3.2 SYSTEMS

3.2.1 Fuel Storage And Handling

3.2.1.1 New Fuel Storage

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

a. The new fuel storage racks provide storage space for 290 fuel assemblies.

b. The new fuel storage racks are designed such that the spacing of fuel bundles in the new fuel storage vault maintains a k_{eff} of less than .90 dry and a k_{eff} less than 0.95 in the flooded condition.

c. The design of the new fuel storage racks is such that a fuel assembly cannot be inserted anywhere other than designated storage locations.

d. New fuel storage racks are designed and arranged so that the fuel assemblies can be handled efficiently.

3.2.1.1.2 Facilities Description

The new fuel storage vault is a reinforced concrete structure, accessible only through top hatches. Racks in the vault can hold a maximum of 290 fuel bundles in an upright position. The center-to-center spacing of bundles in the racks is approximately 6.6 inches by 11 inches. The racks are securely attached to guides embedded in concrete and the space around each element is filled to prevent inadvertently placing fuel in the intermediate spaces.

The storage racks in the vault are full length, top entry and designed to prevent an accidental critical array, even in the event the vault becomes flooded; however, vault drainage is provided to prevent possible water collection.

3.2.1:1.3 Safety Evaluation

The spacing of fuel bundles in the new fuel storage vault maintains $k_{eff} < 0.95$ when flooded. The vault floor drain prevents flooding. Ch. 2

The maximum k_∞ of the reload fuel stored in the new fuel storage racks is less than the maximum k_∞ of the initial fuel assemblies without temporary control curtain; therefore, this configuration is also considered safe.

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 3229 fuel assemblies.

b. Maintain a k_{eff} of less than 0.90 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, Boraflex shrinkage/gaps, seismic induced deflections and calculation uncertainty.

c. The concrete shielding walls are designed as part of the Class 1 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 on four sides, and 2184 storage cells contain B_4C plates for neutron absorption.

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 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 double sealed gates with a monitored drain between the gates. This arrangement permits detection of leaks from the passage and repair of a gate in the event of such leakage. Both gates are permanently closed and form part of the pool boundary. Ch. 3

In response to the NRC I.E. Bulletin 84-03, additional modifications have also been made to augment the leak detection capability in the spent fuel pool/refueling cavity area. They include:

(1) A spent fuel pool level indicator has been installed to indicate high/low level in the spent fuel pool;

(2) Isolation valves and seismic supports have been installed on all lines tied to the refueling cavity that exit the drywell to ensure that an uncontrolled leak path does not exist from the refueling cavity in the event of seismic event; and

(3) The flow switches on the leak detection lines have been upgraded and provide a more reliable method of detecting flow from the refueling cavity.

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

The storage pool is designed to hold 20 fuel channels. A rack is provided in the storage pool for storing up to 19 control rod blades. Ch. 1

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

3.2.1.2.3 Safety Evaluation

The spacing of fuel bundles in the spent fuel storage pool, the presence of neutron absorbing poisons in the fuel storage racks, and the design of the fuel bundles maintains K_{eff} less than or equal to 0.90. 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 down to a spent fuel pool temperature of 68°F.

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 switch and 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 main 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, 3.2-7 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 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. This is significantly longer than required to reinstate external cooling of the water.

Multiple methods are available to cool the pool water and adequate time is available to repair, manually reinstate or line up the system used for pool water cooling. Most significantly, even if none of these systems are 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. 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.

b. To provide high-clarity water to the pools.

c. To remove radioactivity released to the pool water.

3.2.1.3.2 Spent Fuel Pool Heat Load

Millstone Unit No. 1 has permanently ceased power operation and that all irradiated fuel had 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 is 1.781 MBtu/hr. This spent fuel pool heat load is well within the heat capacity of the spent fuel pool cooling system.

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 10 days starting from an initial pool temperature of 100°F. This calculation also establishes a maximum evaporative loss of 3.8 gpm under the above conditions.

3.2.1.3.4 System Description

The spent fuel pool cooling system cools and purifies water in the fuel pool on an as needed basis to maintain water temperature and 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 3.2-5.

The dryer-separator pit is provided to accommodate the steam dryer and shroud head/steam separators.

The bulk temperature monitoring system consists of two thermocouples installed in thermowells with the sensing element located approximately 6' below the normal water level of the fuel storage pool. A recorder located in the Control Room provides both indication of bulk temperature and notification of a high water temperature condition within the fuel storage pool.

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

The fuel pool cooling system is controlled and operated from the control room. The system is provided with indicators and alarms for system flow, water level, and temperature, skimmer surge tank level, and valve position.

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 No. 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. This is significantly longer than required to reinstate external cooling of the water.

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 such as cutting and sawing up of flux monitor tubes and control rods will be possible at any time.

3.2.1.4.2 System Description

The fuel handling system handles both new and irradiated fuel.

A refueling platform, equipped with a refueling grapple and two 1/2-ton auxiliary hoists is provided. Either of these hoists can be positioned for servicing the reactor head cavity or 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 Service Water System

3.2.2.1 Design Bases

The service water system provides strained seawater for equipment cooling. The system design bases are:

Design Temperature:	75° F
Design Flow Rate:	10,000 gpm per pump
Design Pressure:	100 psig

The purposes of the service water system are to supply cooling water to various systems and components, and sealing water to Service Water pumps as required.

The service water system is in operation as required by plant activities during the permanently defueled condition. The service water system is chlorinated by the sodium hypochlorite system.

One or more service water pumps may be in service to provide service water system flow.

3.2.2.2 System Description

Service water pumps provide seawater for cooling the Reactor Building Closed Cooling Water System (RBCCW). The service water pumps are vertical centrifugal pumps each capable of delivering 10,000 gpm.

The service water system is designed to remove heat from RBCCW System. Design service water flow rate to RBCCW is 12,000 gpm with a design heat transfer rate of 78 million Btu/hr.

The RBCCW heat exchangers have sufficient cooling capacity such that system operation is not hampered should repairs be required on one piece of the equipment.

Temperature and pressure indicators located near the heat exchangers indicate a malfunction in the system.

The flow diagram for the service water system is shown in Figures 3.2-6 and 3.2.7.

3.2.2.3 Safety Evaluation

The Service Water systems supplies cooling water to the RBCCW heat exchangers and provides dilution for the liquid radwaste discharged to the quarry. These functions support the fuel pool cooling and radioactive waste management in the permanently defueled condition, but are not safety related.

3.2.2.4 Instrumentation

Pump control switches and pump motor current indicators are located in the Control Room.

Service water related alarms are located in the Control Room. These alarms include liquid process high radiation. This alarm is fed from the radiation monitor on the service water outlet line from the RBCCW heat exchangers.

3.2.2.5 Service Water Chlorination System

The service water system is chlorinated via injection of a sodium hypochlorite solution. This capability provides chlorination of the service water and thereby controls biological fouling in the system.

3.2.3 Reactor Building Closed Cooling Water System

3.2.3.1 Design Bases

The RBCCW system is designed to provide cooling water in the Reactor Building. The system is also designed to minimize the release of radioactive material to the service water system. The system design bases are:

Design Temperature:	85° F
Design Flow Rate (maximum):	4200 gpm per pump
Design Pressure:	150 psig

The RBCCW system is normally in service to supply 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 RBCCW system provides a supply of cooling water to equipment in the Reactor Building. Water is circulated in a closed loop by the required RBCCW pumps. Heat is removed from the system by the RBCCW heat exchangers. System configuration may vary depending on heat load. The remainder of the system consists of a cooling water surge tank, a chemical feeder, piping and valves, and controls and instrumentation. The flow diagram for the system is shown in Figure 3.2-8.

Heat is removed from the system by Service Water on the tube side of the heat exchangers.

A surge tank is located above the highest point in the system to handle system fluctuations and to supply make-up water to the system when necessary. Demineralized water is available for make-up requirements.

A radiation monitor is located on the RBCCW heat exchanger return header and is connected to a recorder in the Control Room.

3.2.3.3 Safety Evaluation

The Reactor Building Closed Cooling Water 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 RBCCW system is not safety related.

3.2.3.4 Testing and Inspection

The radiation detector is tested periodically to ensure equipment calibration.

3.2.3.5 Instrumentation

RBCCW instrumentation and controls are located in the Control Room. A control switch for fuel pool heat exchanger outlet valve is located on the fuel pool pump control panel.

3.2.4 Makeup Water System

3.2.4.1 Demineralized Water

3.2.4.1.1 System Description

Ecolochem can provide makeup demineralized water to the Millstone Unit No. 1 demineralized water storage tank.

The effluent is stored in the Unit No. 1 50,000-gallon demineralized water storage tank until required. The makeup water is then transferred through the demineralized water transfer pumps for appropriate uses.

The piping, tanks and other equipment of the demineralized water transfer system are of corrosion-resistant metals or with coatings which prevent contamination of the makeup water with foreign material.

The demineralized water transfer system consists of two full capacity pumps so that the failure of one pump does not shut down the transfer system. The flow diagram for the system is shown in Figure 3.2-9.

3.2.4.1.2 Safety Evaluation

The demineralized water provides demineralized makeup water to the condensate storage tank, closed cooling water systems (RBCCW and TBSCCW), and for various other purposes. The Demineralized Water system also serves as a backup source of makeup water for the spent fuel pool. These functions support fuel pool cooling, but are 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 demineralized water storage tanks. The equipment is visually inspected periodically. Sampling of the demineralized water storage tank is a standard monitoring procedure.

3.2.4.1.4 Instrumentation

Motor control switches for the demineralized water transfer pumps are located in the Control Room. Also provided in the Control Room are level indications for the demineralized water storage tank.

Separate temperature controllers located on the demineralized water storage control temperature control valves to maintain the water at an elevated temperature.

3.2.4.2 Condensate Storage and Transfer System

3.2.4.2.1 Design Bases

The function of the condensate storage and transfer system is to provide condensate to unit systems.

The original design bases for the system are:

Design Temperature:	40 to 100°F
Design Pressure:	150 psi
Design Storage Capacity:	460,000 gallons

3.2.4.2.2 System Description

The condensate storage and transfer system consists of one 460,000 gallon storage tank, two condensate transfer pumps, and control and support equipment.

The transfer pumps are sized to meet condensate requirements throughout the plant.

The system also provides the principal source for makeup to the spent fuel pool. Other small requirements are also supplied. The flow diagram for the system is shown in Figure 3.2-9.

3.2.4.2.3 Safety Evaluation

The Condensate Storage and Transfer system provides makeup water to the spent fuel pool and to radioactive waste management system components. These functions support fuel pool cooling and radioactive waste handling in the permanently defueled condition, but are not safety related.

3.2.4.2.4 Testing and Inspection

Standard pre-operational tests were performed after installation. Since the system is in frequent use, testing is not required.

3.2.4.2.5 Instrumentation

Motor control switches for the condensate transfer pumps are located in the Control Room. Also provided in the Control Room are level indications for the condensate storage tank.

Separate temperature controllers located on the condensate storage tank control temperature control valves to maintain the water at an elevated temperature.

3.2.5 Instrument Air System

3.2.5.1 Design Bases

The purpose of the instrument air system is to provide clean, moisture-free, compressed air to air-operated control devices and instrumentation.

The system design bases are:

Design Pressure:	125 psig with a nominal pressure of 100 psig
Design Temperature:	120°F
Compressor Design Capacity:	663 SCFM at a nominal pressure of 100 psig
Receiver Capacity:	500 cubic feet
Air Dryer Design Capacity:	665 SCFM at a nominal pressure of 100 psig
Air Dew Point After Dryer:	-40°F

Design temperatures are obtained from extreme ambient conditions. The instrument air dew point is selected to be under the lowest outdoor temperatures expected at the plant location.

