection of Units 1, 2 and 3 at the Millstone Nuclear Power Station, Waterford, Connecticut, against radiological sabotage. The plans have been submitted in accordance with 10 CFR Part 73, Section 73.55, “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage.”

These plans include measures to deter or prevent malicious actions that could result in the release of radioactive materials into the environment though sabotage. This protection is provided through the use of armed guards, physical barriers, monitors, personnel access controls alarms, communications, response to security contingencies, and liaison with appropriate law enforcement agencies.

6.3.4 QUALITY ASSURANCE PROGRAM DESCRIPTION (QAPD) TOPICAL REPORT

The licensee has developed and implemented a comprehensive Quality Assurance Program (QAP) to ensure conformance with established regulatory requirements as set forth by the Nuclear Regulatory Commission, and accepted industry standards. The participants in the QAP assure that the design, procurement, construction, testing, operation, maintenance, repair, and decommissioning of nuclear power plants are performed in a safe and effective manner.

The QAPD Topical Report complies with the requirements set forth in Appendix B of 10 CFR Part 50, along with applicable sections of the Safety Analysis Report.

This QAP is also established, maintained and executed with regard to Radioactive Material Transport Packages as allowed by 10 CFR 71.101(f). In addition, the QAPD Topical Report is submitted periodically to the NRC in accordance with 10 CFR 50.54(a).

6.3.5 REFERENCES

6.3-1 J. F. Opeka letter to U.S. Nuclear Regulatory Commission Document Control Desk transmitting “Revision 6 to the Millstone Nuclear Power Station, Unit Numbers 1, 2, and 3, Emergency Plan,” dated November 4, 1991 [and subsequent revisions thereto submitted on an annual basis].

6.3-2 J. F. Opeka letter to U.S. Nuclear Regulatory Commission, “Millstone Nuclear Power Station, Unit Numbers 1, 2, and 3, Physical Security Plan, Revision 15,” dated December 16, 1991 and subsequent revisions thereto.

6.4 PROCEDURES

Written procedures are required for maintenance, repair, or operational activities related to the structures, systems and components which are safety related (Safety Class 1,2, or 3). Written procedures shall be established, implemented, and maintained in accordance with the Technical Specifications.

6.5 REVIEW AND AUDIT

A program describing the review and audit of activities important to and affecting station safety, has been established and complies with Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)." The program provides a system to ensure that these activities are performed in accordance with company policy, rules, and approved procedures.

6.5.1 ONSITE REVIEW

The membership, duties, areas of review responsibility, and requirements of both the plant and site operations review committees are described in the Quality Assurance Program Description (QAPD) Topical Report (Reference 6.1-1).

6.5.2 INDEPENDENT REVIEW

Independent review of activities affecting the unit's safety is performed by the Management Safety Review Committee as described in the QAPD Topical Report (Reference 6.1-1).

6.5.3 AUDITS

The audit program for activities affecting safety related systems, structures, or components is as described in the QAPD Topical Report (Reference 6.1-1).

CHAPTER 7 – DECOMMISSIONING

7.1 SUMMARY OF ACTIVITIES

Millstone Unit Number 1 was shutdown for a normal refueling outage on November 4, 1995, and has not operated since. On November 19, 1995, transfer of all fuel assemblies from the reactor vessel into the spent fuel pool (SFP) for storage was completed. On July 17, 1998, the licensee decided to permanently cease further operation of the plant. Certification to the NRC of the permanent cessation of operation and permanent removal of fuel from the reactor vessel, in accordance with 10 CFR 50.82 (a)(1)(i) & (ii), was filed on July 21, 1998 (Reference 7.1-1), at which time the 10 CFR 50 license no longer authorized operation of the reactor or placement of fuel in the reactor vessel.

The mission of the licensee is to decommission the plant safely and in a cost effective manner. The information contained in this section of the DSAR is based upon the best information currently available. The plans discussed herein may be modified as additional information becomes available or conditions change.

Specific conditions which are unique to the multi-unit Millstone Station require that certain Millstone Unit Number 1 decommissioning activities be delayed and performed concurrently with the decommissioning of Millstone Unit Numbers 2 and 3. Other considerations may dictate early scheduling of certain decommissioning activities. Therefore, the approach to decommissioning Millstone Unit Number 1 can best be described as a modified SAFSTOR. In this approach, decontamination and dismantlement activities may be undertaken early in the decommissioning wherever it makes sense from a safety or economic viewpoint. For instance, given the future uncertainty over access to a low level waste disposal site, early shipment of certain components will occur. The amount of decommissioning work completed prior to a SAFSTOR period depends upon a number of factors currently under evaluation.

Both the DECON and the SAFSTOR options are approaches found acceptable to the NRC in its Final Generic Environmental Impact Statement (GEIS) (Reference 7.1-2).

Completion of the decommissioning schedule is contingent upon three key factors:

- continued access to licensed low level waste (LLW) disposal sites,
- removal of spent fuel from the site, and
- timely funding of the decommissioning activities.

Currently Millstone Unit Number 1 has access to Chem-Nuclear Systems’ Barnwell, S.C. disposal site and to the Envirocare disposal site in Tooele County, Utah. Escalation costs for the disposal of waste have been incorporated into financial planning. Additionally, the licensee has considered the possibility that during the decontamination and dismantlement phases, access to the Barnwell low level waste disposal site could be denied or that the facility could be closed.

The unavailability of the DOE high level waste repository may affect the decontamination and dismantlement schedule for Millstone Unit Number 1. Delays in the operation of the repository have resulted in a significant increase in the cost of decommissioning and, may require the use of an independent spent fuel storage installation (ISFSI).

Although storage of the Millstone Unit Number 1 spent fuel in an ISFSI is presented in this DSAR as an option; an ISFSI has been contracted to ensure the continued operation of Millstone Unit Numbers. 2 and 3. Currently, after spent nuclear fuel is removed from the Unit 2 and Unit 3 reactor core; it is safely stored in the existing SFPs. Capacity of these pools was designed with the assumption the DOE high level waste repository would provide permanent storage. However, the site selection, construction and licensing of such a repository have been delayed. As is the case with other nuclear facilities as the SFPs approach full capacity, spent fuel from Millstone Unit Numbers 2 and 3 will be stored in the ISFSI. A description of the ISFSI is contained in the Unit Number 3 Final Safety Analysis Report.

Under any eventuality such as unavailability of a LLW disposal site, temporary shortfall in decommissioning funding, or other unforeseen circumstances, 10 CFR 50.82 requires the licensee to maintain the capability to suspend decontamination and dismantlement.

7.1.1 DECOMMISSIONING APPROACH

The licensee is planning on decommissioning Millstone Unit Number 1 using a modified SAFSTOR approach in which the decontamination and dismantlement of the systems, components, plant structures and facilities (i.e., DECON) are completed prior to and following a SAFSTOR period. In this plan, an ISFSI may be constructed and the transfer of spent fuel from the spent fuel pool (SFP) could be completed during the SAFSTOR period. The SAFSTOR period ends with decontamination and dismantlement of any remaining systems, structures, and components commence in coordination with Millstone Unit Number 2 and Millstone Unit Number 3 decommissioning.

Spent fuel shipments from the ISFSI to DOE are scheduled, when practicable, following the repository commencing operations. Delays in the operation of the repository limits the transfer of fuel and increases the cost of long term spent fuel storage.

The following discussion provides an outline of the current decommissioning plan activities completed to date and the remaining significant activities. The planning required for each decommissioning activity, including the selection of the process to perform the work, is completed prior to the start of work for that activity.

7.1.1.1 Planning

The planning includes implementation of a site characterization plan, preparation of a detailed decommissioning plan, and the engineering development of task work packages. The detailed engineering required to support the decontamination and dismantlement of systems, structures, and components are performed prior to the start of field activities.

Significant activites performed to date include:

- Establishment of a spent fuel pool island.
- Sent fuel pool cleanup.
- Removal and disposal of legacy resins and filter media.
- Removal, processing, and disposal of irradiated hardware from the reactor vessel including control rod blades and in-core instrumentation.
- Reactor vessel internals segmentation, including the upper core grid.
- Drain down of the reactor cavity and reactor vessel.
- Installation of a radiation shielding package over the reactor vessel head and cavity floor.

The following activities remain:

- Evaluate and choose a dry fuel storage system, if pursued. Investigate and prepare for the design and licensing of an ISFSI and prepare procurement specifications for a fuel canister system and ancillary equipment.

7.1.1.2 Site Characterization

During the initial portion of the planning period a detailed site characterization was undertaken during which radiological, regulated and hazardous wastes were identified, categorized, and quantified. Surveys were conducted to establish the contamination and radiation levels throughout the Millstone Unit Number 1 portion of the site. This information is used in developing procedures to ensure that hazardous, regulated or radiologically contaminated materials are removed and to ensure that worker exposure is maintained as low as reasonably achievable (ALARA). Selected surveys of the outdoor areas in the vicinity of Millstone Unit Number 1 may be performed, although a detailed survey of the environs would likely be deferred pending decommissioning of Millstone Unit Numbers 2 and 3. It is worthwhile to note that site characterization is a process that continues throughout decommissioning. As decontamination and dismantlement work proceed, surveys are conducted to maintain current characterization and that decommissioning activities are adjusted accordingly.

The activation analysis of the reactor internals, the reactor vessel, and the biological shield wall was undertaken as a part of the site characterization. Using the results of this analysis, these components were classified in accordance with 10 CFR 61 and form the basis for the detailed plans for their packaging and disposal. The interior grid portion of the top guide structure was determined to be greater than class “C” (GTCC) material, was segmented from the reactor vessel, and is stored in the spent fuel pool in canisters sized to be compatible with ISFI dry storage containers.

7.1.1.3 Decontamination

The objectives of the decontamination effort are two fold. First, to reduce the radiation levels throughout the facility in order to minimize personnel exposure during dismantlement. Second, to clean as much material as possible to unrestricted use levels, thereby permitting non radiological demolition and minimizing the quantities of material that must be disposed of by burial as radioactive waste.

The need to decontaminate structures, systems, and components are determined by the schedule to dismantle them and by plant conditions. Early dismantling of contaminated components and systems may benefit from decontamination activities by reducing the radiation exposure to the workforce. Late dismantling may not require the components and systems to be decontaminated since the decay of the radiation sources reduces the radiation levels by significant amounts.

Chemical decontamination of the reactor recirculation system may provide value through reduced worker dose. An evaluation is performed to determine whether the expected reduction in the accumulated workforce exposure is justified by the costs associated with the decontamination. The evaluation results are sensitive to the amount and type of work to be performed prior to a SAFSTOR period. Any decontamination method used employs established processes with well-understood chemical interactions. The resulting waste is disposed of in accordance with plant procedures and applicable regulations.

The second objective of the decontamination effort is achieved by decontaminating structural components including steel framing and concrete surfaces. The method used to accomplish this is mechanical and requires the removal of the surface or surface coating. This process is used regularly in industrial and contaminated sites.

7.1.1.4 Major Decommissioning Activities

As defined in 10 CFR 50.2 a "major decommissioning activity" is “any activity that results in permanent removal of major radioactive components, permanently modify the structure of the containment, or results in dismantling components for shipment containing GTCC waste in accordance with 10 CFR 61.55.”

Major decommissioning activities completed to date include the removal of the drywall head and removal of the reactor vessel internals by segmentation. The drywall head was sectioned and sent to a metal processor. The reactor vessel internals, classified as GTCC, are limited to the interior portion of the top guide structure, which has been segmented from the reactor vessel and is stored in the spent fuel pool. The reactor cavity and reactor vessel have been drained. Without the GTCC internals present, several options are available for later removal and disposal of the reactor vessel: segmentation, sectioning into pieces, or disposal as an intact package.

Based on an evaluation of activity levels, ease of execution, personnel exposure, schedule constraints, disposal facility availability, and cost, segmentation of the internals may be postponed until after the fuel is removed from the SFP.

Removal of the reactor vessel follows the removal of the reactor internals and may not occur until after a SAFSTOR period. It is likely that the vessel would be removed by sectioning or segmenting. Vessel sectioning or segmenting permits a substantial portion of the waste to be sent to a waste re-processor instead of a near surface disposal site. The dismantling of the drywell and suppression chamber is undertaken as part of the reactor building demolition.

7.1.1.5 Other Decommissioning Activities

Other decommissioning activities include:

- Preparation and submittal of the following documents:

    1. A license termination plan pursuant to 10 CFR 50.82

    2. A spent fuel management program, pursuant to 10 CFR 50.54(bb)

In addition to the major decommissioning activities listed above, the following decommissioning activities include:

- Hazardous and regulated materials (e.g., asbestos, lead, mercury, PCBs, oil, chemicals) are identified during characterization and plans are developed for the removal of these materials.

- Plant components removed from the Turbine Building include the Turbine Generator, Condenser, Feedwater Heaters, Moisture Separators and miscellaneous system and support equipment.

- Miscellaneous solid waste removed include: control rod blades, local power range monitors, spent resins and filters, the Reactor Pressure Vessel Head Insulation assembly, the de-tensioner platform, and the Refuel Floor shield plugs. The larger components may be segmented and packaged for removal through the Reactor Building hatchway.

- Liquid wastes are processed and discharged using plant procedures in accordance with applicable regulatory requirements as the liquid waste inventories become available. Initially the inventories of the plant water systems are processed. Upon completion of the segmentation and packaging of the reactor vessel internals, the reactor cavity and reactor may be drained and the waste inventory processed. When the spent fuel is removed, the SFP is drained and the water processed. Systems are then isolated and deactivated in a sequence compatible with the operations previously described. Spent fuel pool systems are isolated after removal of the spent fuel.

Radioactively contaminated or activated materials are removed from the site as necessary to allow the site to be released for unrestricted access. Low level waste is processed in accordance with plant procedures and existing commercial options, and sent to licensed disposal facilities or waste processors for further volume reduction. Wastes may be incinerated, compacted, or otherwise processed by authorized and licensed contractors, as appropriate. Mixed wastes, if any, are managed according to all applicable federal and state regulations. Mixed wastes are transported only by authorized and licensed transporters and shipped only to authorized and licensed facilities.

7.1.1.6 Final Site Survey and Termination of License

Since Millstone Unit Number 1 and Millstone Unit Number 2 are contiguous and have common structural boundaries, the plans for building demolition and for the license termination survey are implemented as a coordinated evolution for the two units. Consequently, the schedule for the Millstone Unit Number 1 license termination is constrained by the need to terminate the Part 50 license coincident with that of Millstone Unit Number 2. As a result of this delay in requesting license termination, the final site survey using Reference 7.1-4 may proceed in two phases: 1) internal structures surveyed as decontamination and dismantlement are completed, and 2) external areas surveyed in conjunction with completion of the Unit 2 decontamination and dismantlement.

The licensee is required to prepare a License Termination Plan (LTP) for Millstone Unit Number 1. The LTP defines the details of the final radiological survey to be performed once the decontamination activities are completed. The LTP conforms to the format defined in Reference 7.1-5 and addresses the limits of 10 CFR 20 using the pathways analysis defined in Reference 7.1-4. Use of this guidance ensures that survey design and implementation is conducted in a manner that provides a high degree of confidence that applicable NRC criteria are satisfied. Once the survey is complete, the results are provided to the NRC in a format that can be verified.

7.1.1.7 Site Restoration

The restoration of the Millstone Unit Number 1 area of the Millstone site will be undertaken when the 10 CFR Part 50 license for Millstone Unit Number 1 is terminated. This event may coincide with Millstone Unit Numbers 2 and 3 license terminations. Buildings, structures, and other facilities which are not currently known to be radiologically contaminated, such as the Strainer Pit, Intake Structure, and the Discharge Structure are dismantled, as part of the building demolition effort after the final license termination survey for Millstone Unit Number 1 is complete. These buildings can be removed late in the building demolition phase since there is no decommissioning operational need to remove them earlier. Site restoration requires that all buildings be removed to an elevation 3 feet below grade or to an elevation consistent with the removal of the necessary amounts of contaminated material.

7.1.2 STORAGE OF RADIOACTIVE WASTE

Table 5.4-1 of the GEIS (Reference 7.1-2) provides an estimate for low-level waste disposal from a referenced boiling water reactor (BWR) of 18,975 cubic meters (669,817 cubic feet) for both the DECON and SAFSTOR options. The licensee estimates the low-level waste burial volume for Millstone Unit No. 1, will be at or below this value for the modified SAFSTOR alternative. The licensee’s estimate includes, by a reduction of approximately 40 percent (industry standard), the utilization of present-day volume reduction techniques. For waste requiring deep geological burial, i.e.,GTCC waste, the licensee estimates that the volume for Millstone Unit Number 1 is at or below the 11.5 cubic meters anticipated for a reference BWR discussed in Section 5.4 of the GEIS. These estimates support the conclusion that the previously issued environmental statements are bounding since the disposal of waste requires fewer resources, i.e., less waste disposal facility area, than what was considered in the GEIS.

7.1.2.1 High Level Waste

Congress passed the "Nuclear Waste Policy Act" in 1982, assigning the responsibility for disposal of spent nuclear fuel created by the commercial nuclear generating plants to DOE. This legislation also created a Nuclear Waste Fund to cover the cost of the program, which is funded, in part, by the sale of electricity from the Millstone Unit Number 1 plant. The current DOE estimate for startup of the federal waste management system is 2010. For planning purposes, the licensee has assumed that the high-level waste repository or some interim storage facility will not be operational until at least 2010. Shipments of fuel and GTCC waste to DOE are planned to be directly from the ISFSI.

The spent fuel is currently stored in the SFP. The licensee may license a dry, ISFSI. Fuel will be transferred from the pool and stored temporarily on site using licensed canisters. For the period of time when the fuel will be stored in the SFP, the systems necessary for SFP operations will be consolidated into an "Island" concept and configured for SFP clean-up and cooling.