3.2.5.2 System Description

The instrument air system incorporates two full-capacity, non-lubricated, water cooled compressors, each having a separate inlet filter, aftercooler, moisture separator, and closed cooling water system. The two air compressors normally discharge to their respective full capacity air dryer and filter assemblies, however cross-tie capabilities exist. The air dryers discharge to a common header connected to an instrument air receiver. The instrument air receiver supplies the instrument air header which divides into branch lines supplying areas of the plant requiring instrument air. The function of the system is to supply instrument and control air for the entire unit on a continuous basis.

Normally, one instrument air compressor operates continuously on an automatic load-unload basis and the other instrument air compressor is on automatic standby.

The flow diagram for the system is shown in Figure 3.2-10.

3.2.5.3 Safety Evaluation

All services essential to operation of plant equipment requiring air during the permanently defueled condition, are supplied from instrument air headers in the plant. The instrument air compressors do not provide any essential safety function in the event of a design basis accident.

3.2.5.4 Testing and Inspection Requirements

Standard pre-operational tests were performed after installation. Since the system is in frequent use, periodic testing is not performed.

3.2.5.5 Instrumentation

Compressor remote control switches are located in the Main Control Room. The compressors may also be started and stopped from local control switches.

The instrument air dryers are controlled locally through a control panel located at each dryer.

Local mounted flow gauges are incorporated in the instrument air system to monitor flow (SCFM) at each instrument air compressor discharge and the instrument air receiver discharge.

A local mounted moisture monitor is installed at the instrument air receiver outlet for dew point indication and alarm.

A local mounted carbon monoxide monitor is installed at the outlet of the instrument air receiver for CO indication and alarm.

3.2.6 Process Sampling System

3.2.6.1 Design Bases

The reason for sampling process fluids and 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

Fluids and gases are sampled continuously and periodically from equipment and systems to detect heat exchanger leaks.

Sampling of stack discharge gases is by collection on particulate and charcoal filters. Stack discharges are monitored for radiation activity.

Stack gas is continuously sampled and monitored for radiation, halogens and particulates. The sampling is taken at about 10 stack diameters above the upper mixing plenum. Four sample probes, across the width of the stack, sample the stream of the effluent to obtain a well-mixed sample of off-gases and ventilation air. The sample is taken with an isokinetic sampler and passed through a particulate filter and a charcoal filter; the sample is then monitored and expelled again to the stack with the aid of a vacuum pump. A sample flow rate of approximately 2 cfm is used to obtain stack gas samples. The monitoring is accomplished with two scintillation counters looking at a continuously flowing gas sample downstream from the filters.

Grab samples are taken periodically from the ventilation exhaust ducts of the Radwaste Building and the Reactor Building and are analyzed for radioactive content.

3.2.6.3 Safety Evaluation

The radiation level is monitored in the RBCCW and the service water systems, to provide an indication of heat exchanger tube leaks and to support the goal of ensuring that radioactivity in effluents released from the site does not exceed applicable limits. Since the RBCCW and service water operate to support the spent fuel pool cooling function, the associated process radiation monitors are required to support the permanently defueled condition, but are not safety related.

3.2.6.4 Testing and Inspection

Standard pre-operational tests were performed after installation. Routine use substitutes for subsequent periodic testing, with the exception of standard calibration and maintenance.

3.2.7 <u>Electrical Systems</u>

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 station electrical system has a preferred and standby power supply. The system is designed to provide reliable power for the permanently defueled condition. Individual power systems are designed with sufficient sources, relay protection, control, and necessary switching.

3.2.7.2 Off-Site Source

The four transmission line circuits terminated at the 345 kV switchyard are:

- Millstone to Card (Line No. 383)
- Millstone to Montville (Line No. 371)
- Millstone to Southington (Line No. 348)
- Millstone to Manchester (Line No. 310)

These circuits connect the station to the system transmission grid and follow a common right-of-way.

The off-site power source is from the 345 kV switchyard through the reserve station service transformer.

The alternate off-site source is through the emergency station service transformer (ESST), which steps down a 23 kV source from the Flanders Substation 11Y2 circuit to 4160V.

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

3.2.7.3 345 kV Switchyard System at Site

The 345 kV switchyard is designed in a breaker and a half switching arrangement as shown on Figure 3.2-11.

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.

Normally AC power is provided through the reserve station services transformer. The reserve station services transformer has adequate capacity to supply all normal auxiliaries required to support the permanently defueled condition.

The 125 V DC power system consists of two independent on-site sources of DC power.

The 120 V AC bus system consists of two independent buses. The 120 V AC vital bus provides an independent source of highly reliable and stable 120 V AC power. The instrument AC bus provides a reliable source of power to instrument racks and various 120V AC loads throughout the plant.

3.2.7.4.2 4160 Volt System

The reserve station service transformer (15G-1S) steps down 24 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 reserve station service transformer.

The 23 kV system, via the emergency station service transformer, provides a possible source of power.

The circuit breakers located on the 4160 V buses may be operated locally at the switchgear or remotely from the Main Control Room. Breakers will trip automatically when over-current conditions exist. The control power to all 4160 volt bus circuit breakers is from the 125 volt DC station batteries.

The major components of the 4160 volt power system are described below.

(1) Reserve Station Service Transformer

The reserve station service transformer is an outdoor, 345,000-4160 volt, three phase, 900 kV BIL, 60 Hz, 25.9/34.5/43.1 MVA, OA/FA/FOA 65°C rise transformer.

(2) 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 buses 14C, 14D, 14E, and 14F is stepped down through transformers, energizing the 480 volt buses 12C, 12D, 12E, and 12F respectively.

A tie breaker can tie buses 12C and 12D together and have both buses energized at the same time from one or the other of their respective transformers. There is a similar tie breaker between buses 12E and 12F.

Some of the circuit breakers located on the 480 volt buses can be opened or closed electrically from the Main Control Room to control manually operated shutdown loads. All breakers will trip automatically when overload conditions exist.

3.2.7.4.4 120 Volt Systems

The vital and instrument AC systems provide 120V, 60 Hz AC power to various electrical instruments throughout the plant.

Instrument Systems

The instrument AC system is designed to supply a reliable source of 120V, 60 Hz power to instruments and controls throughout the plant. The instrument AC bus receives its supply from one of two voltage regulating transformers. Transformer IRP-1 supplies the normal feed to the instrument bus. Transformer IV-1 provides a backup supply to the instrument AC bus and the vital AC bus. An automatic transfer switch aligns instrument AC to IV-1 in the event of a loss of the IRP-1 supply.

Vital AC System

The vital AC system is required to supply a reliable and source of 120 V AC, 60 Hz power to instruments, controls and sensors.

The vital AC bus is designed to receive its AC power from either of two sources of supply, an MG set or transformer IV-1. One supply is considered normal (MG set) and the other an emergency (IV-1) or "backup" 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

The 125 V DC electrical system consists of two separate subsystems. These subsystems are powered by two types of DC sources, batteries and battery chargers.

3.2.7.4.7 24V DC System

The 24V DC power system provides a reliable source of isolated energy for the process radiation monitoring and other instrumentation. The system is designed to have a minimum of electrical noise and be highly reliable so that the loads needed most have the highest probability of being served. The batteries are maintained fully charged at all times except during loss of plant AC power, battery charger failure, or testing.

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 No. 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 Heating and Ventilation System

3.2.8.1.1 Design Bases

The reactor building heating and ventilation system is operated to maintain a temperature above freezing within the areas of that building.

The system is also used to 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.

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.

Main supply and exhaust fans are provided with a spare unit so that full design flow capacity is available upon failure of any single unit. A unit consists of fan, motor, and their associated controls.

3.2.8.1.2 System Description

The Reactor Building HVAC system is provided 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, recirculation, and exhaust ventilation to ensure proper air flow direction and remove heat generated from equipment.

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

The supply segment of the system provides fresh air to all levels in the Reactor Building. Outside air passes through fixed louvers, a motor-operated damper, filters, and steam heating coils. Two 100% capacity fans are available to deliver air flow. Unit heaters are provided to maintain desired air temperatures on the refueling floor level.

Exhaust transfer fans draw air flow from the enclosed areas in the Reactor Building. At the operating level, exhaust air passes through slots in the sides of the fuel storage pool and dryer/ separator pit, and through exhaust lines from the reactor well. Air flow combines downstream of the transfer fans 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 and isolation valves, supply outlets, return and exhaust intakes, heating coils, exhaust hoods and enclosures, and instrumentation and controls. Control actuation, indication, and alarm instrumentation are incorporated in a central HVAC master control panel.

3.2.8.1.3 Safety Evaluation

The Reactor Building heating and ventilation system maintains 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. The Reactor Building heating and ventilation system are required in the permanently defueled condition because it houses SSCs that are associated with the safe storage and handling of irradiated fuel or radioactive waste, but is not safety related.

3.2.8.2 Radwaste Building Heating and Ventilation System

3.2.8.2.1 Design Bases

The Radwaste Building heating and ventilation system operates to supply filtered and heated air to this building's areas at a rate of approximately one complete volume of air change per hour.

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 heating and ventilating flow through the Radwaste Building. The heating and ventilating system is designed to supply filtered and heated air at approximately 10,900 standard cubic feet per minute and exhaust it after filtration. This corresponds to about one change of air per hour. Air from the building is discharged through the stack.

Exhaust fans and exhaust filters are provided in full-capacity duplicates. Either exhaust fan can then be used to operate the system while the other fan is on standby.

Outside air is drawn into the system through a fixed louver housed above the roof of the building and protected by bird and insect screening. The air is drawn through a filter designed to remove dust. A steam heating coil is provided before fan intake to temper the supply air in the winter. A common 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 stack.

A shunt circuit draws air from the exhaust and monitors its airborne radioactivity. High activity is alarmed in both the Radwaste Building Control Room and the Main Control Room.

The exhaust duct divides into two full-sized parts, each of which contains a roughing filter followed by a high-efficiency filter and an exhaust fan. Dampers with motor operators are provided before the filters and are used to control which of the alternate routes the exhaust is to take. The discharge ducts are combined in a common header and discharge to the plenum leading to the stack. Backflow from other systems is prevented by dampers which are closed if the exhaust fans are not in operation.

3.2.8.2.3 Safety Evaluation

The Radwaste Building heating and ventilation system maintains 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. The Radwaste Building heating and ventilation system are required in the permanently defueled condition because it houses SSCs that are associated with the safe storage and handling of radioactive waste, but is not safety related.

3.2.8.3 Radwaste Storage Building Heating and Ventilation System

3.2.8.3.1 Design Bases

The Radwaste Storage Building's heating and ventilation system operates to supply and exhaust filtered and heated air to and from the potentially contaminated areas of this building.

3.2.8.3.2 System Description

The heating and ventilation system is designed to supply filtered and heated air at approximately 3350 cfm and exhaust from potentially contaminated areas. The exhaust system is arranged with full-capacity spares to ensure availability when required.

3.2.8.3.3 Safety Evaluation

The Radwaste Storage Building heating and ventilation system maintains 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. The Radwaste Storage Building heating and ventilation system are required in the permanently defueled condition because it houses SSCs that are associated with the safe storage and handling of radioactive waste, but is not safety related.

3.2.8.4 Turbine Building Heating and Ventilation

3.2.8.4.1 Design Bases

The Turbine Building heating and ventilation system is operated to above freezing temperatures in all areas of that building.

This ventilation system is also operated to maintain a slight negative pressure in the building to prevent any radioactive out-leakage, as well as, to maintain a ventilating air flow path from the least contaminated areas of the building to the highest contaminated areas. This is done in order to limit the spread of radioactivity within the building. Supply air is filtered because the presence of dust particles potentially increases the spread of radioactive contamination.

3.2.8.4.2 System Description

The Turbine Building ventilation system is designed to provide filtered and heated air at an approximate rate of 92,000 cfm. Two independent air supply systems are provided, each consisting of a fresh air intake, filter, steam heating coil, two 100% capacity fans, dampers and ductwork to distribute air to various areas in the Turbine Building.

The exhaust air system is arranged with supplementary transfer fans and connecting ductwork to induce flow of air through areas of progressively higher contamination potential prior to discharge to the stack.

An air exhaust is located in each room and at each piece of equipment and places where radioactive contamination in the form of dust, gas, or vapor is released. Ducts from these areas lead to an exhaust air header and each inlet has a manually-set control damper.

The chemistry laboratory facilities ventilating system discharges directly to the main exhaust header which runs to the stack.

A shunt circuit draws air from the exhaust manifold and monitors its airborne radioactivity. The circuit is located so that it monitors building air conditions and not the exhaust from individual equipment areas. High activity initiates an alarm in the Control Room.

The building exhaust system discharges into a plenum which also receives air from the Reactor Building. Three 50 percent capacity fans are furnished to handle the combined exhaust from the Turbine and Reactor Buildings. The fans discharge to the common header that is run out to the stack. 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 3.2-12.

3.2.8.4.3 Safety Evaluation

The Turbine Building heating and ventilation system maintains 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. The Turbine Building heating and ventilation system are required in the permanently defueled condition because it houses SSCs that are associated with the safe storage and handling of irradiated fuel or radioactive waste, but is not safety related.

3.2.9 Fire Protection Systems

The Northeast Utilities 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 No. 1. This program establishes the fire protection policy for the protection of structures, systems, and components important to the safety of the plant 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 No. 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 No. 1 and 2 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 No. 1 pumphouse contains one electric driven pump (M7-8), fed from Millstone Unit No. 1 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 No. 1 M7-8 electric pump then will start at 85 psig and it is fed 480 VAC from MCC-CD-6 (MCC # 22A-2 Compartment # 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 No. 1 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 No. 1 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

(1) Sprinkler and Waterspray Systems

Fixed sprinkler and waterspray systems, provided in various areas of the plant where in-situ combustible loading warrants such protection, have been designed using the guidance of National Fire Protection Association (NFPA) Standard No. 13, for the "Installation of Sprinkler Systems" or NFPA Standard No. 15 for "Waterspray Fixed Systems".

Fixed water systems are provided in the following design arrangement:

- Automatic and manual operating, wet-pipe sprinkler;
- Automatic and manual operating waterspray; and
- Automatic operating pre-action sprinkler.

The individual system details and general locations are indicated in the FHA, Ref. 3.2-19.

(2) Automatic and Manual Operating, Wet Pipe Sprinkler Systems

Automatic and manual closed head, wet-pipe design sprinkler systems have an alarm check valve or alarm flow switch, except for the Gas Turbine Building system which is not provided with an alarm. All systems are provided with an outside screw and yoke (OS&Y) isolation valve between the supply connections and system distribution piping.

(3) Automatic and Manual Operating Waterspray Systems

Open spray head, deluge type waterspray systems are of both automatic and manual operating design. All systems have a deluge valve located between the supply header and the distribution piping. An outside stem and yoke (OS&Y) isolation valve or post indicating valve (PIV) is used on all systems. Upon actuation of the deluge valve, water flows into the distribution piping and discharges from all spray heads.

Automatic operation is initiated by a single zone heat detection circuit installed in the hazard area.

Manual operation of the automatic systems is provided electrically via pull stations located at the deluge valve control panel or mechanically at the deluge valve.

(4) Halon Systems

Total flooding Halon 1301 fire suppression systems are provided for the Fire Pump House (FP-1), Computer Room (T-9), Cable Vault Room (T-13), and Control Room (T-17). These systems are designed using the guidance of NFPA 12A, "Halon 1301 Suppression Systems".

Each system is actuated by cross-zoned detectors. The Control Room system provides a minimum concentration within the enclosure of 7% by volume. All other systems provide a minimum concentration within the enclosure of 5% by volume.

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 1/2 inch hose stations or by use of 2 1/2 inch diameter hose with gated wye connections available from outside hose houses.

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 No. 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 No. 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protective Signaling Systems".

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 rate-of-rise and fixed temperature detectors and smoke detectors. Smoke detectors are of the spot type, employing either the ionization or photoelectric 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 No. 72E, "Standard on Automatic Fire Detectors".

Fire/smoke detectors, as with waterflow indicators, Halon actuation 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 Control Room. Trouble signals for these devices are transmitted in a similar manner. Ch. 1

The alarm system also monitors other miscellaneous fire protection system features such as Halon system trouble. Ch. 1

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 (GDC) 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 No. 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.

(f) Acceptability of Manual Fire Suppression: When it can be shown that safe-shutdown capability is independent of any credible fire, manual fire-fighting capability is sufficient to protect safety-related systems.

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.

A detailed description of the analysis method is given in the following summary:

(1) Plant design features related to fire safety were determined. These include the overall plant layout, type, and location of combustible materials, type of construction and its fire resistant characteristics, fire detection, and fire suppression systems, separation distance, etc.

(2) Areas containing equipment and components important to safety were identified. These areas and adjacent areas with fire hazard potential were subdivided into fire areas and zones within areas on the basis of existing fire barrier boundaries

and other logical physical divisions or equipment groupings. For each fire area/zone, the following were determined:

(a) Total heat potential (Btu/ft^2) in the area/zone, assuming total combustion of cable insulation, oil, charcoal, and other identifiable combustibles including transients.

(b) Fire severity is determined by the material burned and its rate of burning. To evaluate the fire resistance needed for any fire barrier, a fire severity (duration) is developed for each area. Severity is measured in terms of temperature and fire duration. Once the total heat potential (Btu/ft^2) of an area or zone has been computed and has been corrected to be equivalent to wood, an equivalent fire severity may be determined.

To determine an equivalent fire severity, Table 5-9B, Estimated Fire Severity for Offices and Light Commercial Occupancies, National Fire Protection Association, Fire Protection Handbook, Fifteenth Edition, was used for this analysis.

(c) Safety-related equipment and systems in the area.

(d) Fire detection and suppression systems.

(e) Fire area/zone boundaries.

(3) For each area/zone, the adequacy of existing fire detection and fire suppression systems was evaluated considering the combustibility of materials, potential ignition sources, and the concentration of combustible materials.

(4) Plant features were evaluated that impact directly or indirectly on the plant fire brigade's ability to reach and effectively fight credible fires.

The specific analysis results for each fire area or zone are provided in the FHA, Reference 3.2-19.

The NFPA Standard Time-Temperature Curve is representative of the severity of a fire completely burning out a brick, wood-joisted building and its contents. The curve has been adopted by the American National Standards Institute and the American Society for Testing and Materials in ANSI/ASTM E 119, Standard Methods of Fire Tests of Building Construction and Materials. This curve has been used in the fire hazard analysis to evaluate fire duration.

3.2.9.3.3 Fire Hazard Analysis Assumptions

The fire hazard analysis was based on the following assumptions:

(1) Fire areas were established as defined in Section 3.2.9.3.4.

(2) Fire zones were established as defined in Section 3.2.9.3.4.

(3) Floor area calculations assume that the entire floor area within the boundaries, as defined by exterior walls or other accepted divisions, is available for fire loading. Obstructions within a given fire area, such as equipment or rooms that do not extend to the above floor elevation, are available for fire loading and, therefore, the surface area of these obstructions is included in the floor area calculation. The areas occupied by interior walls and columns (except in limited occurrences) are considered negligible. Stairwells and elevators are excluded from the floor area calculation.

(4) Complete combustion of all the in-situ and transient combustible contents of a fire area are postulated.

(5) All cable trays are assumed filled to the maximum allowable depth, allowing 50 percent of the volume for combustible cable insulation in determining the fire loading.

(6) Transient combustibles are identified as those combustibles not normally considered in-situ, but assumed to be within an area for maintenance purposes. Normally, one quantity of the largest single volume of transient combustibles is based on complete change out of lubricating/hydraulic oil, charcoal, and resins. For areas containing replaceable resins and/or charcoal, one open container of each, as applicable, is assumed to be in the area. For areas containing replaceable oils, quantity is determined by the amount of in-situ oil and the logical container(s) in which the oil would be transported.

(7) Elimination of the fire hazard (or sufficient limitation of sphere of influence) is an acceptable method of providing fire protection.

(8) The adequacy of fire doors and dampers or other protected penetrations of fire area boundaries is based on the design and the rating of the door or damper compared to the rating of the penetrated boundary. A one-hour or three-hour rating has been used as a general guideline for the rating of barriers. Barrier ratings are based upon the hazard they are to protect. Enclosures, especially shielding walls, without openings could have significantly higher ratings. Protection of the openings is qualified to the required or desired fire rating of three hours or less based upon the required rating of the barrier. (e.g., one-hour barrier needs one-hour door or better).

(9) Resins that are stored in steel tanks are not considered in the combustible loading provided the tanks are adequately vented.

(10) Cables that are routed in conduit and junction boxes do not function as intervening combustibles and do not contribute to the fire loading of the area.

(11) BTP APCSB 9.5.1 requires that the interior finishes should be noncombustible or listed by a nationally recognized testing laboratory, such as Factory Mutual or Underwriters Laboratories, Inc., for flame spread, smoke, and fuel contribution of 25 or less in its use configuration (ASTM E-84 Test, Surface Burning

Characteristics of Building Materials). Finishes in, this context are not directed at paint or coating systems for walls and equipment since these items do not normally constitute an appreciable fire loading. Paint is considered in the fire hazard analysis as part of the negligible fire loading.

(12) All motors are metal except for winding insulation and bearing lubricant. The winding insulation consists of mica and resins. The mica is nonflammable. The resins may burn if ignited, but they require high temperatures. The total amount of resin in a motor is about 2 percent by weight. Except for the reactor recirculation pump motor generator the amount of bearing lubricant in a motor is small. On these bases, they are considered in the fire hazard analysis as part of the negligible fire loading.

(13) All insulation and replaceable filter media shall be listed by the Underwriters' Laboratories and shall have a flame spread rating of not more than 25 in accordance with ASTM E-84. For this reason insulation and filter media are considered in the fire hazard analysis as part of the negligible fire loading.

(14) Office areas are assumed to have a fire loading of 6 PSF (heat potential of 48,000 Btu/ft^2).

(15) Heat content for various combustible materials is based on the materials heat of combustion as derived from the NFPA Handbook, Section 4, Chapter 12 (15th edition). Heat of combustion is defined as the heat generated when 1 lb. of combustible material is burned completely in an experimental calorimeter.

Any total heat potential less than 3,500 Btu/ft^2 is considered negligible.

(16) Maximum credible spill has been determined based on anticipated leakage rates associated with pipe breakage.