7.1.2.2 Low Level Waste

Radioactively contaminated or activated materials are removed to allow the site to be released for unrestricted access. Low level waste is processed in accordance with federal and state regulations, plant procedures and existing commercial options, and transported to license disposal facilities.

7.1.2.3 Waste Management

A major component of the total cost of decommissioning Millstone Unit Number 1 is the cost of packaging and disposing of systems, components and structures, contaminated soil, water and other plant process liquids. A waste management plan incorporates the most cost effective disposal strategy consistent with regulatory requirements for each waste type. The waste management plan will be based on the evaluation of available methods and strategies for processing, packaging, and transporting radioactive waste in conjunction with the available disposal facility options and associated waste acceptance criteria.

7.1.3 RADIATION EXPOSURE MONITORING

Personnel radiation exposure is maintained ALARA and monitoring is conducted in accordance with the radiation protection program described in Chapter 4. Exposure specifically related to decommissioning activities is identified and tracked. Exposure monitoring is used to identify activities that are causing excessive exposure and to initiate corrective actions to reduce personnel exposure.

7.1.4 REFERENCES

7.1-1 Letter B17388 from Bruce D. Kenyon to U. S. Nuclear Regulatory Commission,"Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor," dated July 21, 1999.

7.1-2 U. S. Nuclear Regulatory Commission report NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated August, 1988.

7.1-3 Letter B17790 from R. P. Necci to U. S. Nuclear Regulatory Commission, "Post Shutdown Decommissioning Activities Report," dated June 14, 1999.

7.1-4 U. S. Nuclear Regulatory Commission report NUREG-1575, "Multi-Agency Radiation Site Survey and Investigation Manual (MARSSIM)," Final Report.

7.1-5 U. S. Nuclear Regulatory Commission report NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," (Currently in Draft form).

7.2 ESTIMATE OF RADIATION EXPOSURE

The decommissioning of Millstone Unit No. 1 is accomplished with no significant adverse environmental impacts, in that no Millstone Unit No. 1 site specific factors should alter the conclusions of the GEIS (Reference 7.1-2) or the Millstone Environmental Statement. The radiation dose to the public during decommissioning is typically minimal. Decommissioning workers receive a fraction of the dose which radiation workers receive in an operating plant. The low-level radioactive waste that is removed from the site occupies only a small portion of the burial volume at approved waste disposal sites. The non-radiological environmental impacts are temporary and not significant.

The occupational dose exposure for decommissioning Millstone Unit No. 1 is less than described in the GEIS because of two main reasons. First, the licensee initiated a zinc injection program for Millstone Unit No. 1 in 1987 that significantly reduced the buildup of contaminated corrosion products during the remaining plant operation period. Second, with the plant shutdown since 1995, natural decay of leading radionuclides have reduced overall plant general dose levels significantly by the time decontamination and decommissioning activities occur.

The activities identified in this chapter resemble the DECON option. Therefore, the modified SAFSTOR occupational and public dose exposure is compared to the DECON option dose in the GEIS. The occupational and public dose effects for a modified SAFSTOR alternative is bounded by the DECON option. The exposure from decontamination and dismantlement activities and the exposure during transportation of the low-level wastes is included in this dose estimate. NUREG-0586 (Reference 7.1-2), Table 5.3-2, estimates a total occupational dose of 18.74 person-Sv (1874 person-rem) for the DECON alternative for the reference BWR plant. The values estimated by the licensee will be at or below this value.

7.2.1 NUCLEAR WORKER

Detailed estimates for external occupational radiation exposure that accumulate dose for decommissioning workers during the dismantlement program are developed based on a task by task analysis of personnel hours and expected radiation dose rates associated with each task. These estimates are based on the following:

1. ALARA principles are implemented.
2. Radiation exposure is monitored to identify jobs that are causing excessive exposure and corrective actions are taken to reduce the severity.

7.2.2 GENERAL PUBLIC

Radiation dose to the public is maintained below comparable levels when the plant was operating through the continued application of radiation protection and contamination controls combined with the reduced source term available in the facility.

7.2.3 NORMAL TRANSPORTATION

Shipments of spent fuel and radioactive wastes are performed by exclusive use vehicles. Shipments will be in accordance with the Department of Transportation (DOT) regulations. Generic industry estimates of the doses from routing transportation of radioactive materials are based on the following assumptions:

- Two truck drivers during a 500 mile trip would probably spend no more than 12 hours inside the cab and 1 hour outside the cab at an average distance of 6 feet from the truck.
- Normal truck servicing en route would require that two garage men spend no more than 10 minutes about 6 feet from a shipment.
- Onlookers from the general public might be exposed to radiation when a truck stops for fuel or for the drivers to eat. The onlooker dose is calculated on the basis that 10 people spend an average of 3 minutes each at a distance of about 6 feet from a shipment.
- The cumulative dose to the general public from truck shipments is based on population dose of 2.3×10^{-6} man-rem per km.

NUREG/CR-0672, Table 11.4-2, provided a generic estimate of the routing radiation doses from truck transportation of radioactive wastes. The doses are based on the maximum allowable dose rates for each shipment in exclusive use trucks and are conservatively high, on the number of truck shipments, and on the shipping distances. The estimated external radiation dose for routing transportation operations is 110 man-rem to transportation workers and 10 man-rem to the general public.

The licensee estimates the volume of both high level and low level wastes to be less than the volumes used in NUREG/CR-0672. The total number of shipments of radioactive wastes is less than those used to determine the exposure in the NUREG/CR, and therefore the exposure to the transportation workers and the general public is less than those identified above.

7.3 CONTROL OF RADIATION RELEASES ASSOCIATED WITH DECOMMISSIONING EVENTS

During the decommissioning, processes may concentrate source terms. Non-routine events may occur with the potential to disperse the source term. This section of the DSAR establishes controls and requirements to maintain potential consequences of such event to below analyzed accidents.

7.3.1 IN PLANT EVENTS

The DBA for Millstone Unit Number 1 is the fuel handling accident and a detailed discussion can be found in DSAR Chapter 5. The acceptance criteria for all other potential events at the plant are controlled such that the potential consequences of any postulated event are less than 1 REM at the exclusion area.

7.3.2 TRANSPORTATION ACCIDENTS

Transportation accidents have a wide range of severities. Most accidents occur at low speeds and have relatively minor consequences. In general, as speed increase, accident severity also increases. However, accident severity is not a function of vehicle speed only. Other factors, such as the type of accident, the equipment involved, and the location can have an important bearing on accident severity.

Damage to a package in a transportation accident is not directly related to accident severity. In a series of accidents of the same severity, or in a single accident involving a number of packages, damage to packages may vary from none to extensive. In relatively minor accidents, serious damage to packages can occur from impacts on sharp objects or from being struck by other cargo. Conversely, even in very severe accidents, damage to packages may be minimal.

The probabilities of truck accidents used in the NUREG/CR-0672 study were based on accident data supplied by the DOT. Accidents are classified into five categories as functions of speed and fire duration. The five categories in order of increasing severity are: minor, moderate, severe, extra severe, and extreme. Table N.5-3 of NUREG/CR-0672 provides the probabilities of occurrence for each classification.

Estimated accident frequencies, release amounts and radiation doses to the maximum exposed individuals for selected accidents for transportation of radioactive material are discussed in Appendix N.5.2.3 of NUREG/CR-0672. The frequencies are calculated by multiplying the total distance of transport with the total probability of accident per distance traveled for each accident severity class.

The maximum exposed individual is assumed to be located 100 meters from the point of a transportation accident. The calculated dose values provided in Table N.5.6 of NUREG/CR-0672 are the first year dose and the fifty year dose commitment to the bone, lung, thyroid and whole body.

The licensee anticipates that site specific analysis on the expected number of shipments and the shipping distance will confirm its applicability to the generic analysis provided in NUREG/CR-0672.

7.4 NON-RADIOLOGICAL ENVIRONMENTAL IMPACTS

The non-radiological environmental impacts from the Millstone Unit Number 1 decommissioning is temporary and not significant. The largest occupational risk associated with the decommissioning is the risk of industrial accidents. This risk is minimized by adherence to work controls during decommissioning similar to the procedures followed during power operation. Procedures controlling work related to asbestos, lead, and other non-radiological hazards remain in place during the decommissioning. The primary environmental effects of the decommissioning are temporary and include small increases in noise levels and dust in the immediate vicinity of the site, and small increases in truck traffic to and from the site for hauling equipment and waste. These effects are similar to those experienced during normal refueling outages and certainly less severe than those present during the original plant construction. No significant socioeconomic impacts or impacts to local culture, terrestrial or aquatic resources have been identified.

7.4.1 ADDITIONAL CONSIDERATIONS

While not quantitative, the following considerations are also relevant to concluding that decommissioning activities do not result in significant environmental impacts not previously reviewed:

- The release of effluents continues to be controlled by plant license requirements and plant operating procedures throughout the decommissioning.

- With respect to radiological releases, Millstone Unit No. 1 continues to operate in accordance with the Offsite Dose Calculation Manual during decommissioning.

- Release of non-radiological effluents continues to be controlled per the requirements of the NPDES and State of Connecticut permits.

- Systems used to treat or control effluents during power operation are either maintained or replaced by temporary or mobile systems for the decommissioning activities.

- Radiation protection principles used during plant operations remain in effect during decommissioning to ensure that protective techniques, clothing, and breathing apparatus are used as appropriate.

- Sufficient decontamination and source term reduction prior to dismantlement are performed to ensure that occupational dose and public exposure do not exceed those estimated in the Final Generic Environmental Impact Statement (Reference 7.1-2.

- Detailed site radiological surveys are performed prior to starting the waste campaigns to confirm the burial volume of low-level radioactive waste and highly activated components which require deep geological disposal.

- Transport of radioactive waste is in accordance with plant procedure, applicable Federal regulations, and the requirements of the receiving facility.

- Plant ventilation systems, or alternate, temporary systems, are maintained as long as needed in areas they service.

- Site access control during decommissioning ensures that residual contamination is minimized or eliminated as a radiation release pathway to the public.