3.2.9.3.4 Fire Areas and Zones

The plant arrangement is divided into fire areas and fire zones for purposes of conducting the fire hazard analysis. Fire areas are defined as plant areas bounded by fire-rated assemblies of either three-hour rated construction or lesser fire resistance as specifically identified and justified in the fire hazards analysis. For this analysis a fire area may also be defined by physical separation of combustible materials and built-in fire suppression features which will contain a fire to an area of origin. Examples of built-in features include water curtains and gaseous fire suppression systems.

Fire zones are zones within fire areas that are used to more thoroughly describe the fire area. Fire zones may or may not be bounded by fire-rated construction.

For the purposes of the Fire Hazards Analysis, the fire areas and fire zones were divided in accordance with the original divisions in the 1977 Fire Hazards Analysis. These divisions provide a clear indication of the combustibles within a location and their effect on equivalent fire severity.

Drawings provided in the FHA show fire zone/area division and serve as reference for this section.

3.2.9.3.5 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 Operations 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. This program emphasizes those elements of fire protection that are associated with safe shutdown and their significance when evaluating program and equipment deficiencies.

The technical requirements found in Millstone Unit No. 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) Pre-action water spray systems
(5) Pre-action sprinkler systems Ch. 1
(6) Foam application system
(7) Halon total flooding systems
(8) Local application Halon systems
(9) Fire pumps
(10) Fire barriers (walls, fire doors, penetration seals, fire dampers)
(11) 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 Fire Protection Engineer for evaluation.

As conditions warrant, remedial actions would include compensatory measures to ensure an equivalent 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 overall responsibility for the Fire Protection Program at Millstone Unit No. 1 rests with the Chief Nuclear Officer Millstone. Formulation, and assessment of the effectiveness of the program are delegated as indicated in Reference 3.2-24, the Fire Protection Program Manual.

3.2.9.5.2 Fire Brigade and Training

The Millstone Site Fire Brigade consists of a minimum of a Shift Captain and four Fire Fighter Technicians. The affected unit shall supply an advisor, who is at a minimum a fully qualified Plant Equipment Operator, to the Fire Brigade Shift Captain. 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 Protection personnel are responsible for responding to all fires, fire alarms, and fire drills. To ensure availability, a minimum of a Shift Captain and four Fire Fighter Technicians shall remain in the owner controlled area and shall 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 Captain coordinates the Site Fire Protection 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 Protection 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 shall be planned and critiqued by the Site Fire Marshal and members of the management staff responsible for plant fire protection. Performance deficiencies of Fire Protection or of individual Fire Protection personnel shall be remedied by scheduling additional training for Site Fire Protection or individuals.

3.2.9.5.3 Quality Assurance

The QA Program has been applied via the Fire Protection Program Manual to the FPSs, components, and programs providing fire detection and suppression capabilities to those areas of the plant that are important to the defueled condition.

3.2.10 References

3.2-1 Docket No. 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 No. 50-245, B10301, W.G. Counsil to D.M. Crutchfield, 'Millstone Nuclear Power Station, Unit No. 1, SEP Topic IX-1, Fuel Storage,' August 31, 1981.

3.2-3 Docket No. 50-245, B10346, W.G. Counsil to D.M. Crutchfield, 'Millstone Nuclear Power Station, Unit No. 1, SEP Topic IX-1, Fuel Storage,' December 14, 1981.

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

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

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

3.2-7 Docket No. 50-245, BI 5683, F. R. Dacimo to USNRC, "Millstone Nuclear Power Station, Unit No. 1, Proposed Technical Specification Revision - Fuel Storage," May 5, 1996.

3.2-8 Docket No. 50-245, Millstone Nuclear Power Station, Unit No. 1, Issuance of Amendment No. 97. October 1996.

3.2-9 Holtec Report No. H1-971914, Revision 1, "Analysis Of 1675 Pound Fuel Assembly System Drop Onto The Irradiated Fuel Assembly"

3.2-10 Holtec Report No. AH1-971691, Revision 0, "Criticality Safety Analysis Of The MP1 Racks With A Dropped Fuel Assembly"

3.2-11 Holtec Report No. H1-971698, Revision 0, "Flow And Temperature Field Analysis Of Localized Cell Blockage In The Millstone Unit No. 1 Spent Fuel Pool"

3.2-12 Holtec Report No. H1-971675, Revision 1, "Analysis Of Tetrabor And Boraflex Racks Under 1675 Pound Fuel Assembly System Impact"

3.2-13 Holtec Report No. H1-98210, Revision 1, “Decay Heat Load Calculation for the Millstone Unit 1 Spent Fuel Pool”

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

3.2-15 Docket No. 50-245, B10291, W.G. Counsil to D.M. Crutchfield, "Millstone Nuclear Power Station, Unit No. 1, SEP Topic IX-3, Station Service and Cooling Water Systems," November 24, 1981.

3.2-16 Docket No. 50-245, NUREG-0824, Integrated Plant Safety Assessment, Systematic Evaluation Program, Millstone Nuclear Power Station, Unit No. 1, February 1983, Topic IX-3, "Station Service and Cooling Water Systems."

3.2-17 Docket No. 50-245, B10292, W.G. Counsil to D.M. Crutchfield "Millstone Nuclear Power Station, Unit No. 1, SEP Topic IX-5, Ventilation Systems", November 19, 1981.

3.2-18 Docket No. 50-245, LS05-82-09-043, J. Shea to W.G. Counsil, "SEP Topic IX-5, Ventilation Systems, Millstone Nuclear Power Station, Unit No. 1, September 14, 1982.

3.2-19 Fire Hazard Analysis Millstone Unit No. 1, September 1986

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

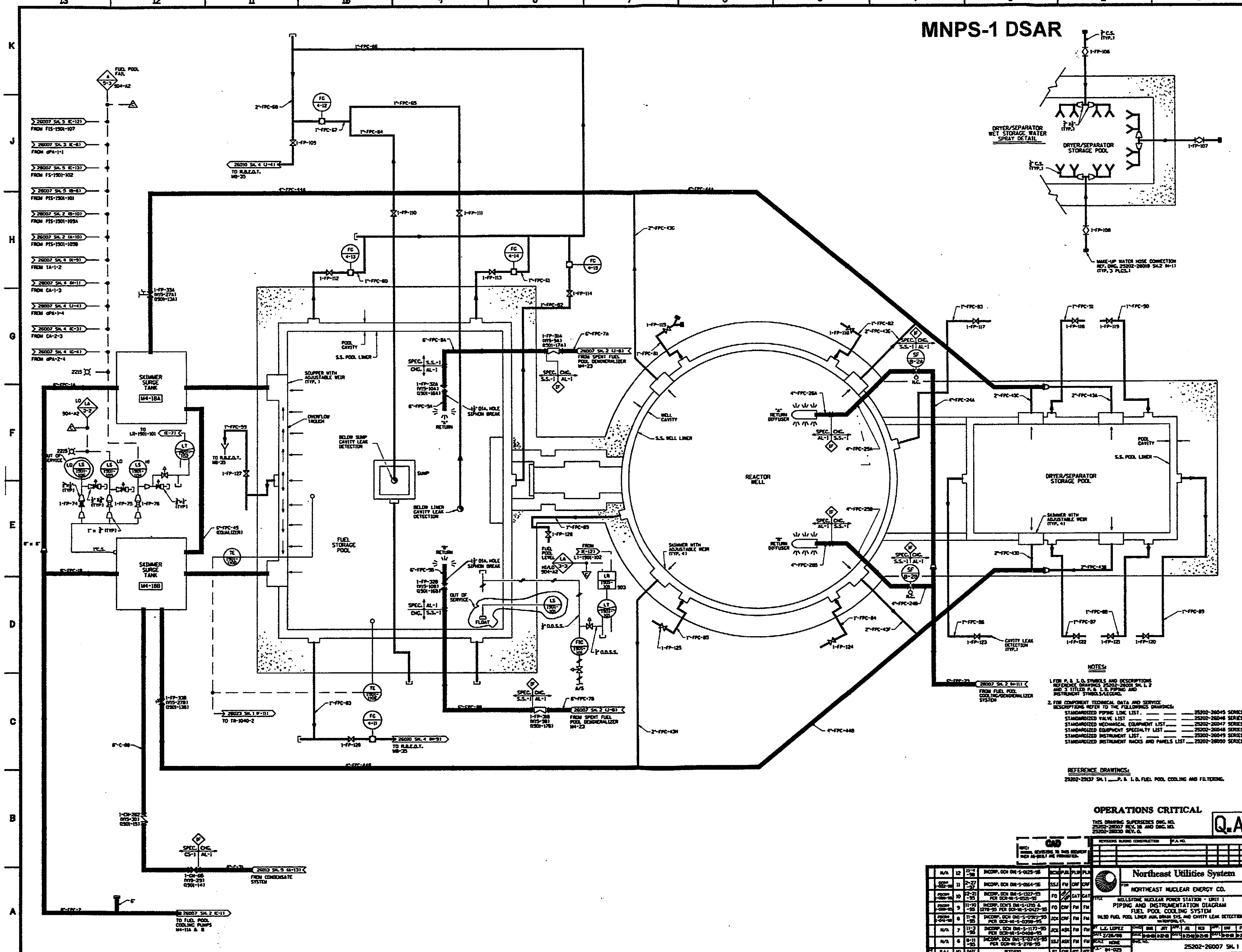

FIGURE 3.2-1

September 1999

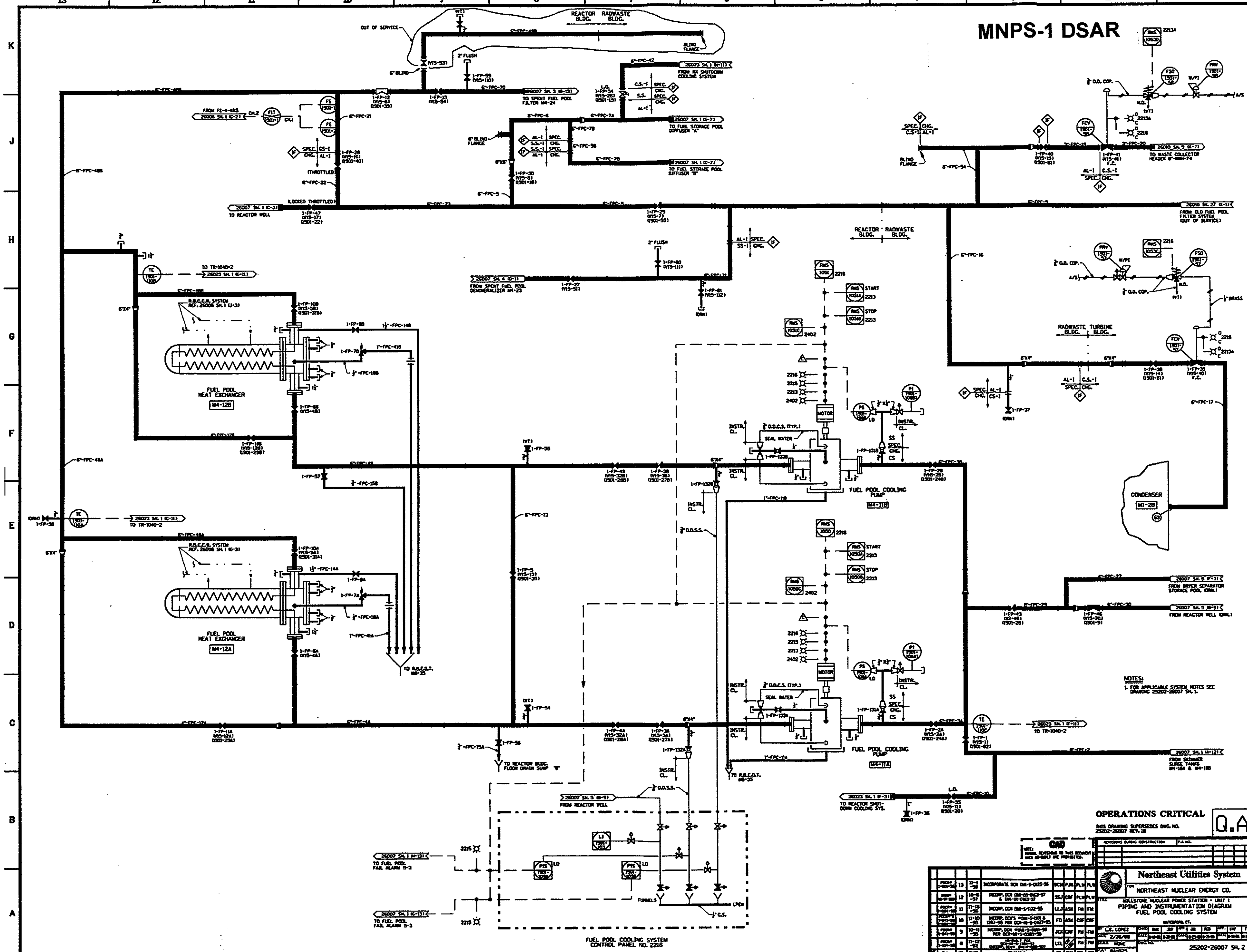
REACTOR RADWASTE BLDG. BLDG.
OUT OF SERVICE
FUEL POOL HEAT EXCHANGER
M4-12B
M4-12A
FUEL POOL COOLING PUMP
M4-11B
M4-11A
CONDENSER
M1-28
RADWASTE TURBINE BLDG. BLDG.
REACTOR RADWASTE BLDG. BLDG.
FUEL POOL COOLING SYSTEM CONTROL PANEL NO. 2216
NOTES:
1. FOR APPLICABLE SYSTEM NOTES SEE DRAWING 25202-26007 SH. 1.
OPERATIONS CRITICAL
Q.A.
Northeast Utilities System
NORTHEAST NUCLEAR ENERGY CO.
MILLSTONE NUCLEAR POWER STATION - UNIT 1
PIPING AND INSTRUMENTATION DIAGRAM
FUEL POOL COOLING SYSTEM
25202-26007 SH. 2

FIGURE 3.2-2

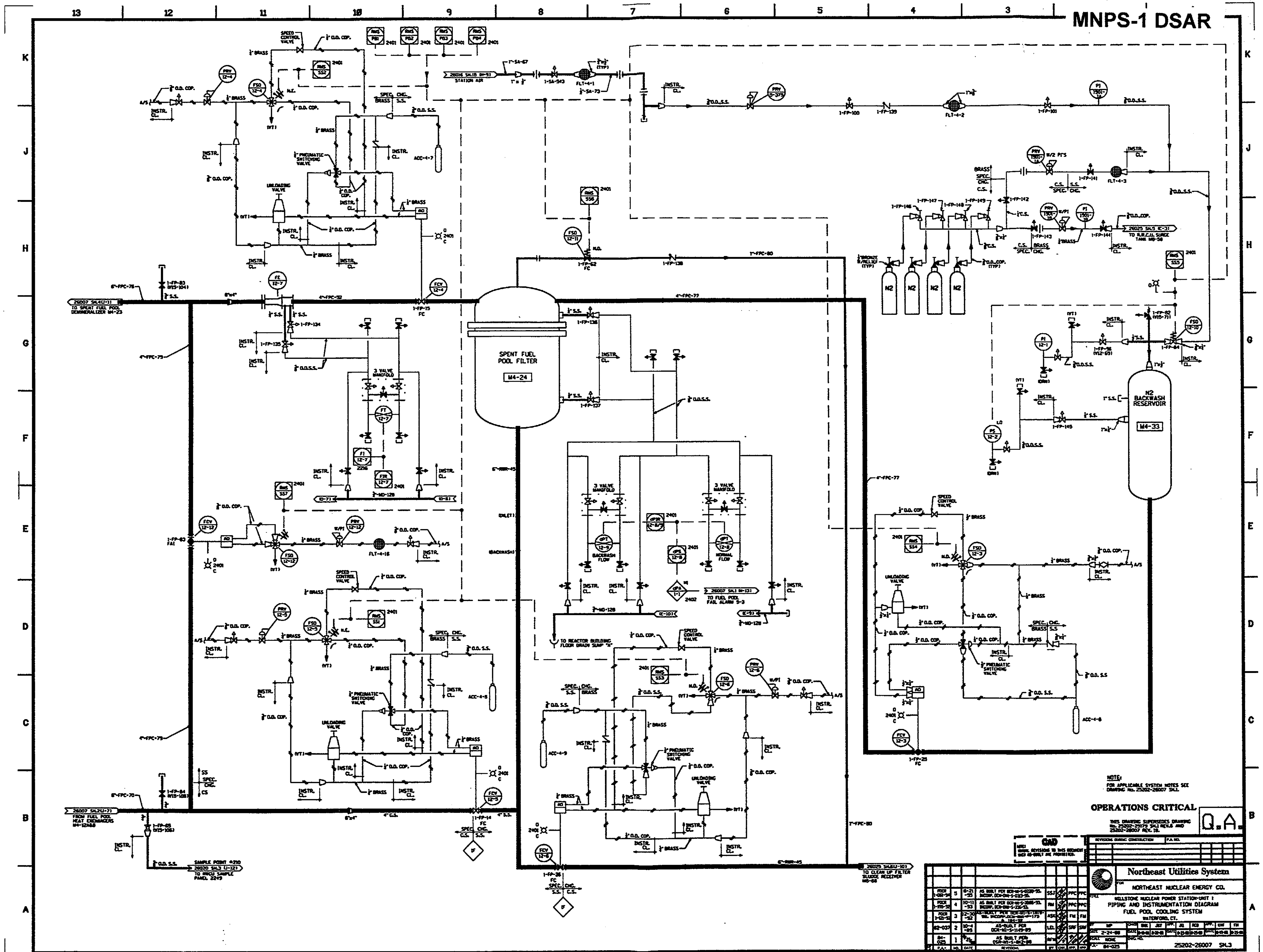

FIGURE 3.2-3

September 1999

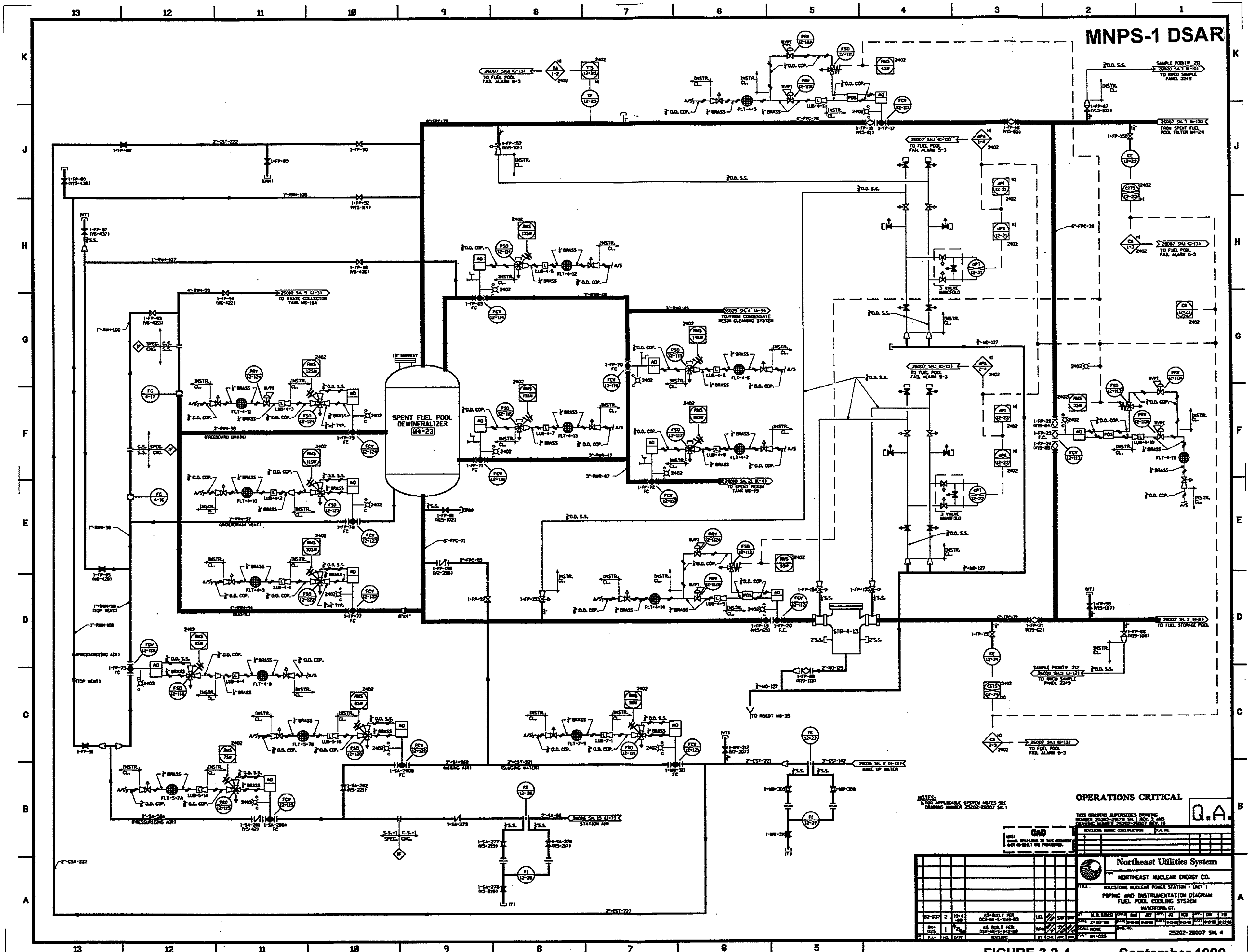
SPENT FUEL POOL DEMINERALIZER
M4-23
OPERATIONS CRITICAL
Q.A.
Northeast Utilities System
NORTHEAST NUCLEAR ENERGY CO.
MILLSTONE NUCLEAR POWER STATION - UNIT 1
PIPING AND INSTRUMENTATION DIAGRAM
FUEL POOL COOLING SYSTEM
WATERFORD, CT.
25202-26007 SH. 4
NOTES:
1. FOR APPLICABLE SYSTEM NOTES SEE DRAWING NUMBER 25202-26007 SH. 1
TO RBCCT M8-35
MAKE UP WATER
STATION AIR
TO FUEL STORAGE POOL
TO FUEL POOL FAIL ALARM 9-3
TO SPENT RESIN TANK M8-19
TO/FROM CONDENSATE RESIN CLEANING SYSTEM
TO WASTE COLLECTOR TANK M8-16A
STR-4-13