FIGURE 1.2–1 PLOT PLAN

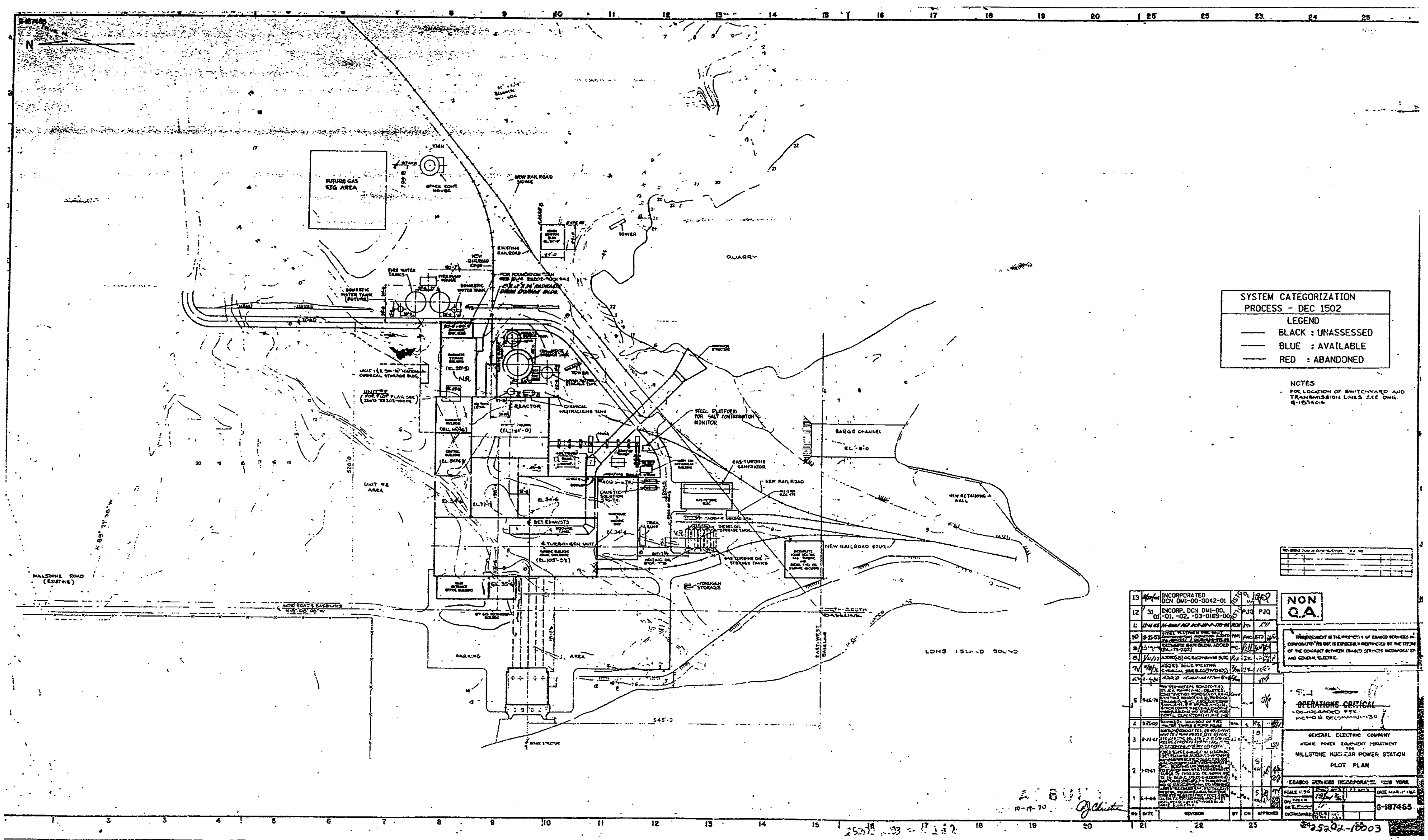

FIGURE 1.2 - 2A GENERAL ARRANGEMENT RAD WASTE BUILDINGS - PLANS

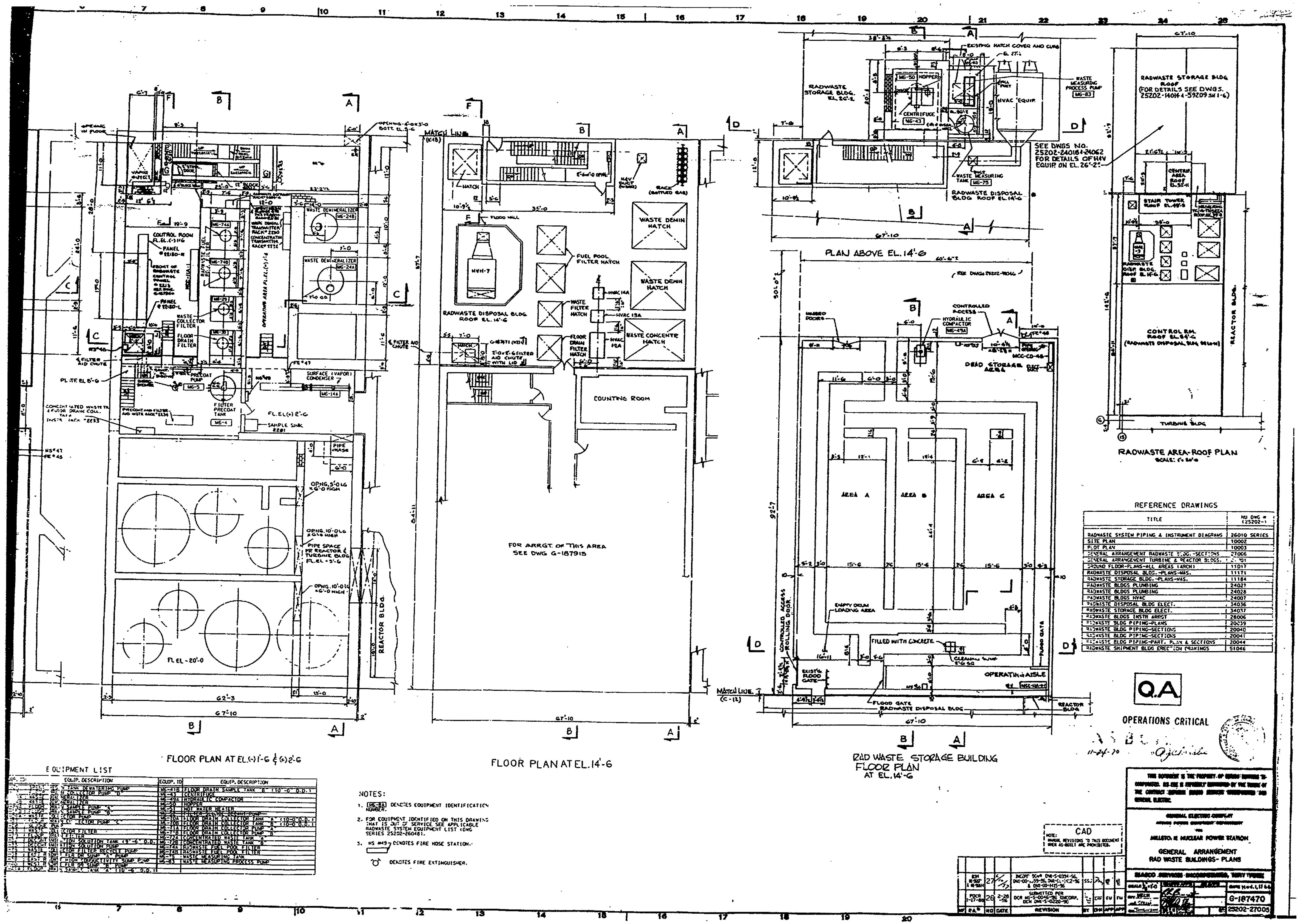

FIGURE 1.2 - 2B GENERAL ARRANGEMENT RAD WASTE BUILDINGS - PLANS

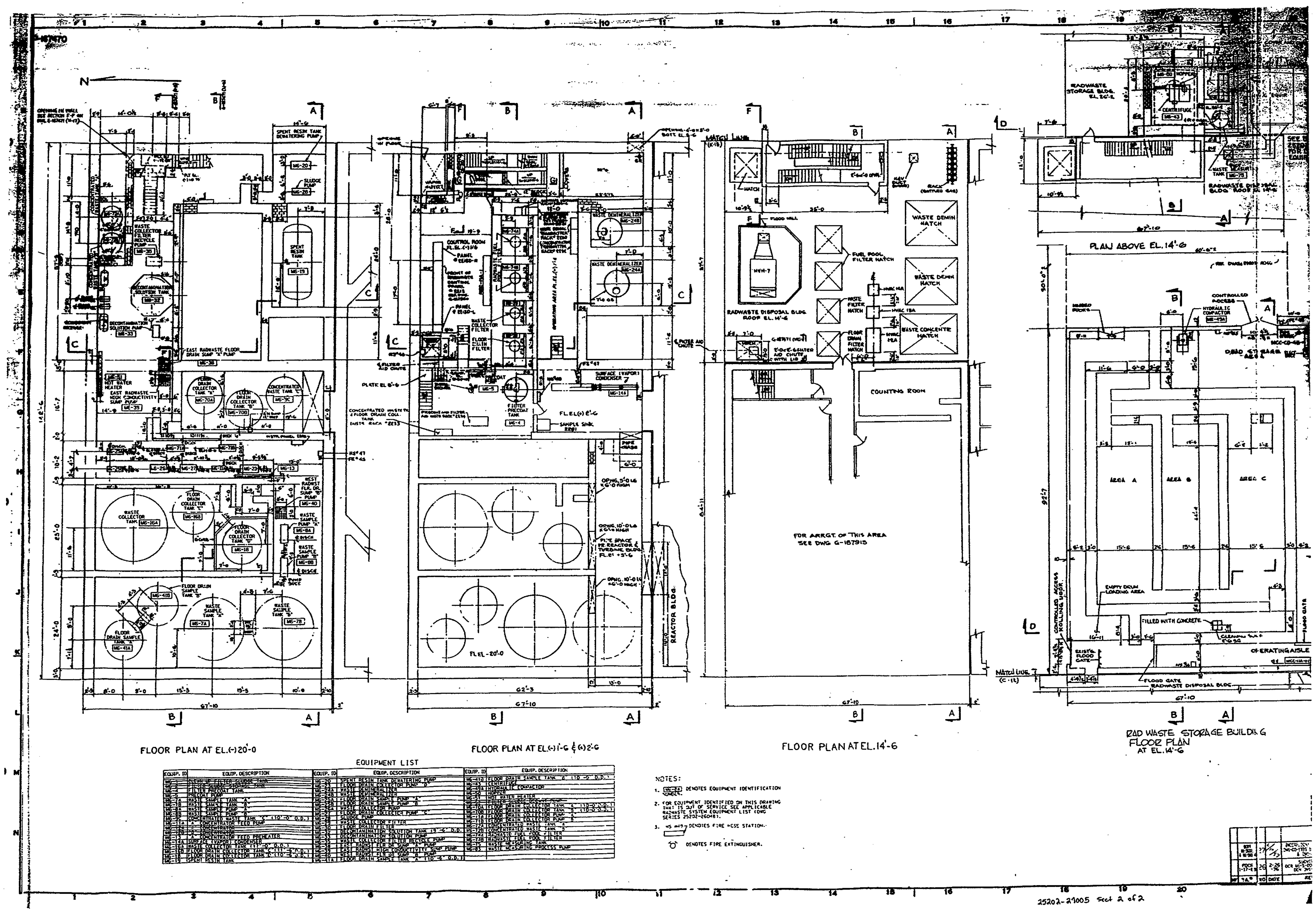

FIGURE 1.2 - 3 GENERAL ARRANGEMENT BUILDINGS RAD WASTE BUILDINGS - SECTIONS

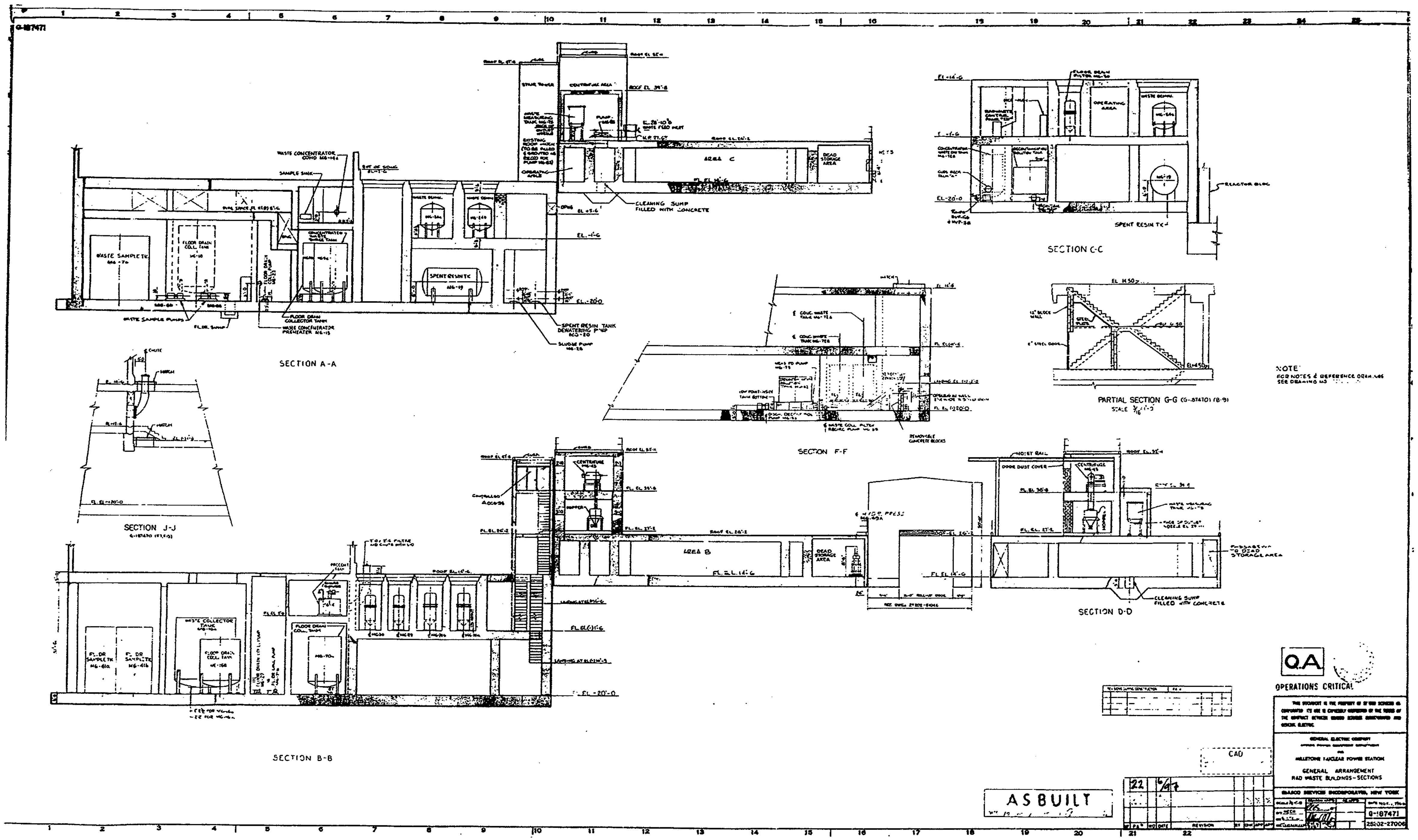

FIGURE 2.1–1 GENERAL SITE LOCATION

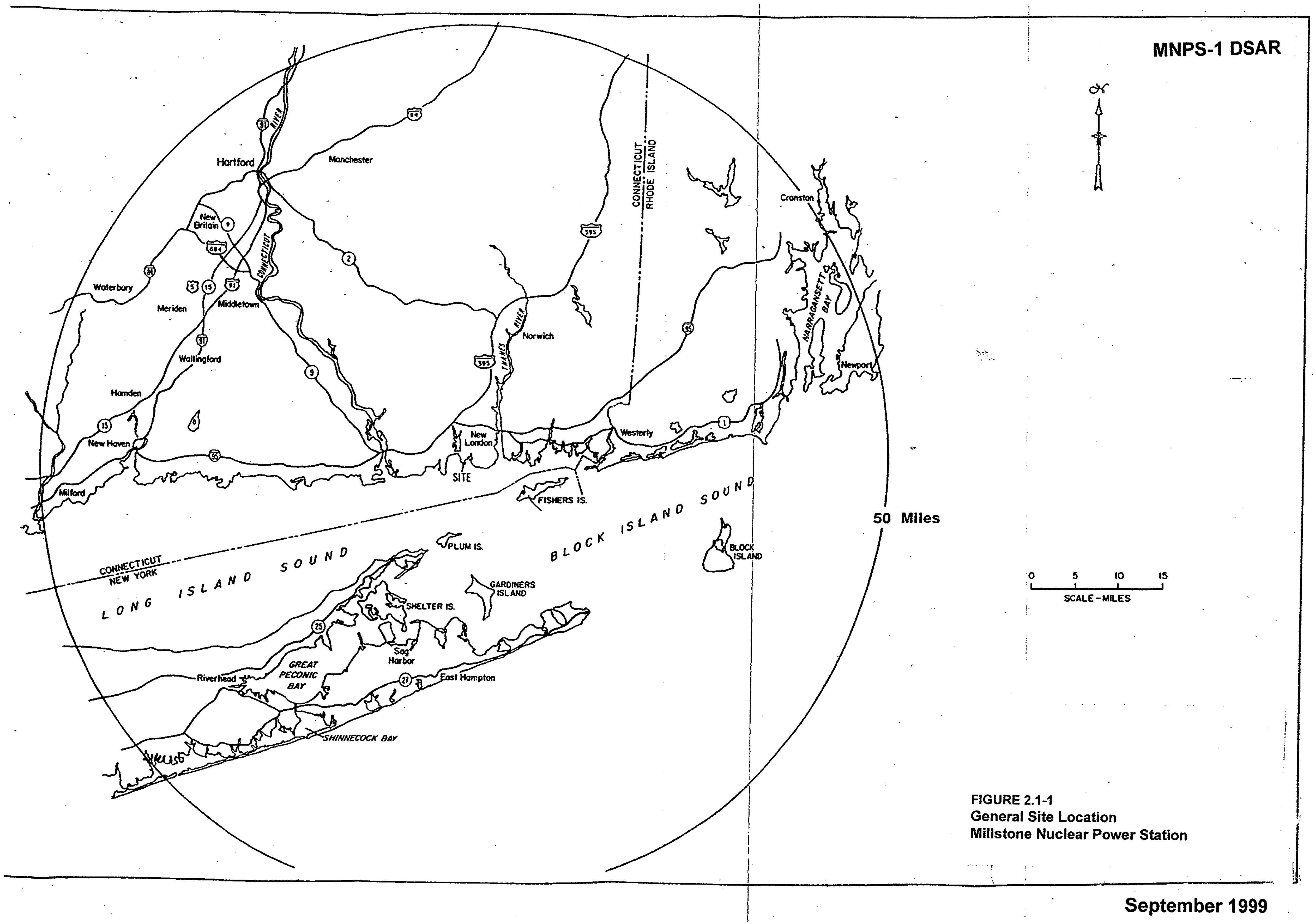

FIGURE 2.1-1
General Site Location
Millstone Nuclear Power Station

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FIGURE 2.1–2 GENERAL VICINITY

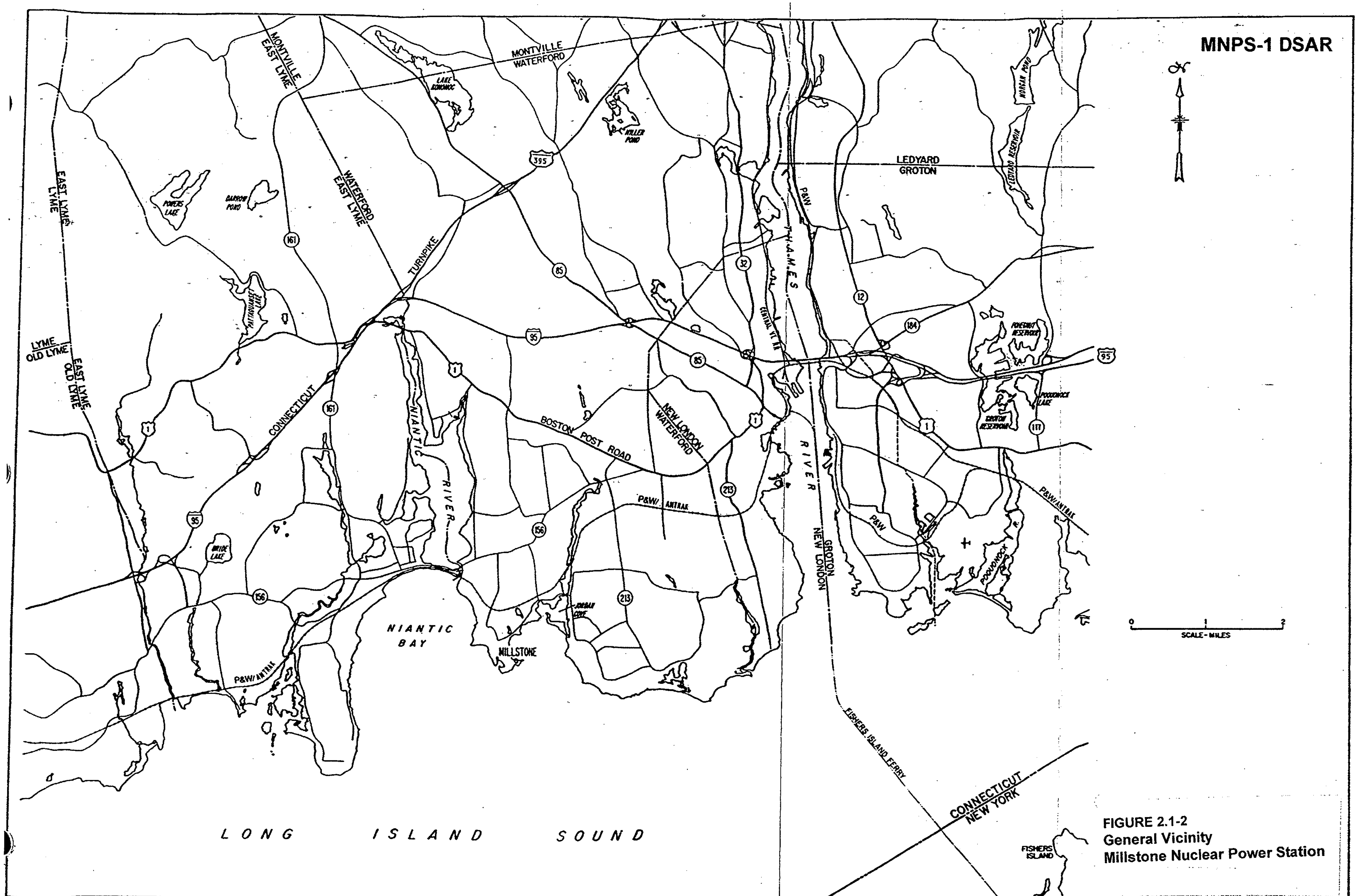

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FIGURE 2.1–3 SITE LAYOUT

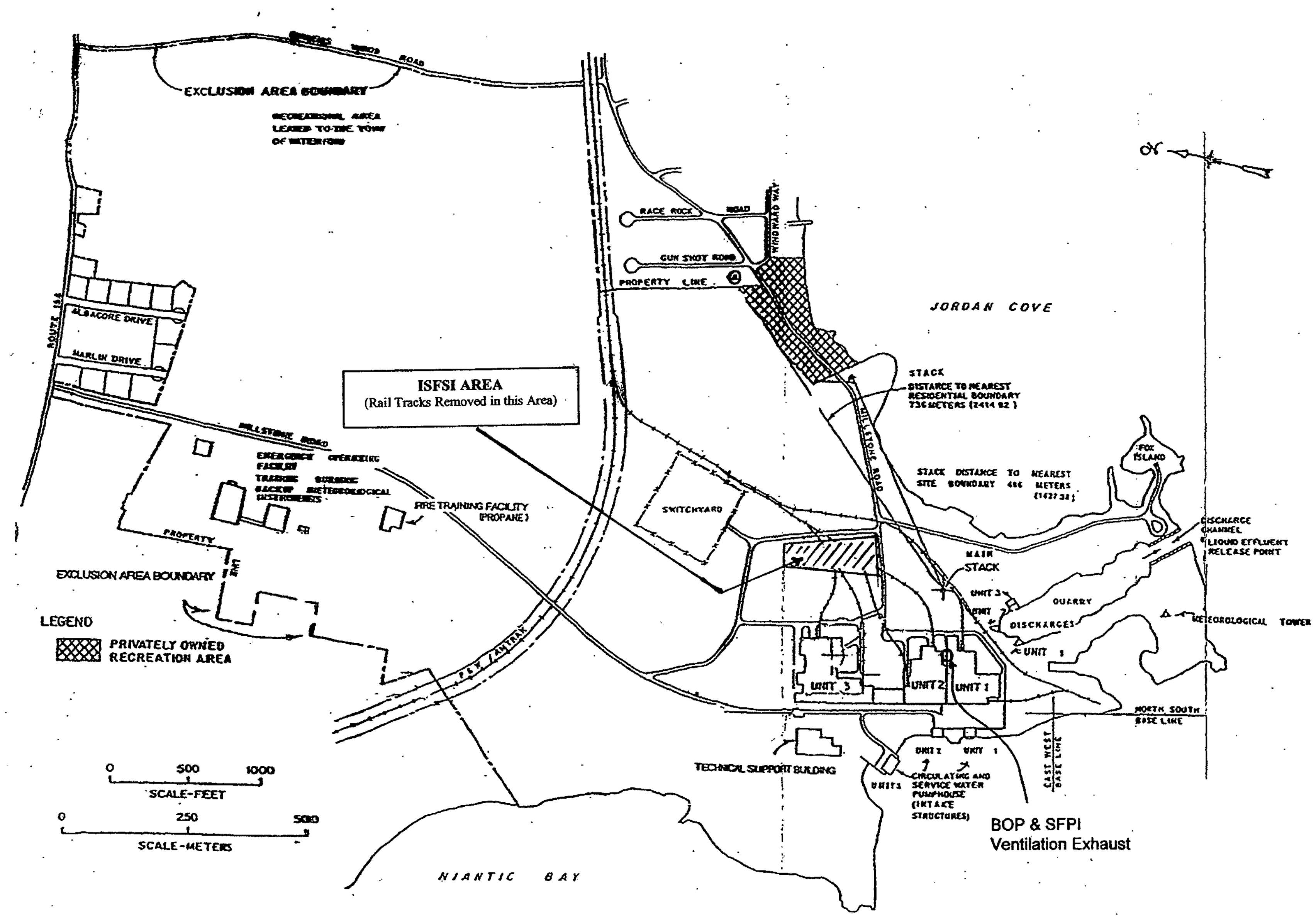

FIGURE 2.1–4 SITE PLAN

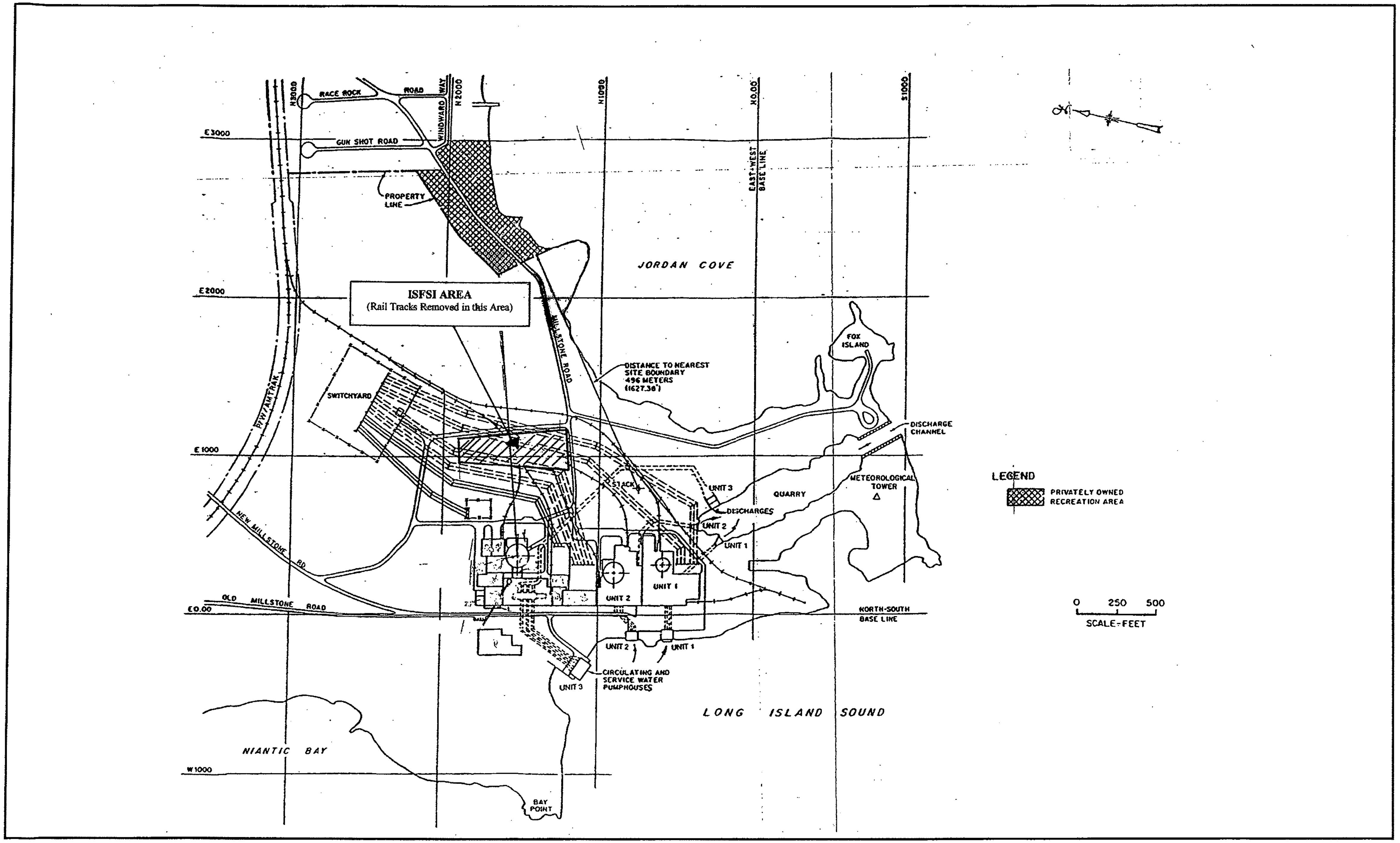

FIGURE 2.1–5 TOWNS WITHIN 10 MILES

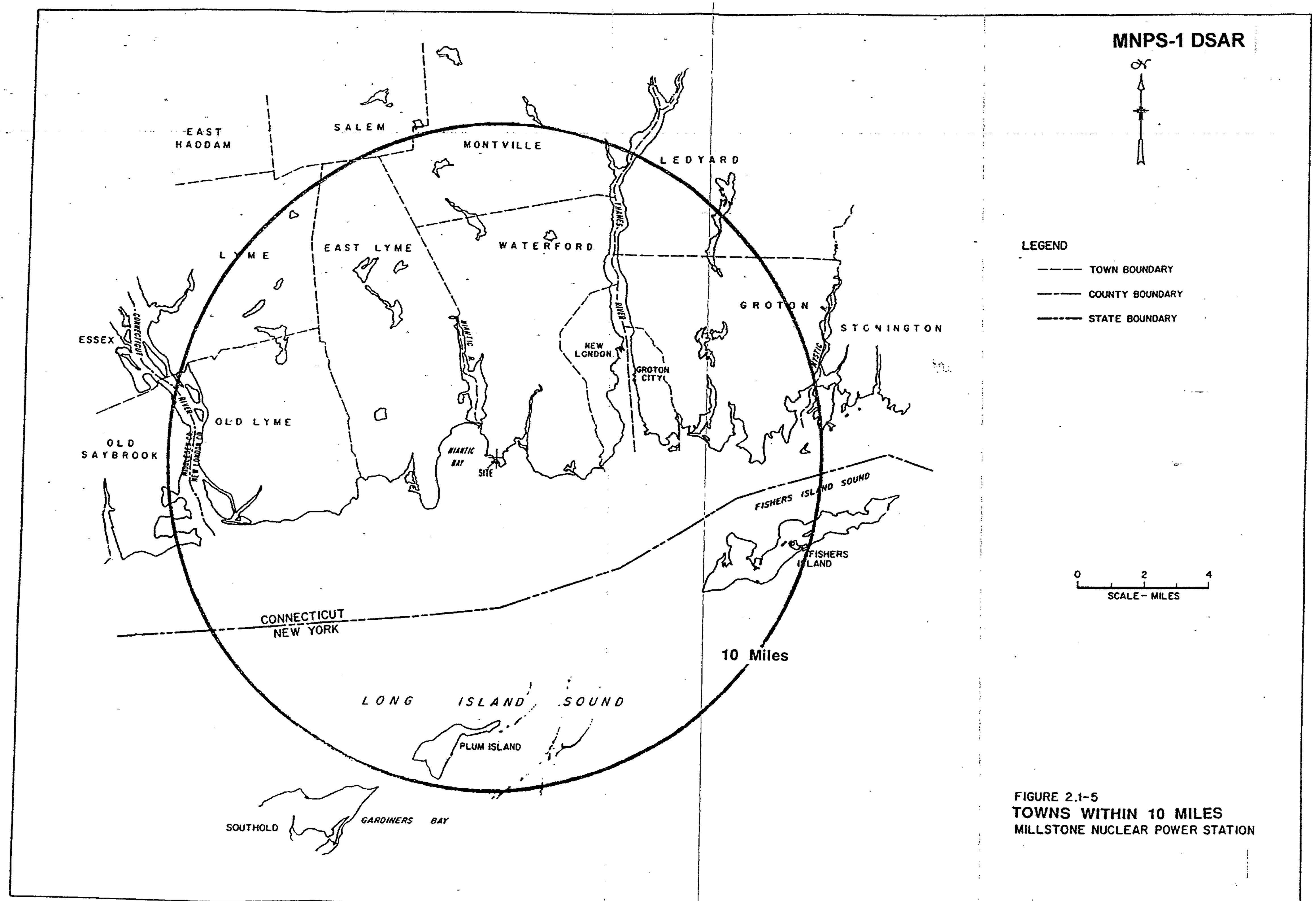

FIGURE 2.1-5
TOWNS WITHIN 10 MILES
MILLSTONE NUCLEAR POWER STATION

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FIGURE 2.1–6 POPULATION SECTORS FOR 0 - 10 MILES

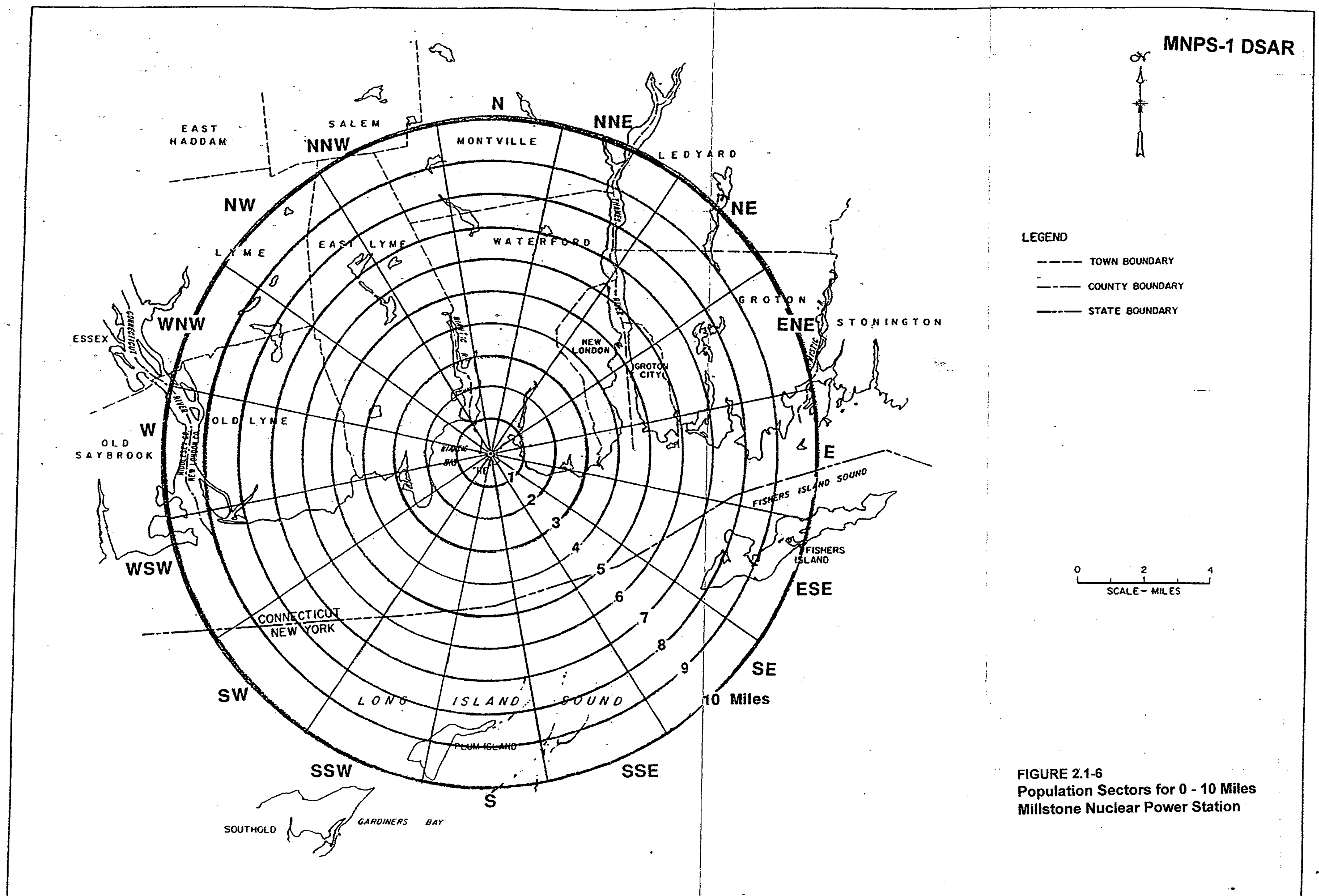

FIGURE 2.1-6
Population Sectors for 0 - 10 Miles
Millstone Nuclear Power Station

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FIGURE 2.1–7 POPULATION SECTORS FOR 0 - 50 MILES

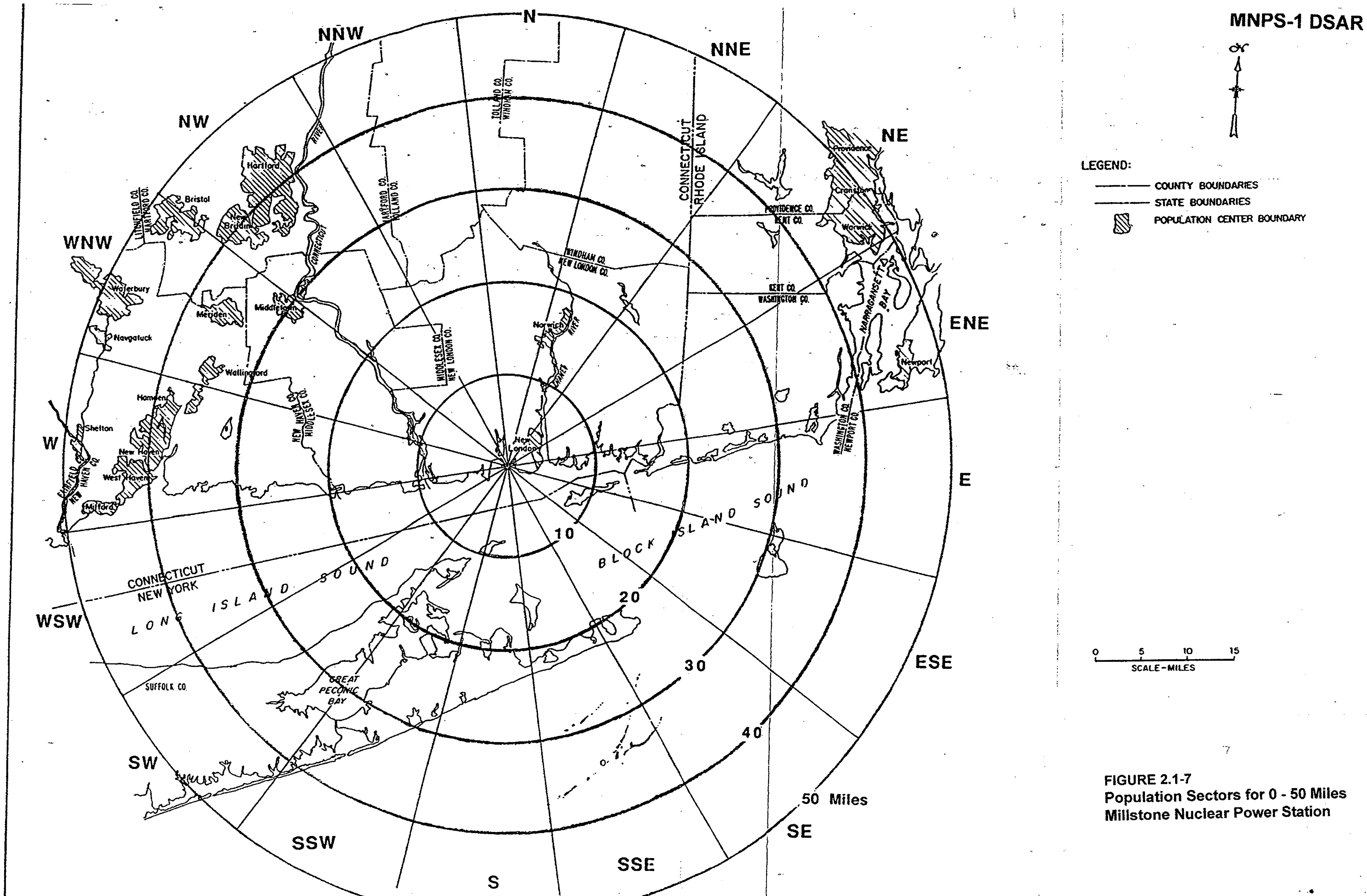

FIGURE 2.1-7
Population Sectors for 0 - 50 Miles
Millstone Nuclear Power Station

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FIGURE 2.1–8 ROADS AND FACILITIES IN THE LPZ