FIGURE 3.2-4

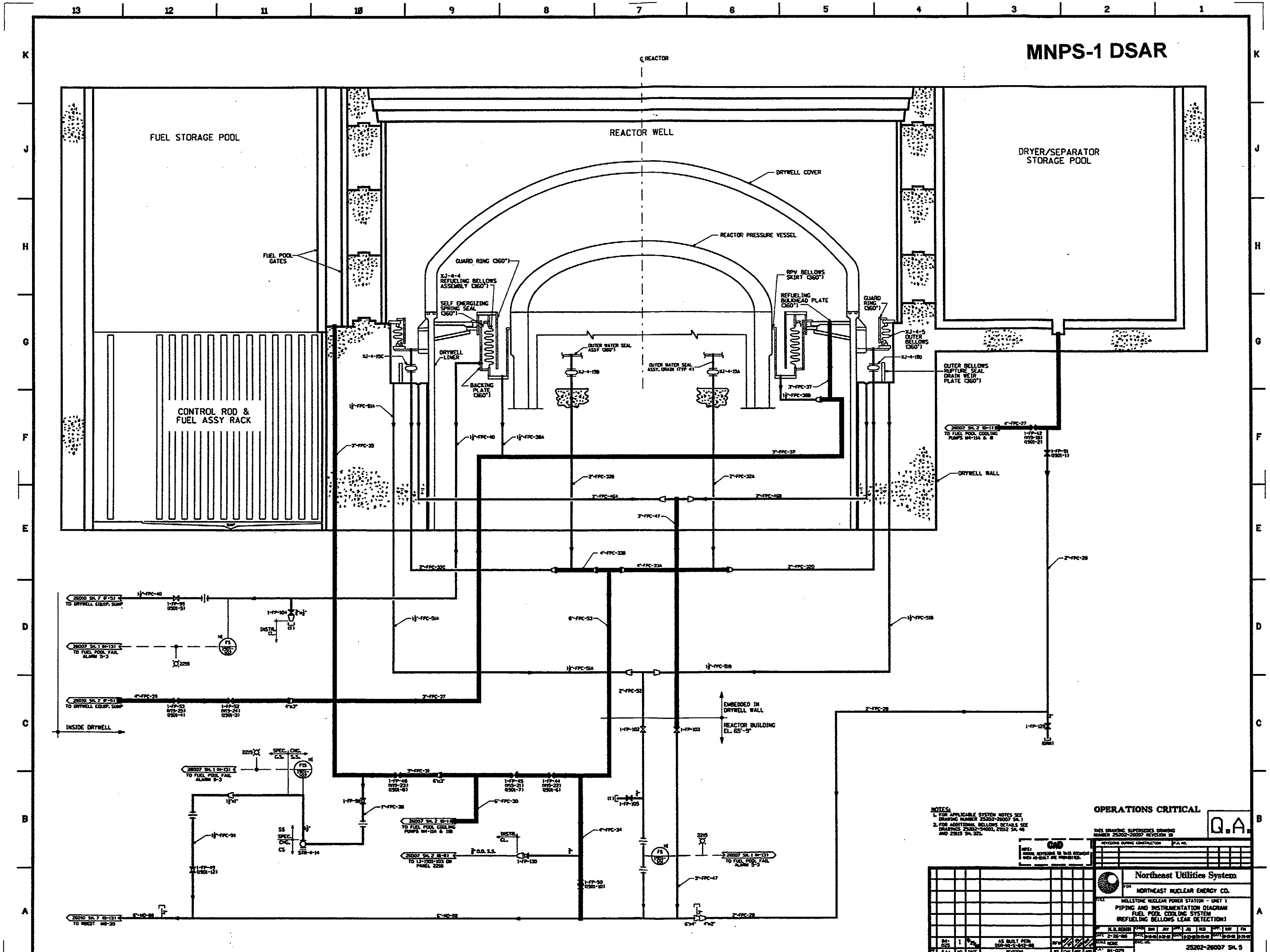

FIGURE 3.2-5

September 1999

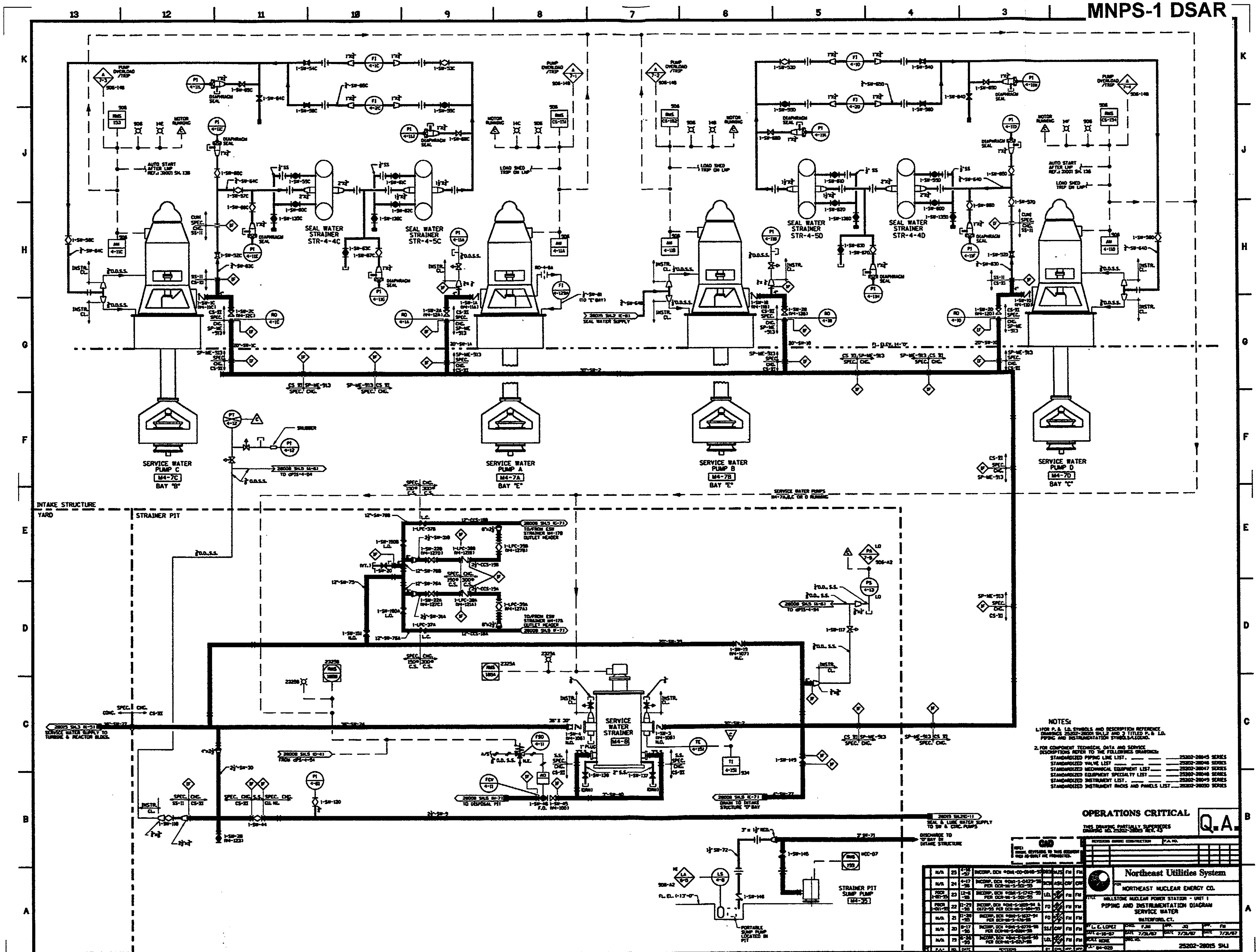

FIGURE 3.2-6

September 1999

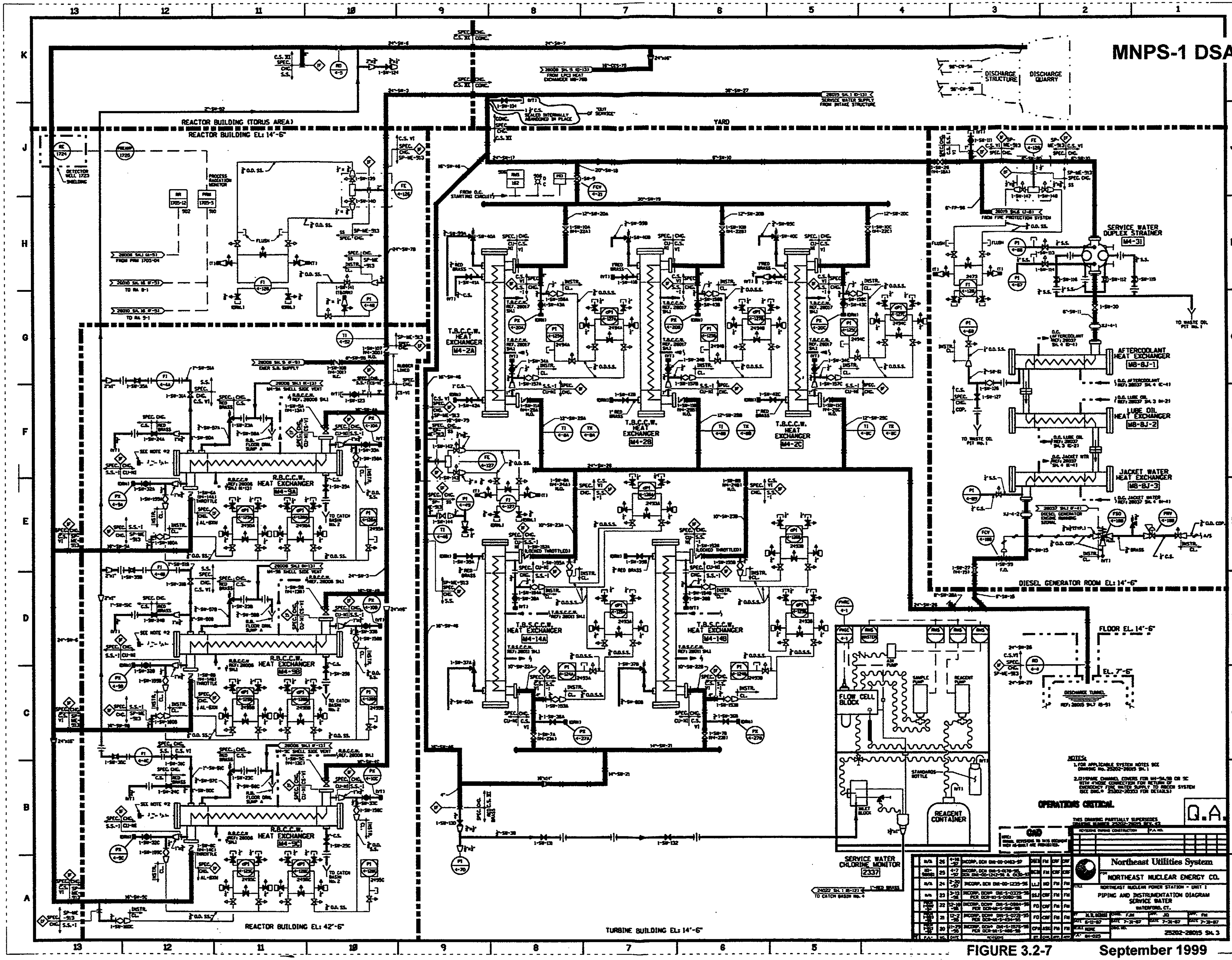

FIGURE 3.2-7 September 1999

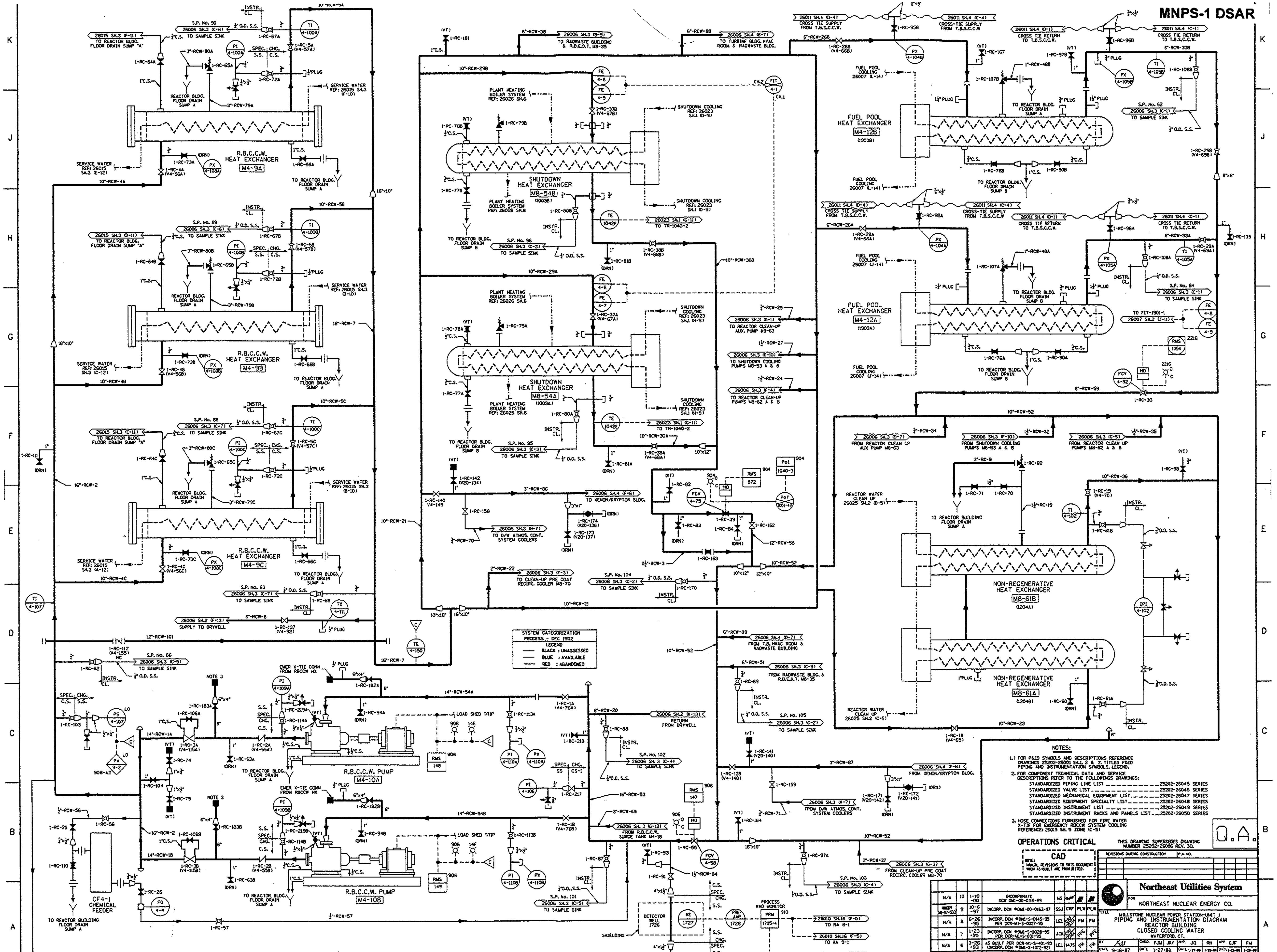

FIGURE 3.2-8 June 2000

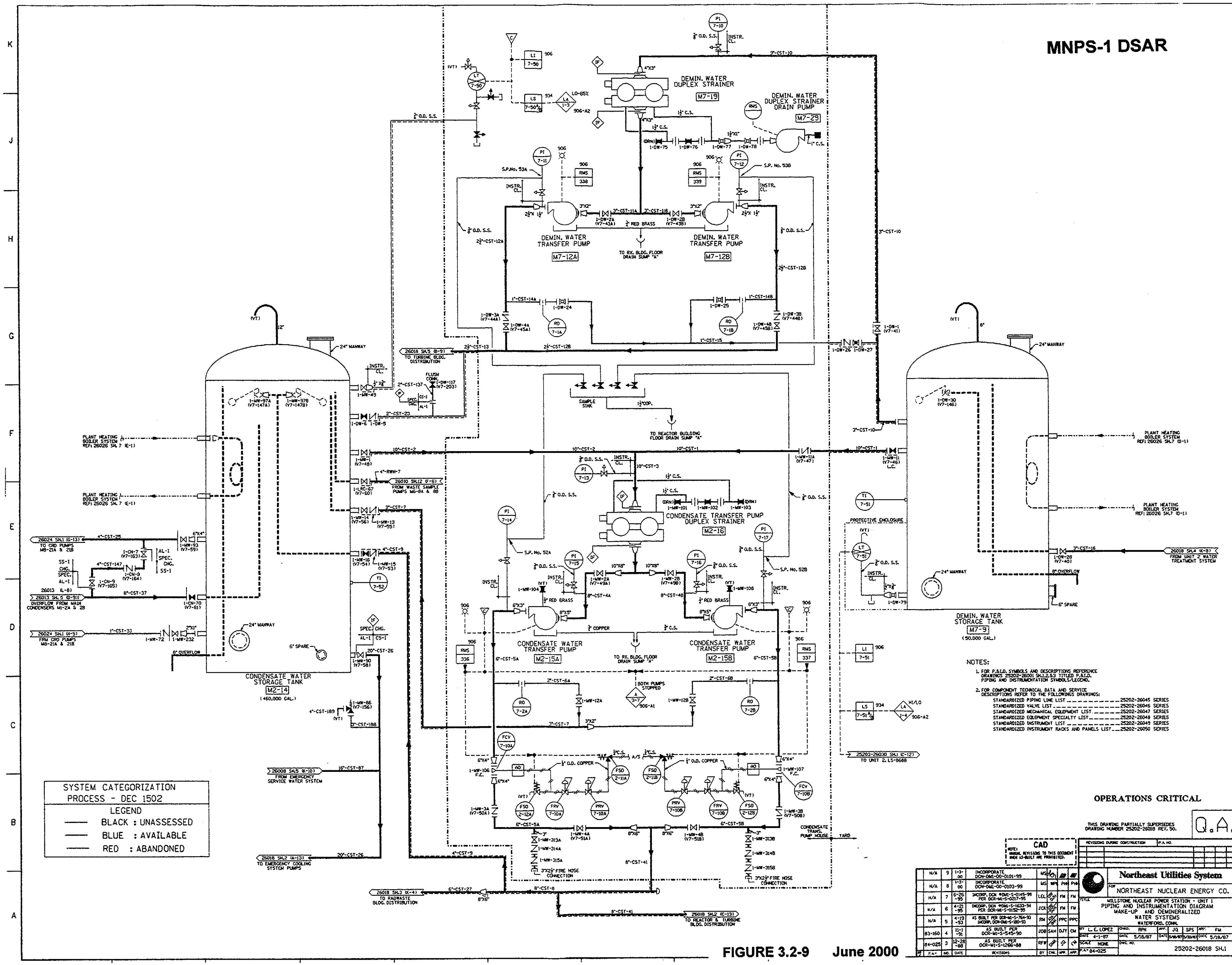

FIGURE 3.2-9 June 2000

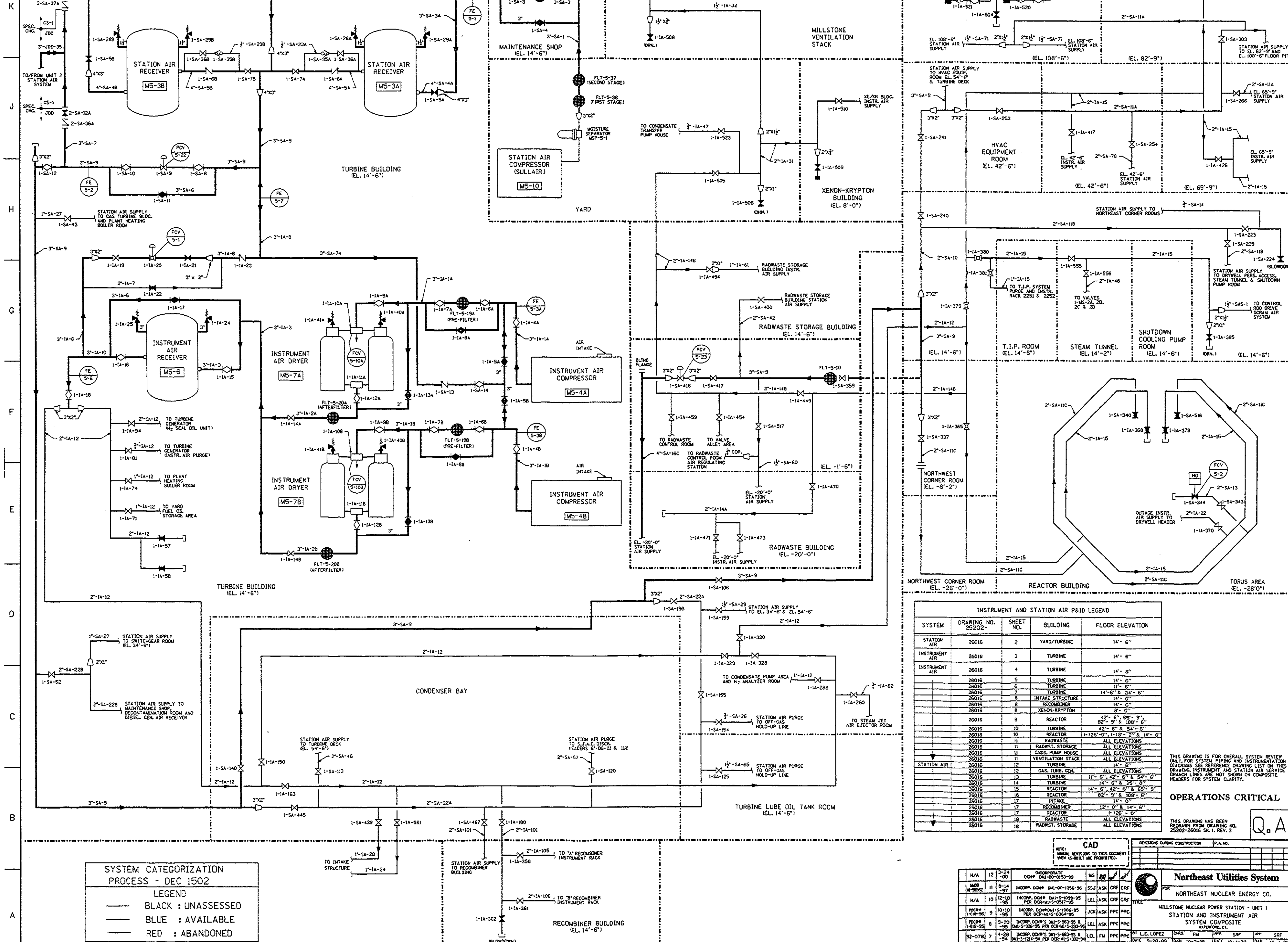

FIGURE 3.2-10 June 2000

MNPS-1 DSAR