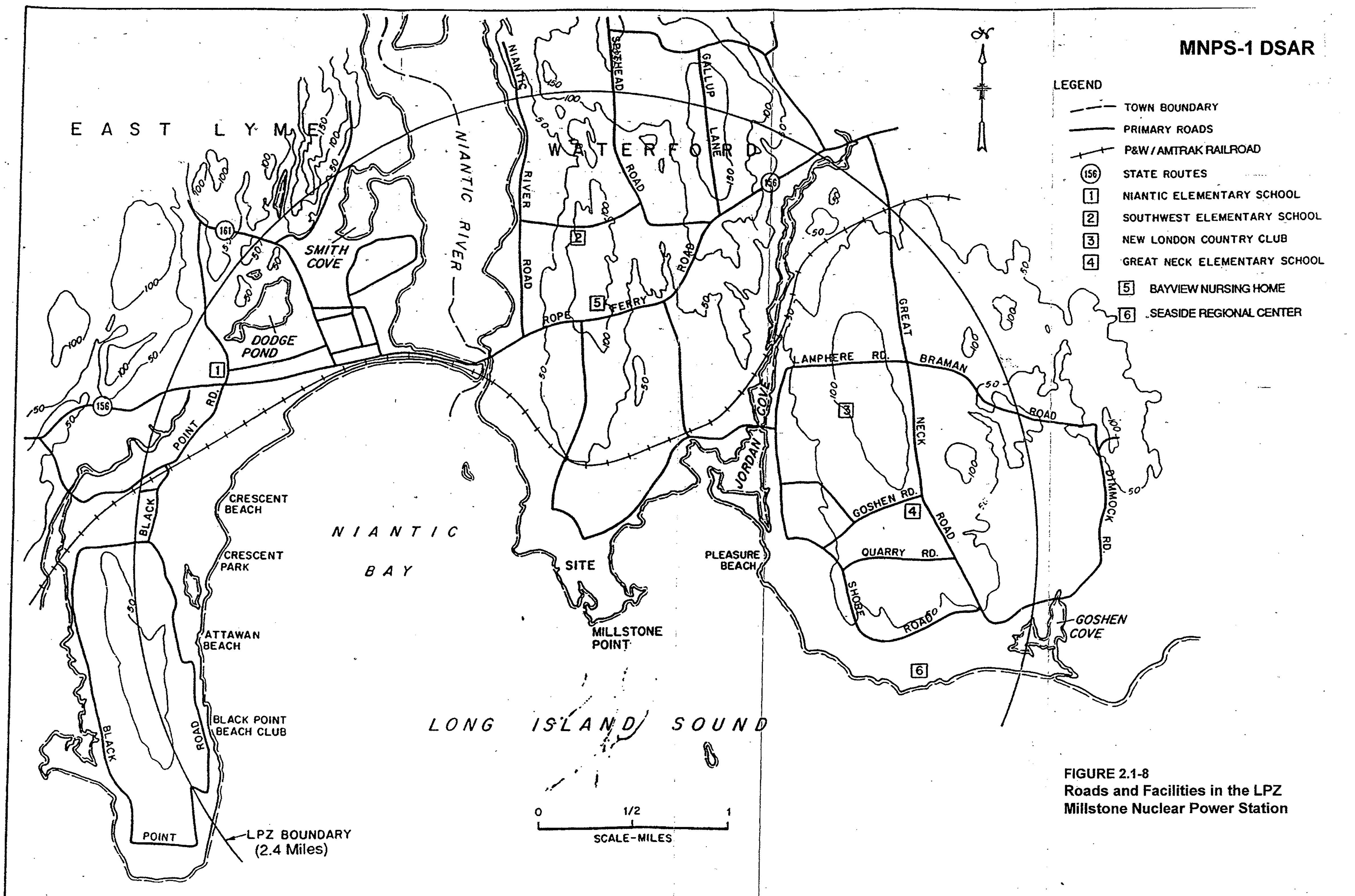

FIGURE 2.1-8
Roads and Facilities in the LPZ
Millstone Nuclear Power Station

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FIGURE 2.1–9 LPZ POPULATION SECTORS DISTRIBUTION

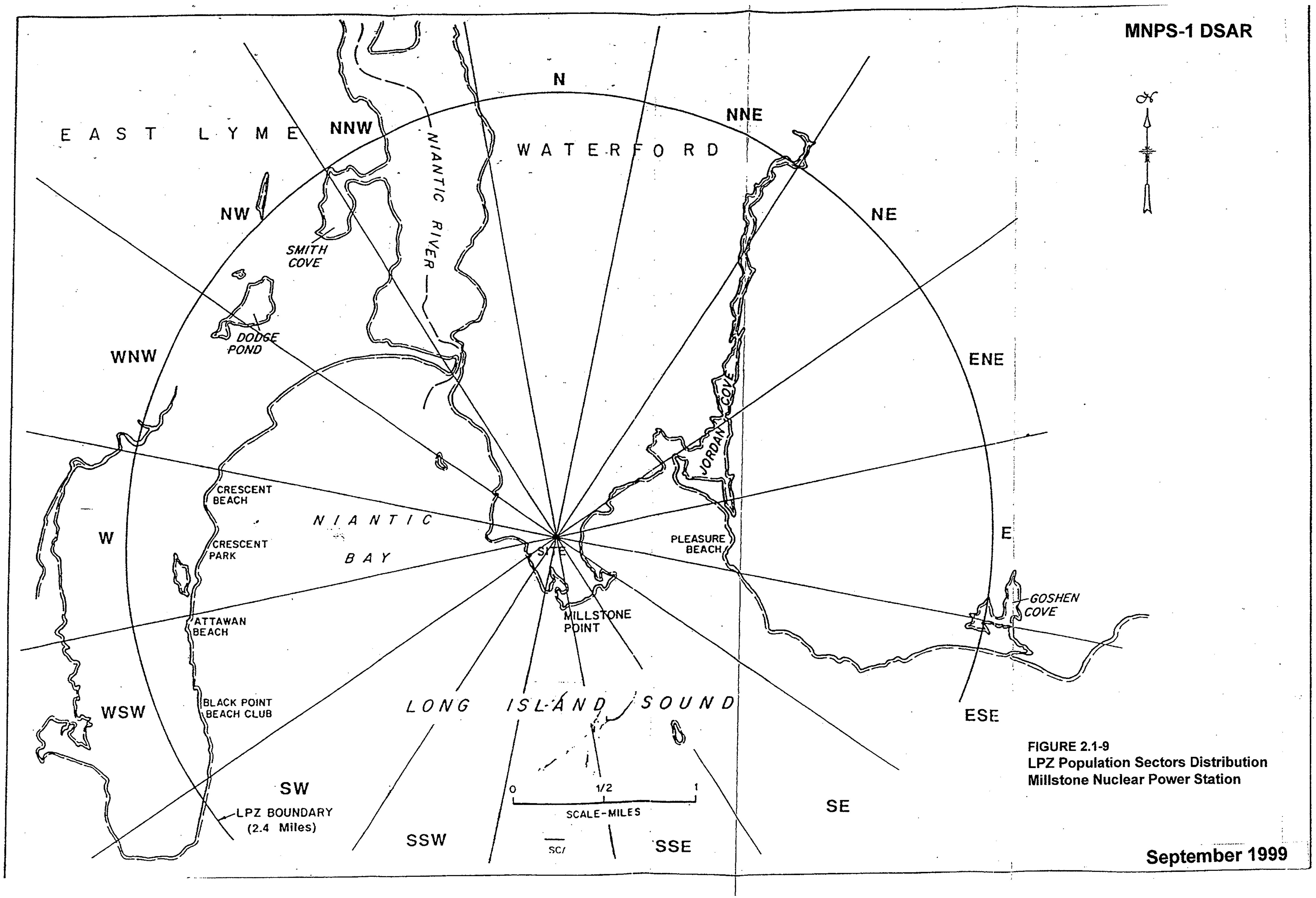
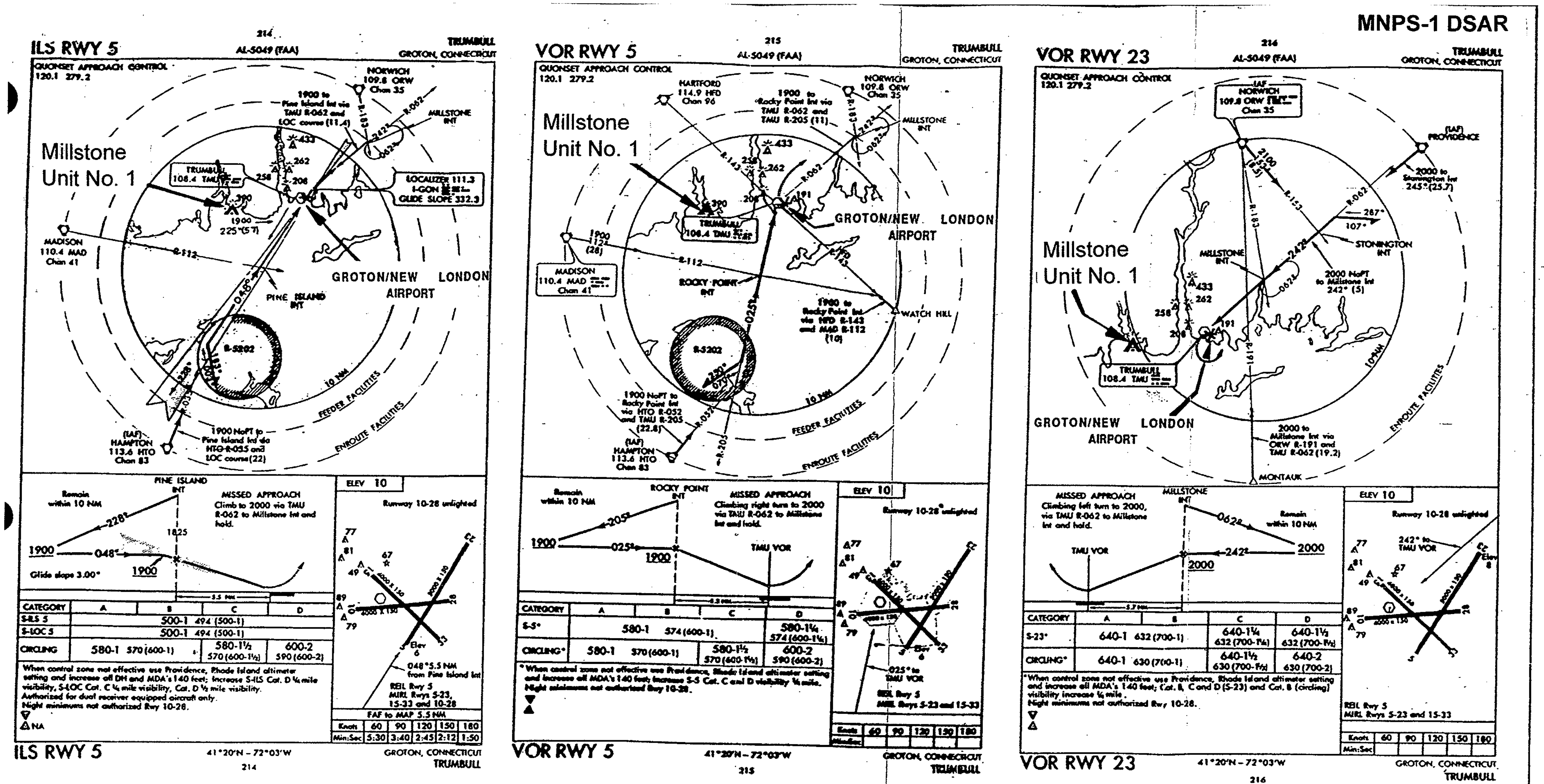


FIGURE 2.1–10 INSTRUMENT LANDING PATTERNS AT TRUMBELL AIRPORT



NOTE:
PAGES FROM DOD FLIGHT INFORMATION PUBLICATION-
LOW ALTITUDE INSTRUMENT APPROACH PROCEDURES
NORTHEAST UNITED STATES VOL-7

FIGURE 2.1-10
Instrument Landing Patterns
at Trumbull Airport
Millstone Nuclear Power Station

FIGURE 2.1–11 AIR LANES ADJACENT TO MILLSTONE POINT

MNPS-1 DSAR

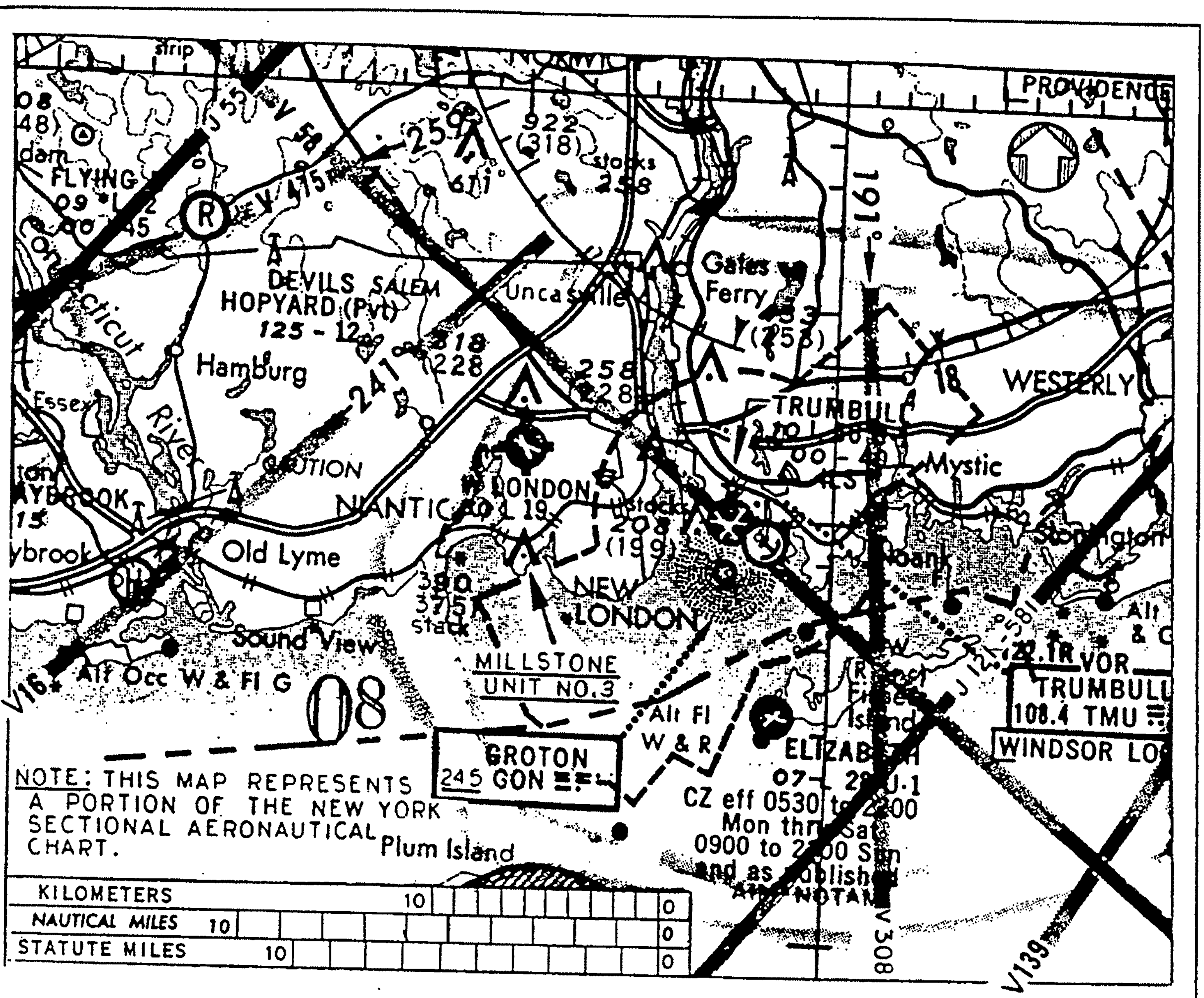


FIGURE 2.1-11
Air Lanes Adjacent to Millstone Point
Millstone Nuclear Power Station

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FIGURE 2.1–12 NEW LONDON COUNTY - STATE HIGHWAYS AND TOWN ROADS

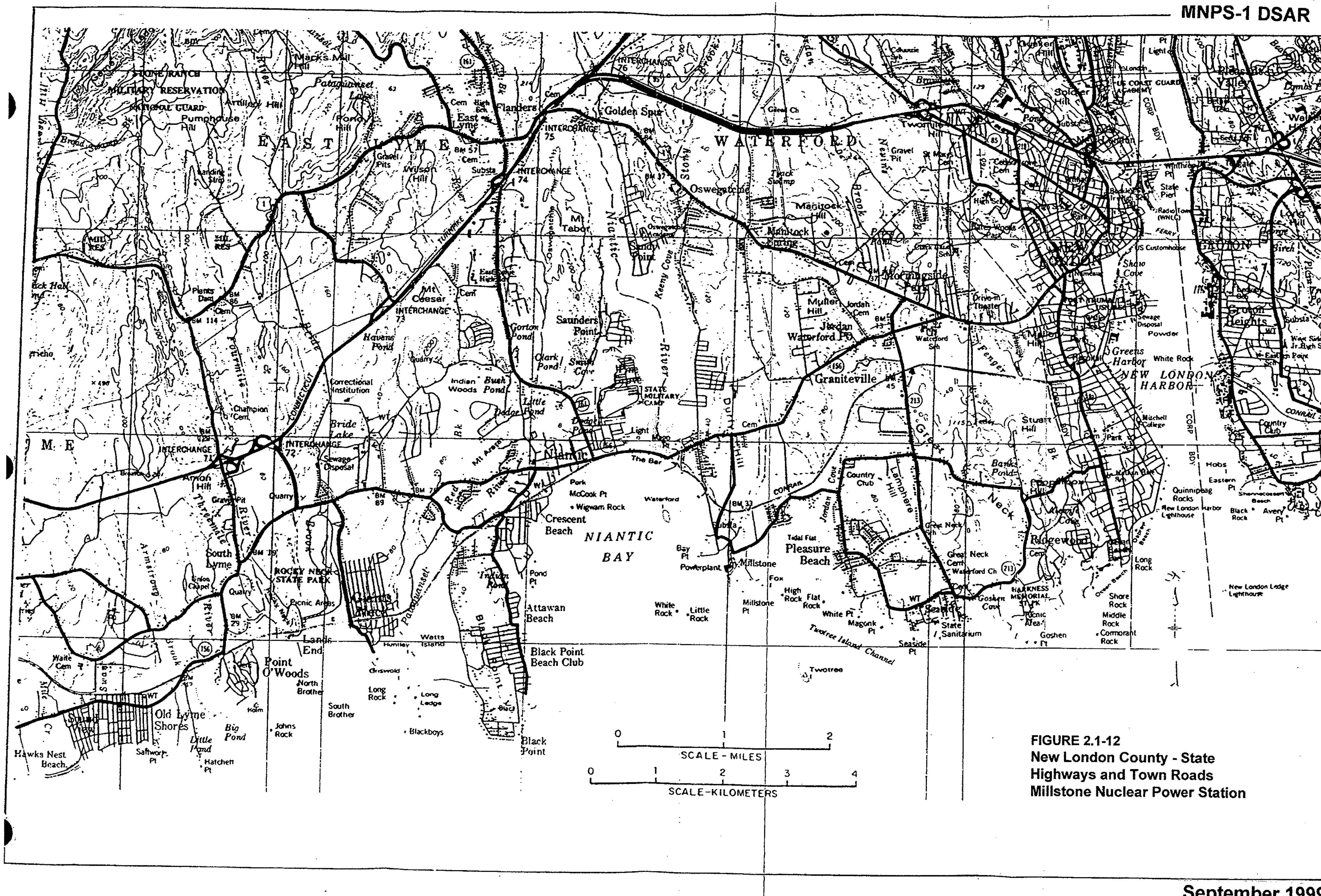

FIGURE 2.1-12
New London County - State
Highways and Town Roads
Millstone Nuclear Power Station

FIGURE 2.3–1 TOPOGRAPHY IN THE VICINITY OF MILLSTONE POINT

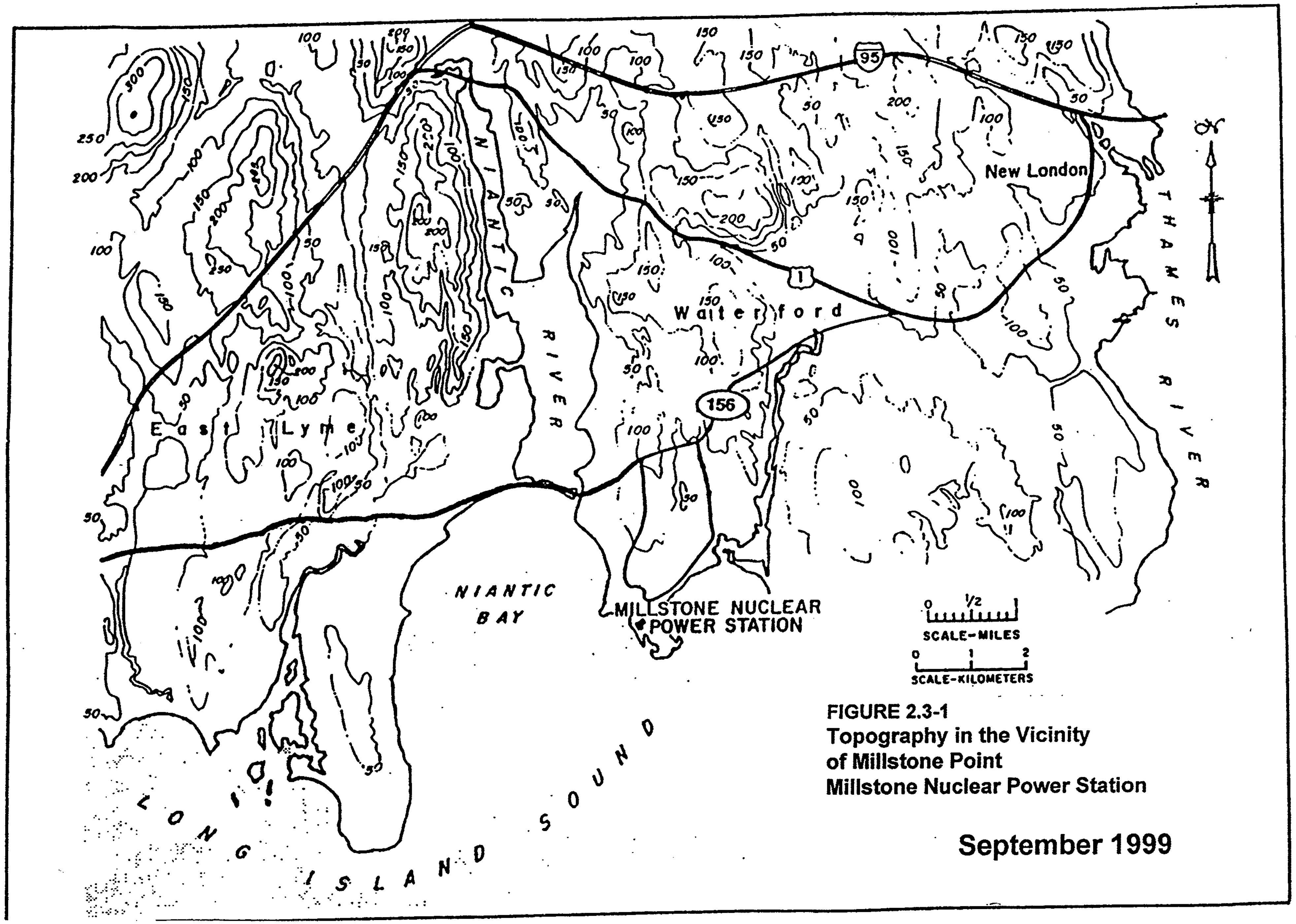

FIGURE 2.3-1
Topography in the Vicinity of Millstone Point
Millstone Nuclear Power Station

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MNPS-1 DSAR

FIGURE 3.1–1 REACTOR BUILDING SEISMIC LOADS

BUILDING WEIGHT AND SECTION PROPERTIES

Node	Elevation	Segment length	Weight / Mass	I_C (FT.^{4})	A_C (FT.^{2})	A (FT.^{2})*
1	EL. 147 FT. - 2½ IN.		1924 K			
		8FT 2½IN	60.0 K-SEC²/FT.	1,464,266	527.0	271.0
2	EL. 129 FT. - 0 IN.		2244 K			
		20FT 6IN	69.7 K-SEC²/FT.	1,464,266	527.0	271.0
3	EL. 108 FT. - 6 IN.		17,751.05 K			
		25FT 8 IN	571.6 K-SEC²/FT.	2,281,875	2842.0	1421.0
4	EL. 82 FT. - 9 IN.		14,170.3 K			
		17FT 0 IN	440.1 K-SEC²/FT.	2,856,030	3205.0	1603.1
5	EL. 65 FT. - 9 IN.		14,680.5 K			
		23FT 3IN	445.9 K-SEC²/FT.	2,355,605	2364.0	1182.0
6	EL. 42 FT. - 6 IN.		17,811.3 K			
		28FT 0 IN	553.2 K-SEC²/FT.	5,138,000	2925.0	1463.0
7	EL. 14 FT. - 6 IN.		17,421.9 K			
		14FT 6IN	541.1 K-SEC²/FT.	8,063,880	3543	1772
8	EL. 0 FT. - 0 IN.		25,344.1 K			
		26FT 0IN	787.1 K-SEC²/FT.	8,758,388	6275	3138
	EL. -26 FT. - 0 IN.					

* EFFECTIVE SHEAR AREA

FIGURE 3.1-1 Reactor Building Seismic Loads

FIGURE 3.1–2 ACCELERATION DIAGRAM UNDER SEISMIC LOADS 5 PERCENT DAMPING

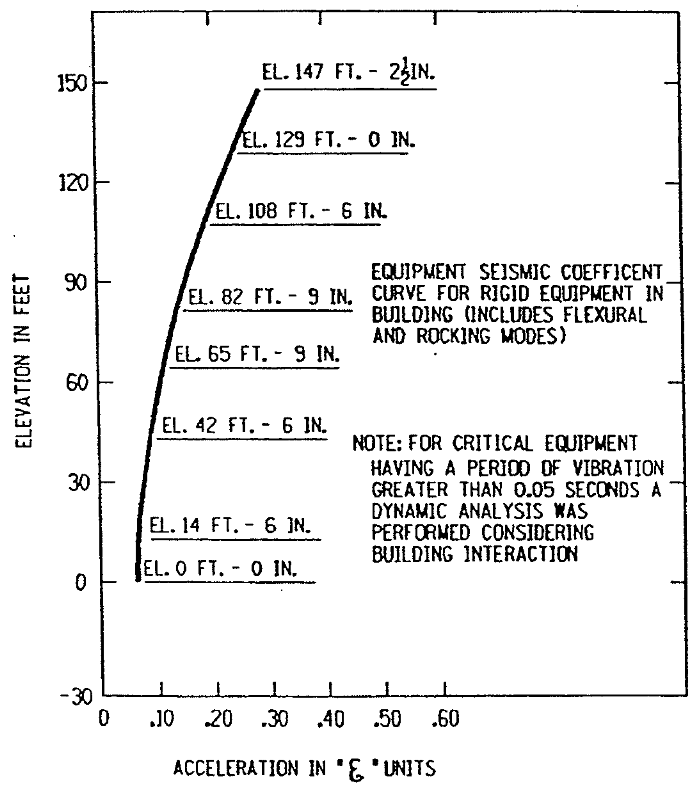

FIGURE 3.1-2 Acceleration Diagram Under Seismic Loads 5 Percent Damping

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FIGURE 3.1–3 SHEAR DIAGRAM UNDER SEISMIC LOADS

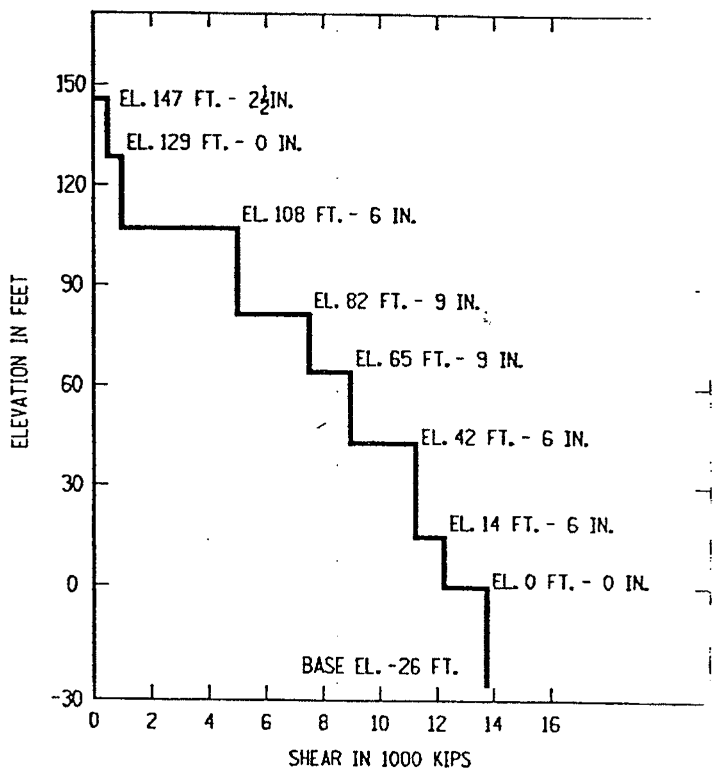

FIGURE 3.1-3 Shear Diagram Under Seismic Loads

FIGURE 3.1–4 MOMENT DIAGRAM UNDER SEISMIC LOADS

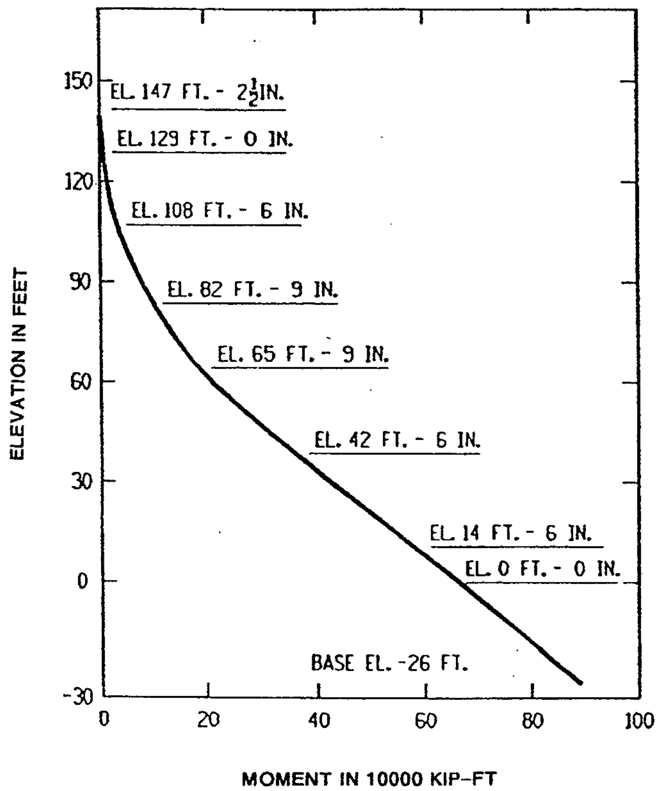

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FIGURE 3.1-4 Moment Diagram Under Seismic Loads

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FIGURE 3.1–5 DISPLACEMENT DIAGRAM UNDER SEISMIC LOADS

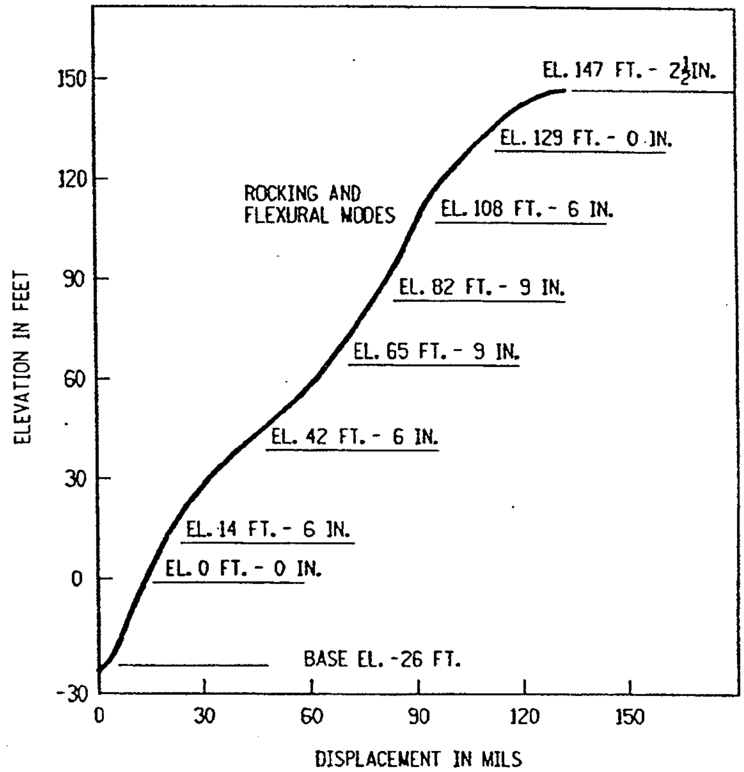
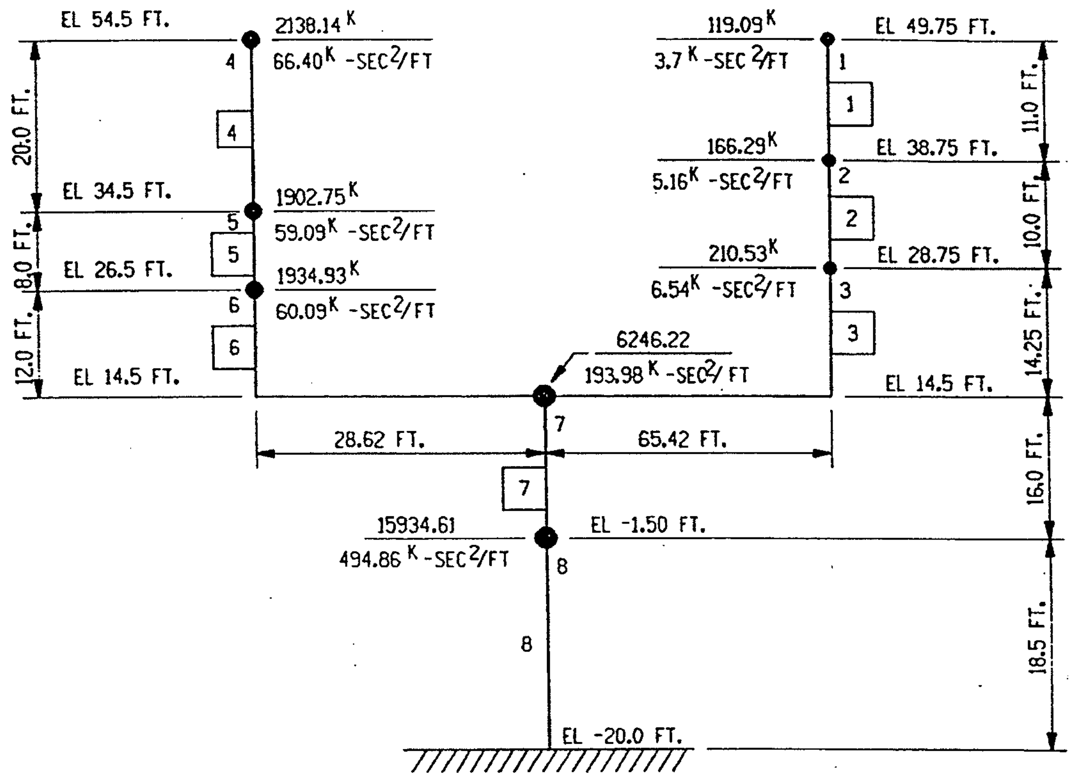
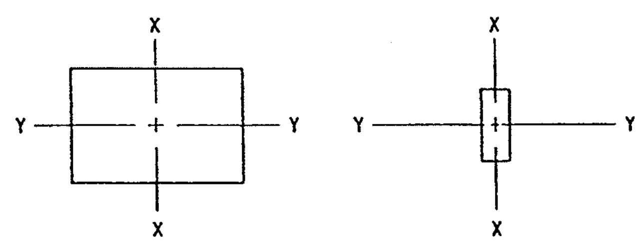


FIGURE 3.1-5 Displacement Diagram Under Seismic Loads

FIGURE 3.1-6 RADWASTE BUILDING - MATHEMATICAL MODEL





KEY PLAN

FIGURE 3.1-6 Radwaste Building - Mathematical Model

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FIGURE 3.2–1 P&ID: SFPI, FUEL POOL COOLING SYSTEM

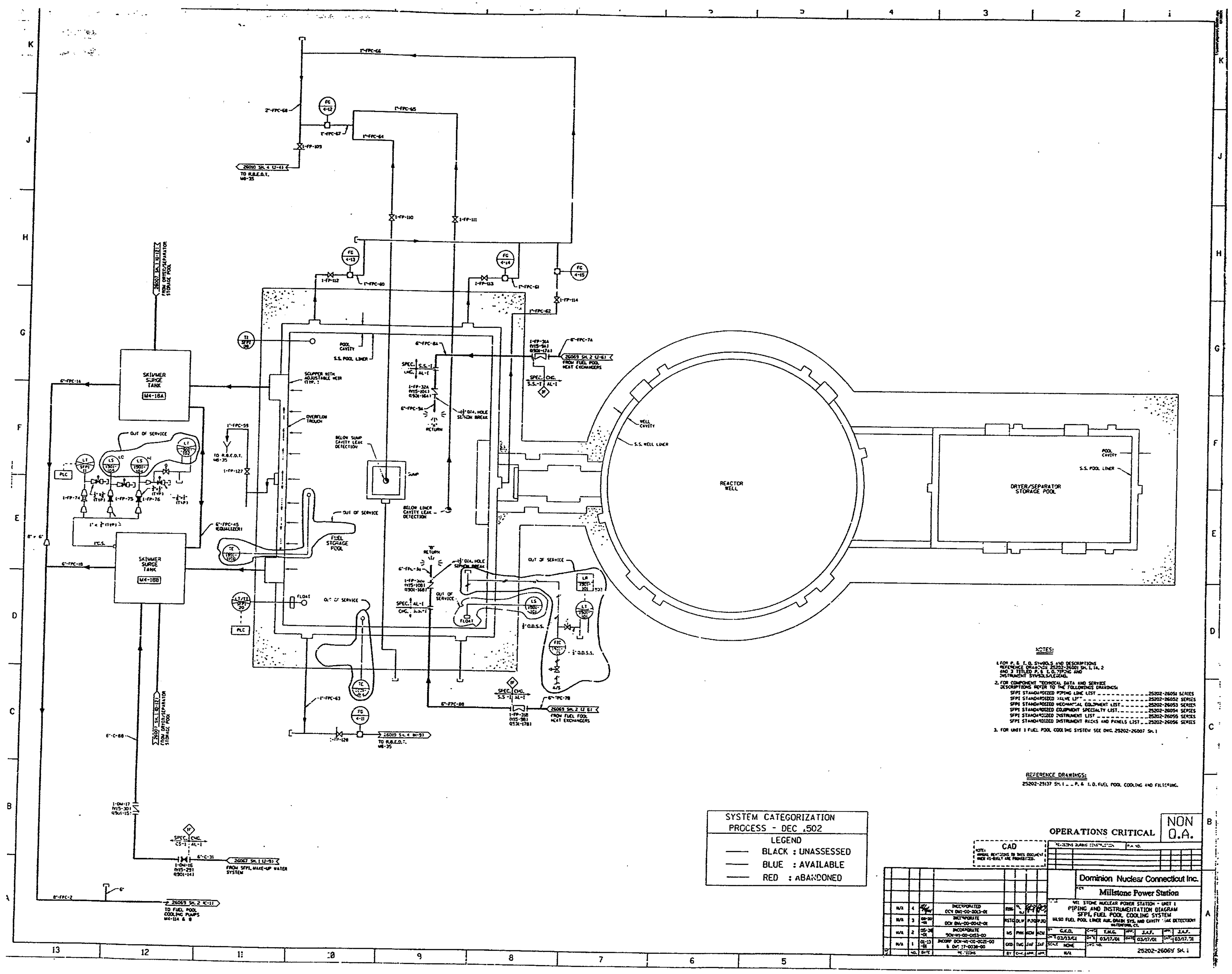

FIGURE 3.2-2 P&ID: SFPI, FUEL POOL COOLING SYSTEM

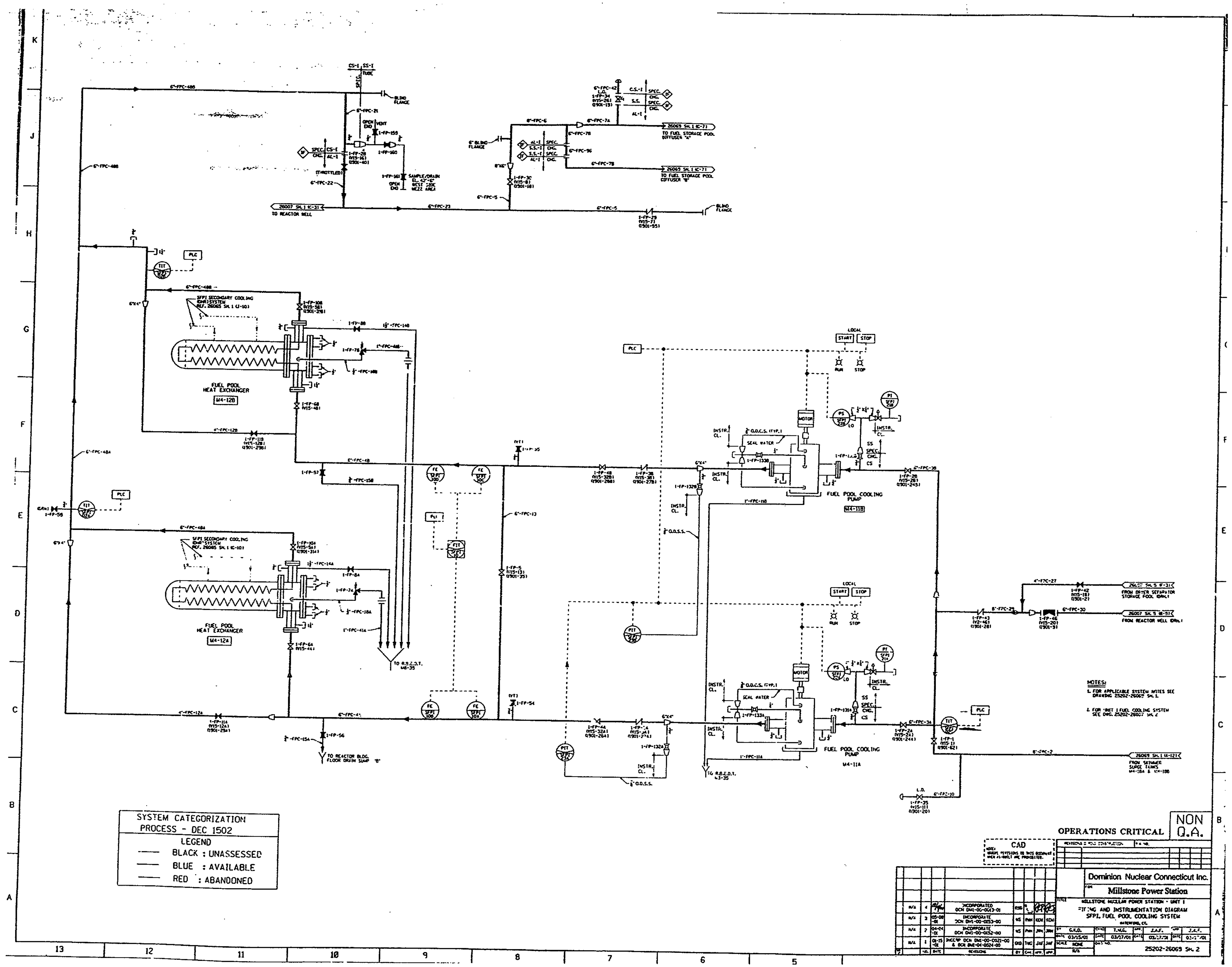

FIGURE 3.2–3 P&ID: SFPI, FUEL POOL COOLING SYSTEM

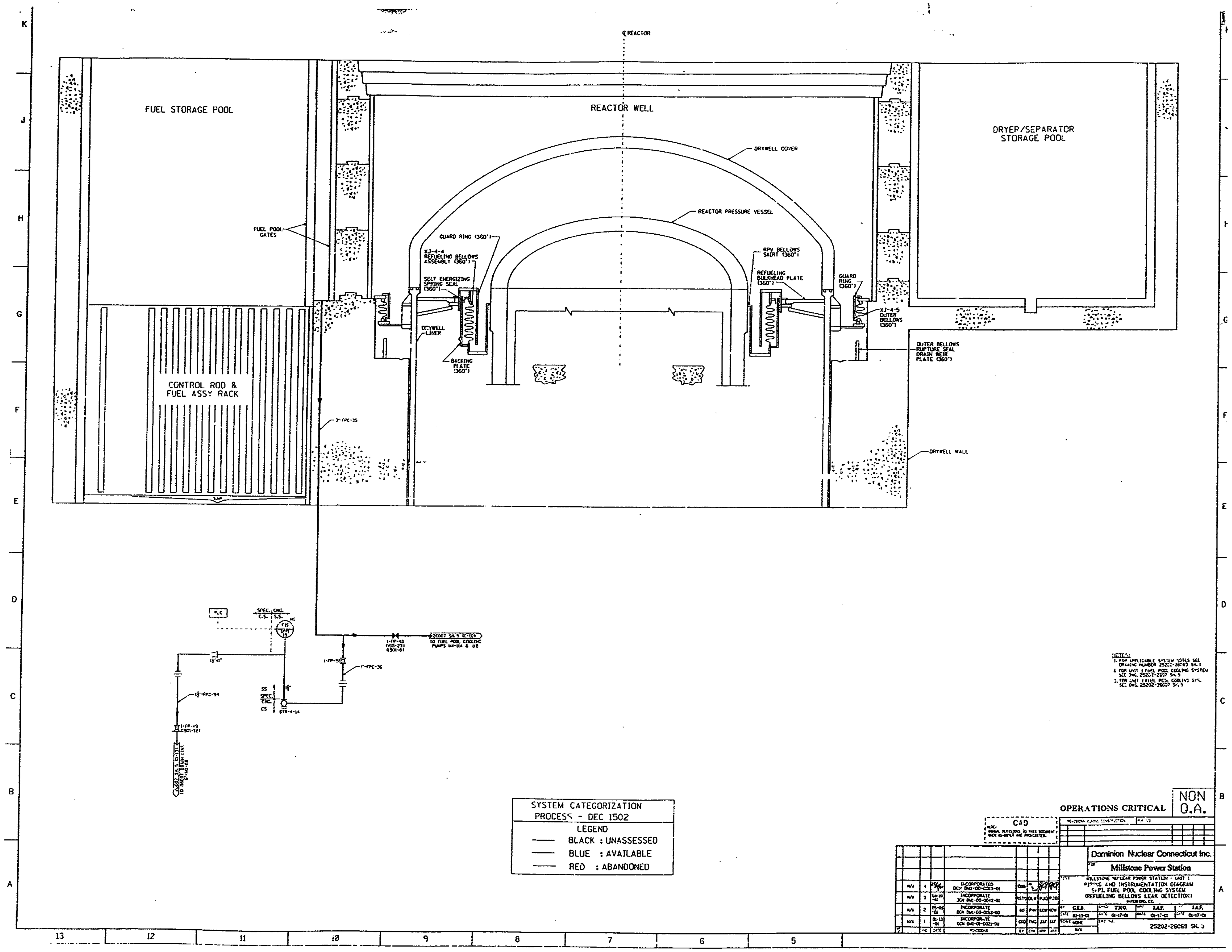

FIGURE 3.2–4 P&ID: REACTOR BUILDING AND HVAC ROOM SFPI SECONDARY COOLING (DHR) SYSTEM

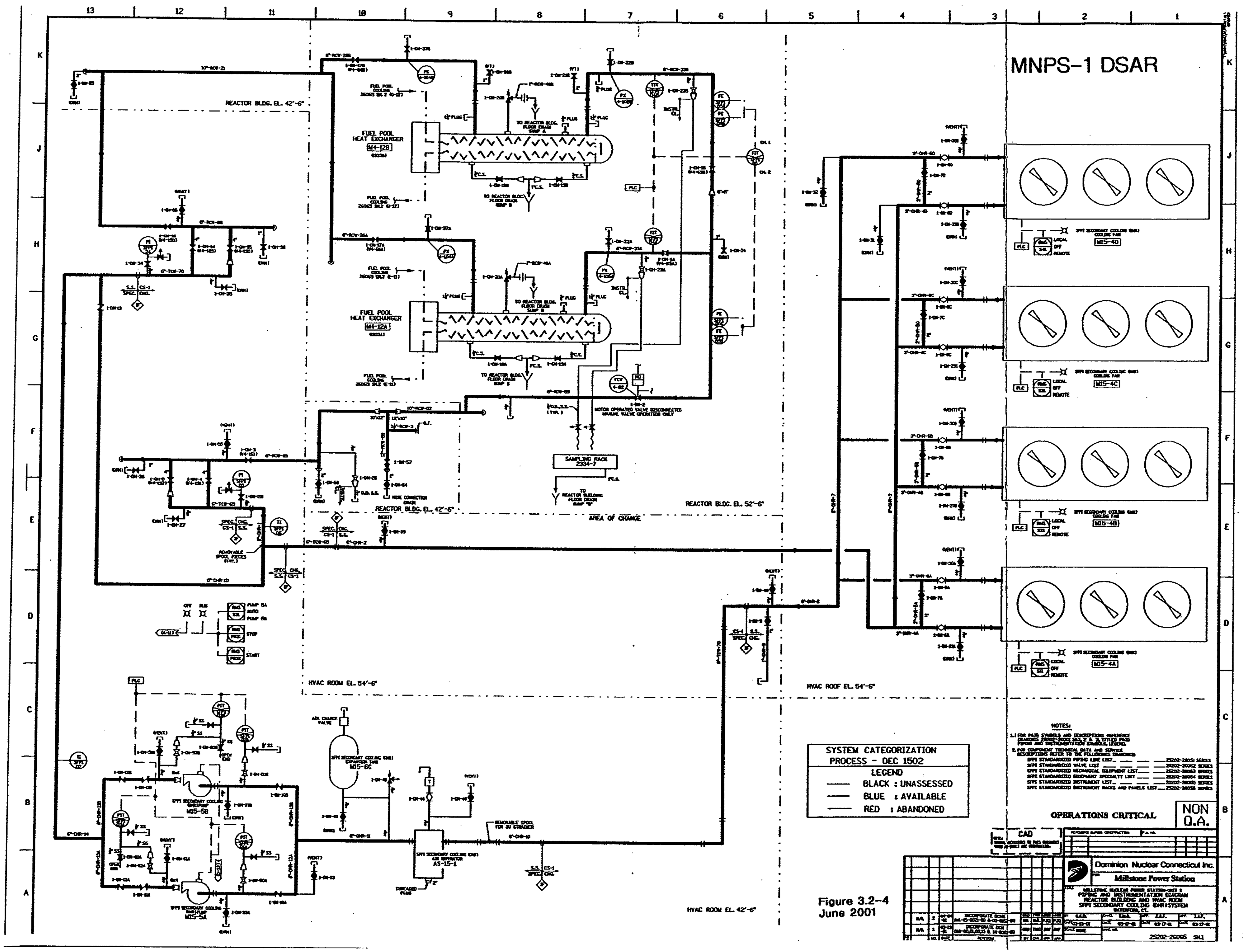

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FIGURE 3.2-5 P&ID: REACTOR BUILDING SFPI, MAKE-UP WATER SYSTEM

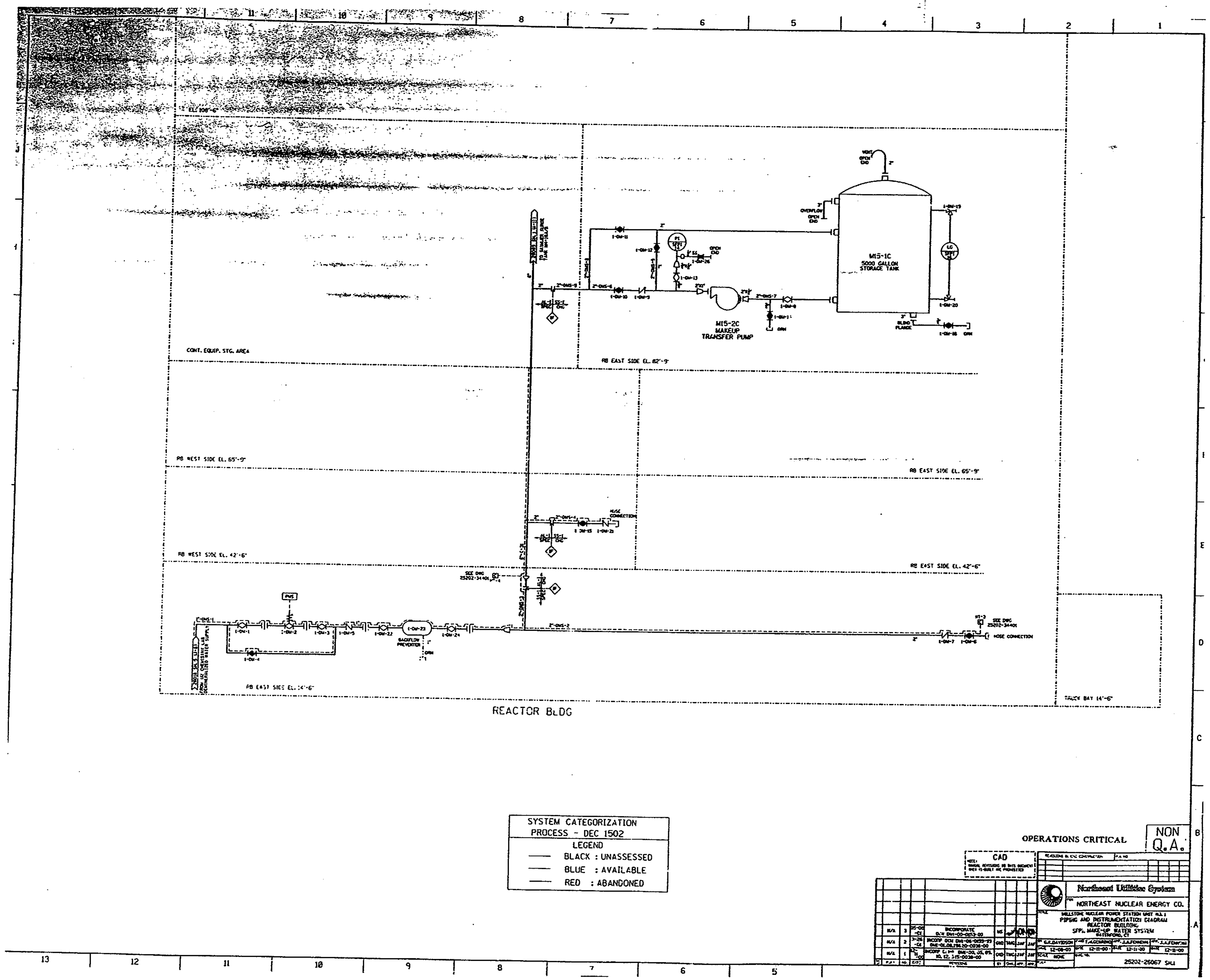

FIGURE 3.2–6 P&ID SFPI HVAC SYSTEM COMPOSITE

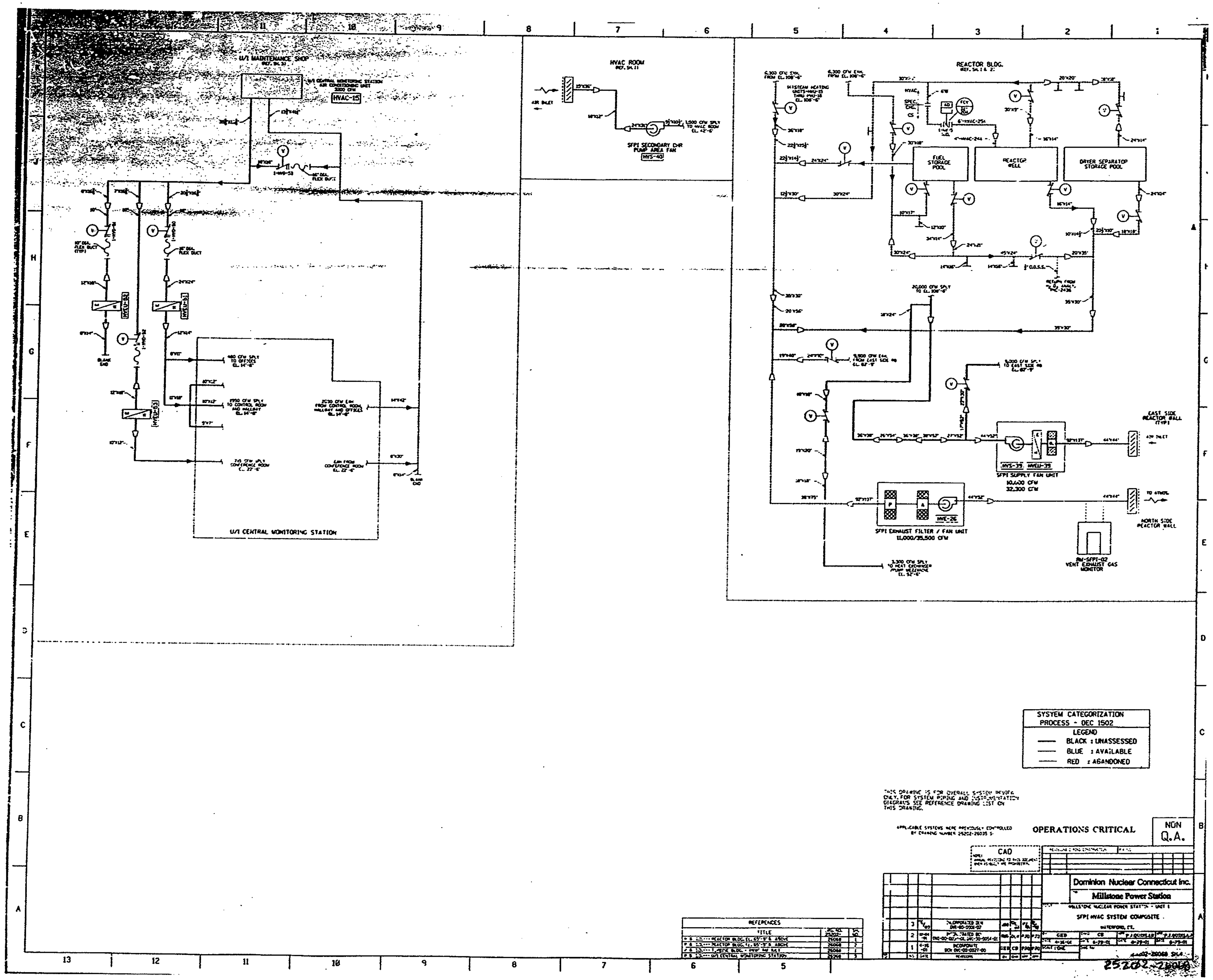

FIGURE 3.2–7 P&ID: HVAC B.O.P. SYSTEM COMPOSITE

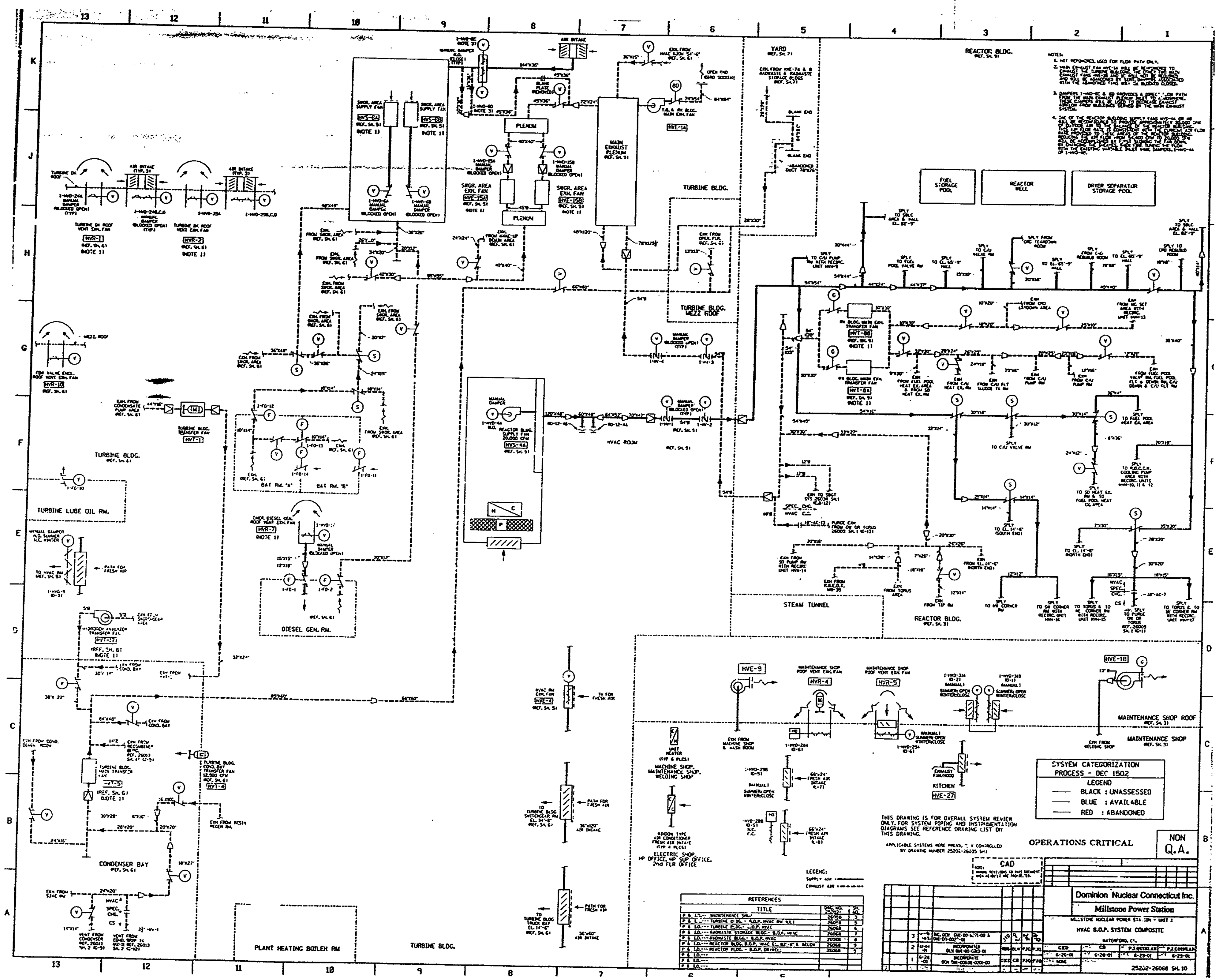

FIGURE 3.2–8 THROUGH 3.2-11 INTENTIONALLY DELETED

FIGURE 3.2–12 P&ID: HVAC BALANCE OF PLANT SYSTEM COMPOSITE

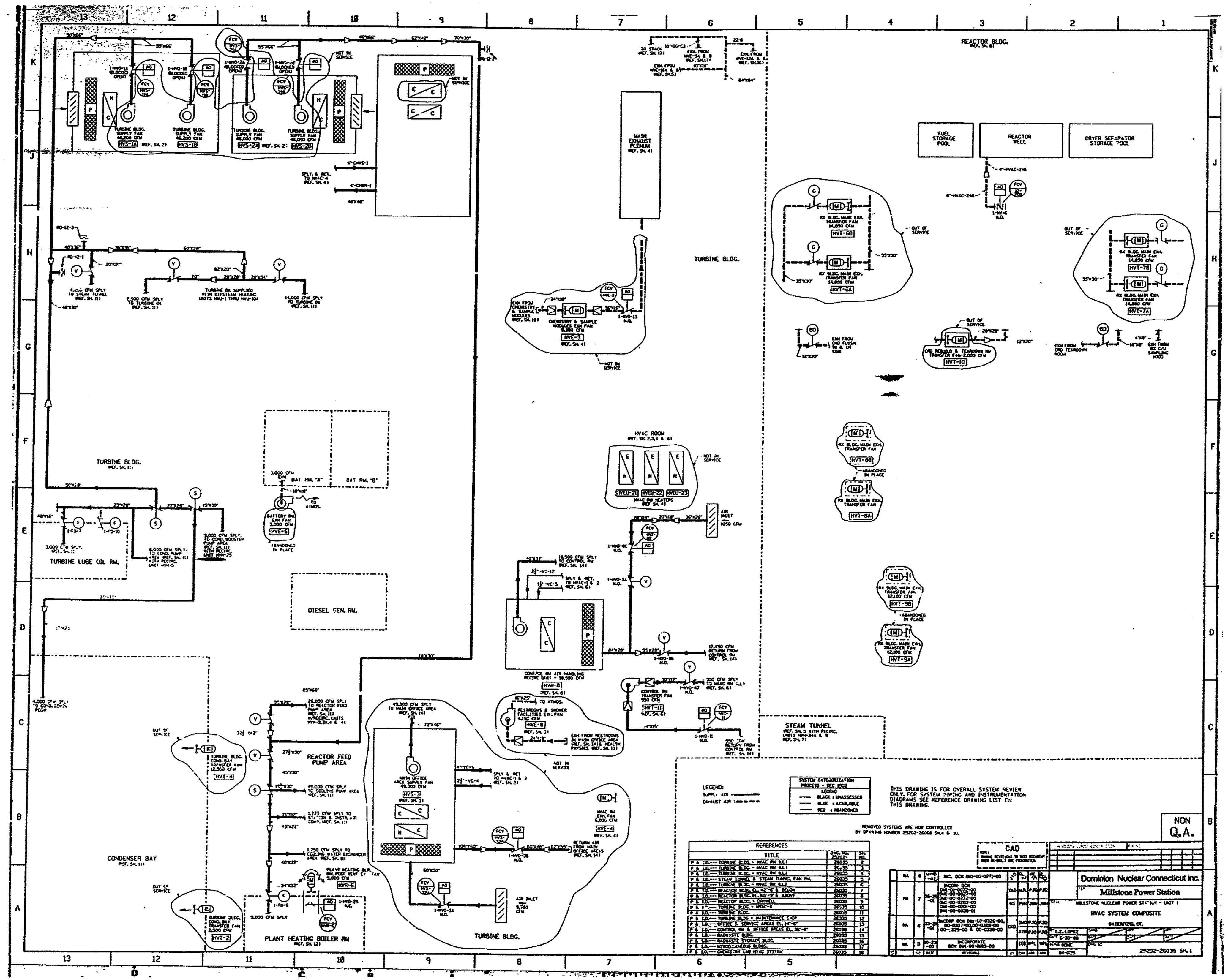

FIGURE 3.2–13 P&ID: HVAC SYSTEM (RADWASTE STORAGE BUILDING)

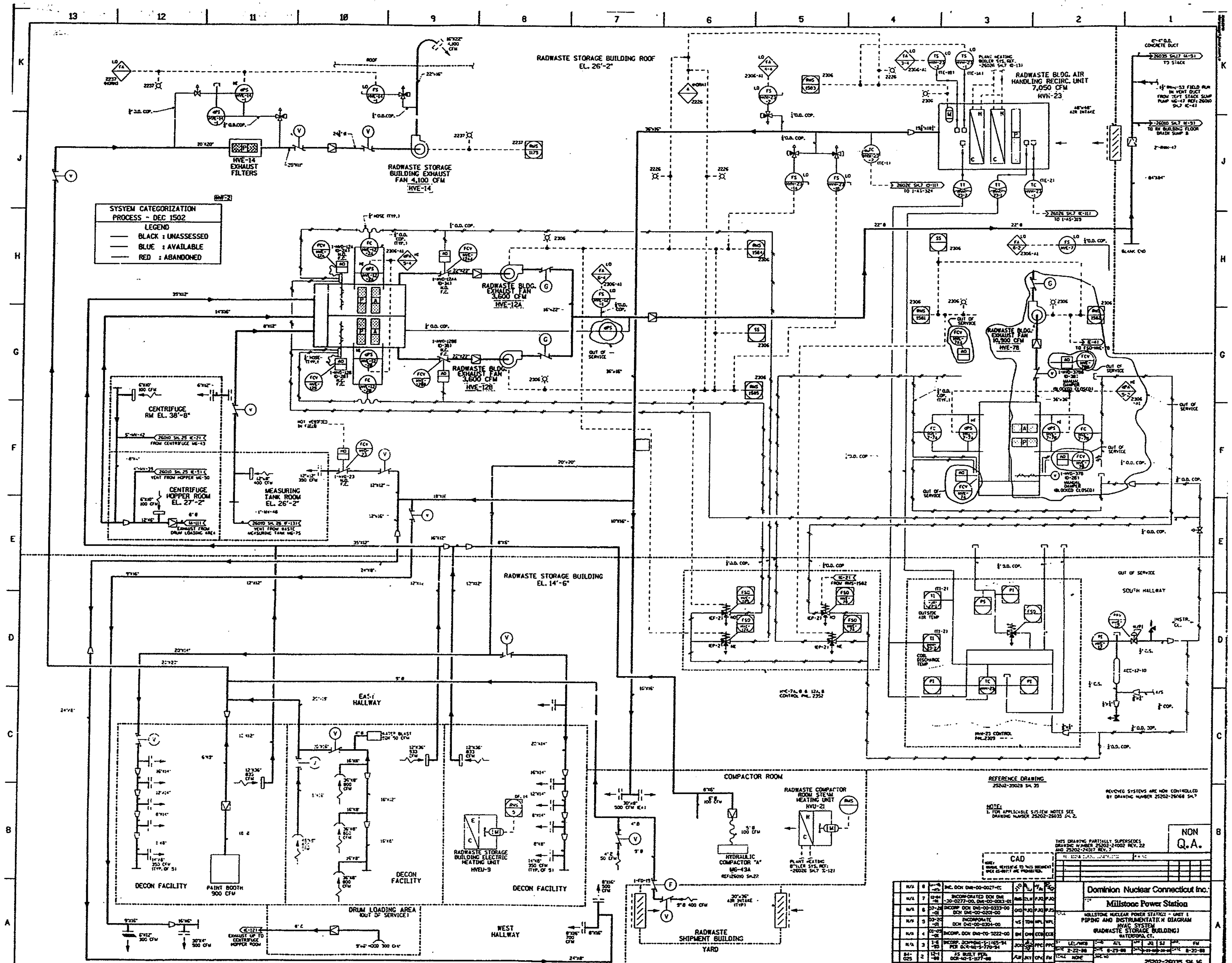

FIGURE 3.2–14 FIRE PROTECTION COMPOSITE