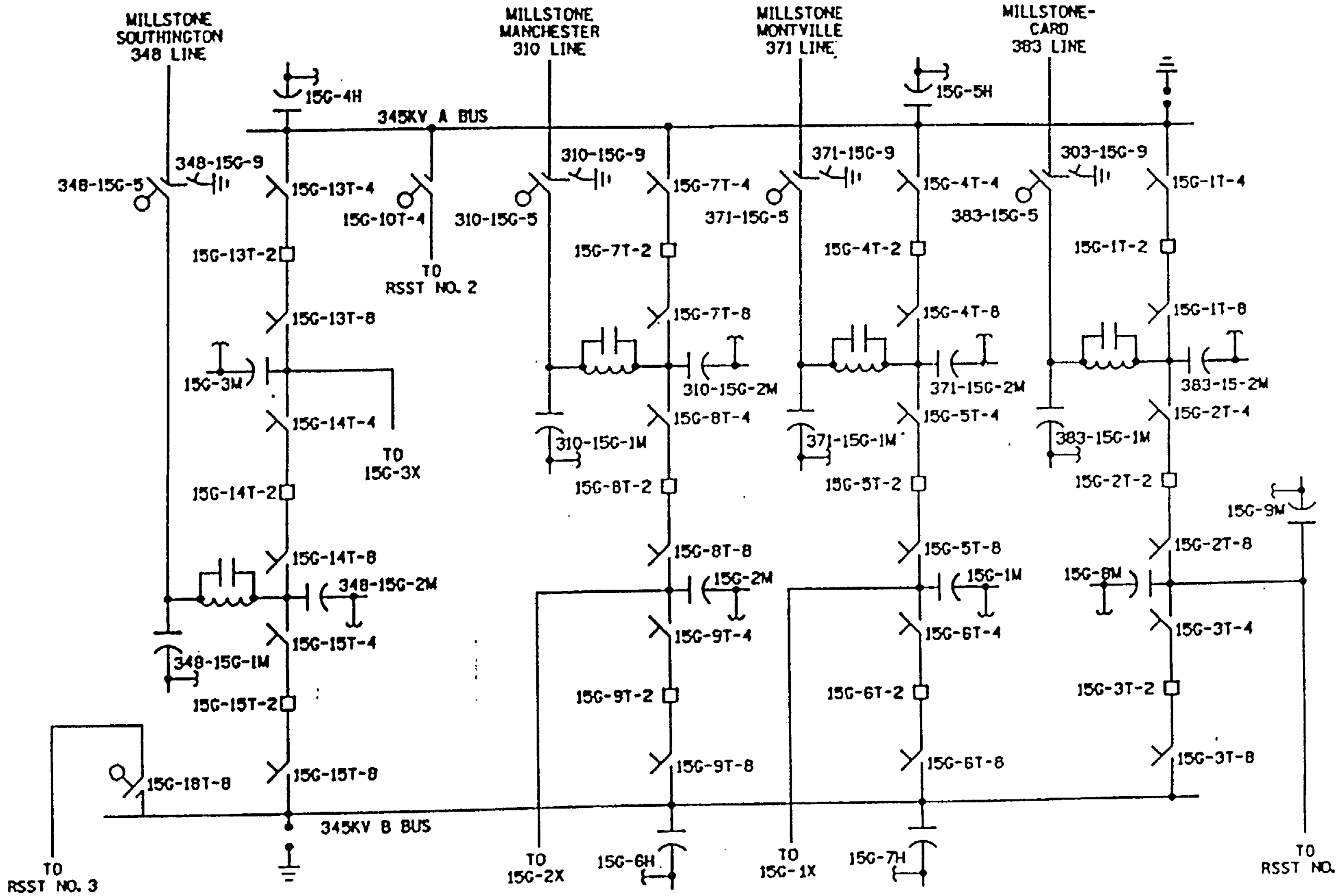

FIGURE 3.2-11
345 KV SWITCHYARD

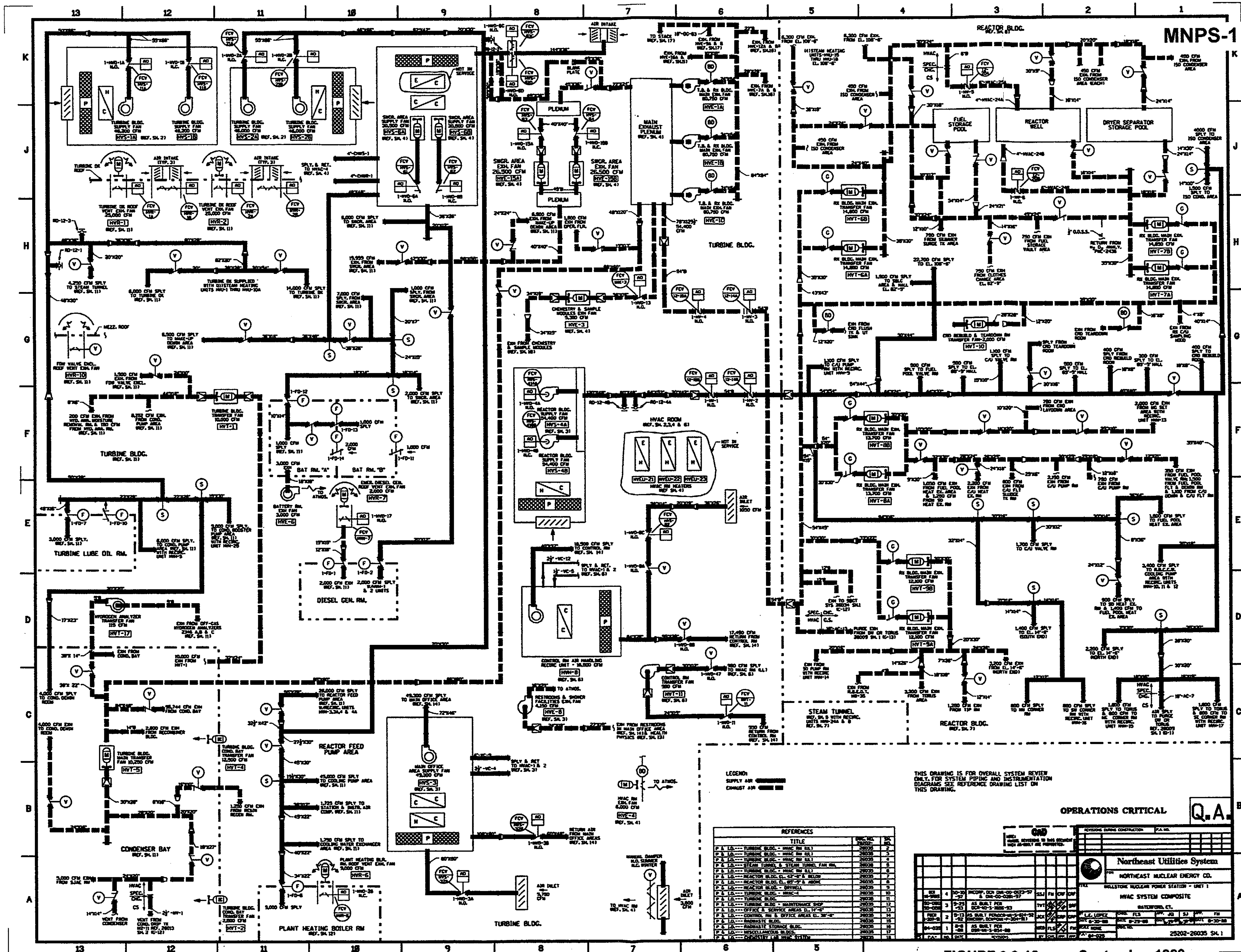

FIGURE 3.2-12

September 1999

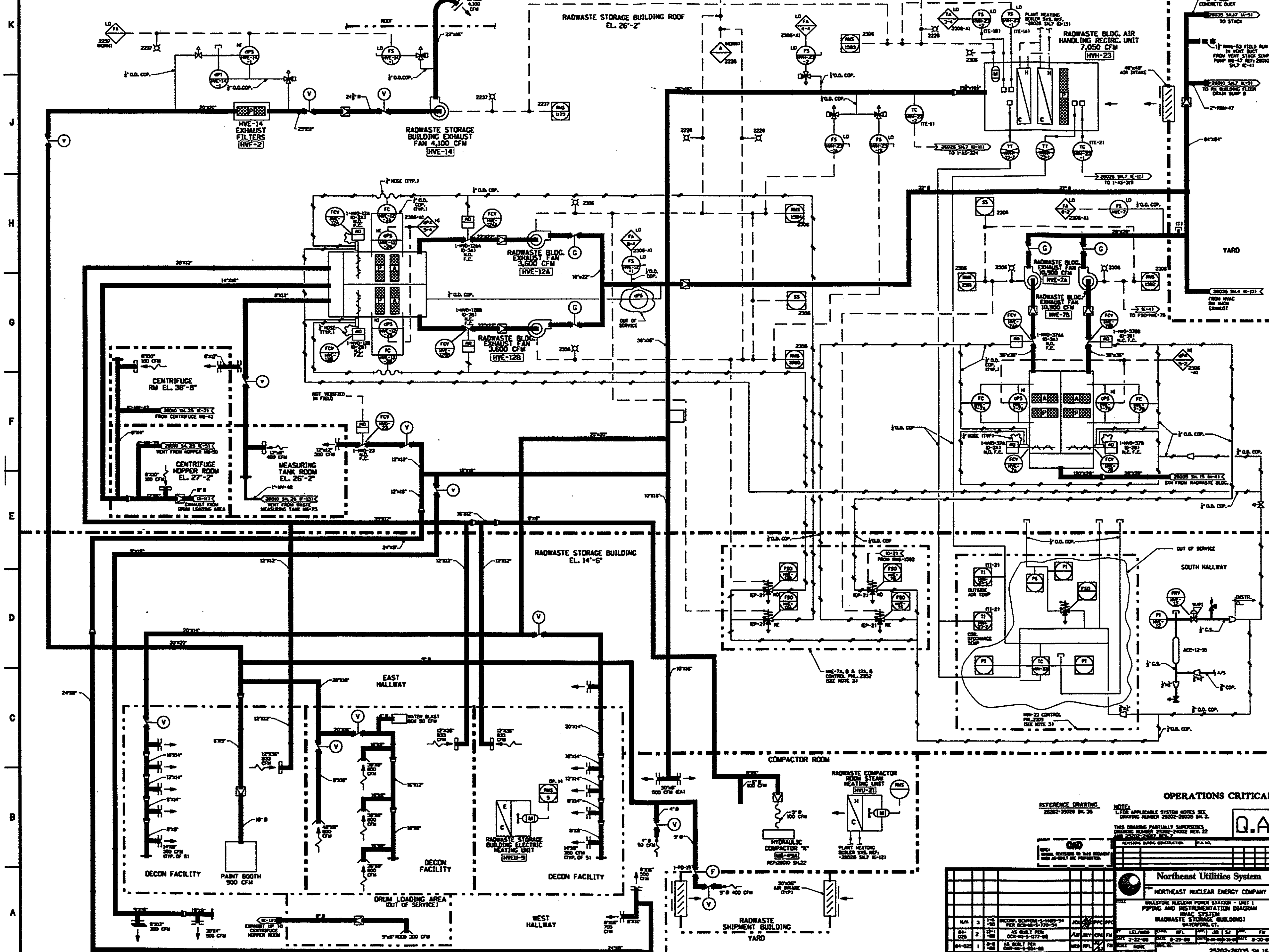

FIGURE 3.2-13 September 1999

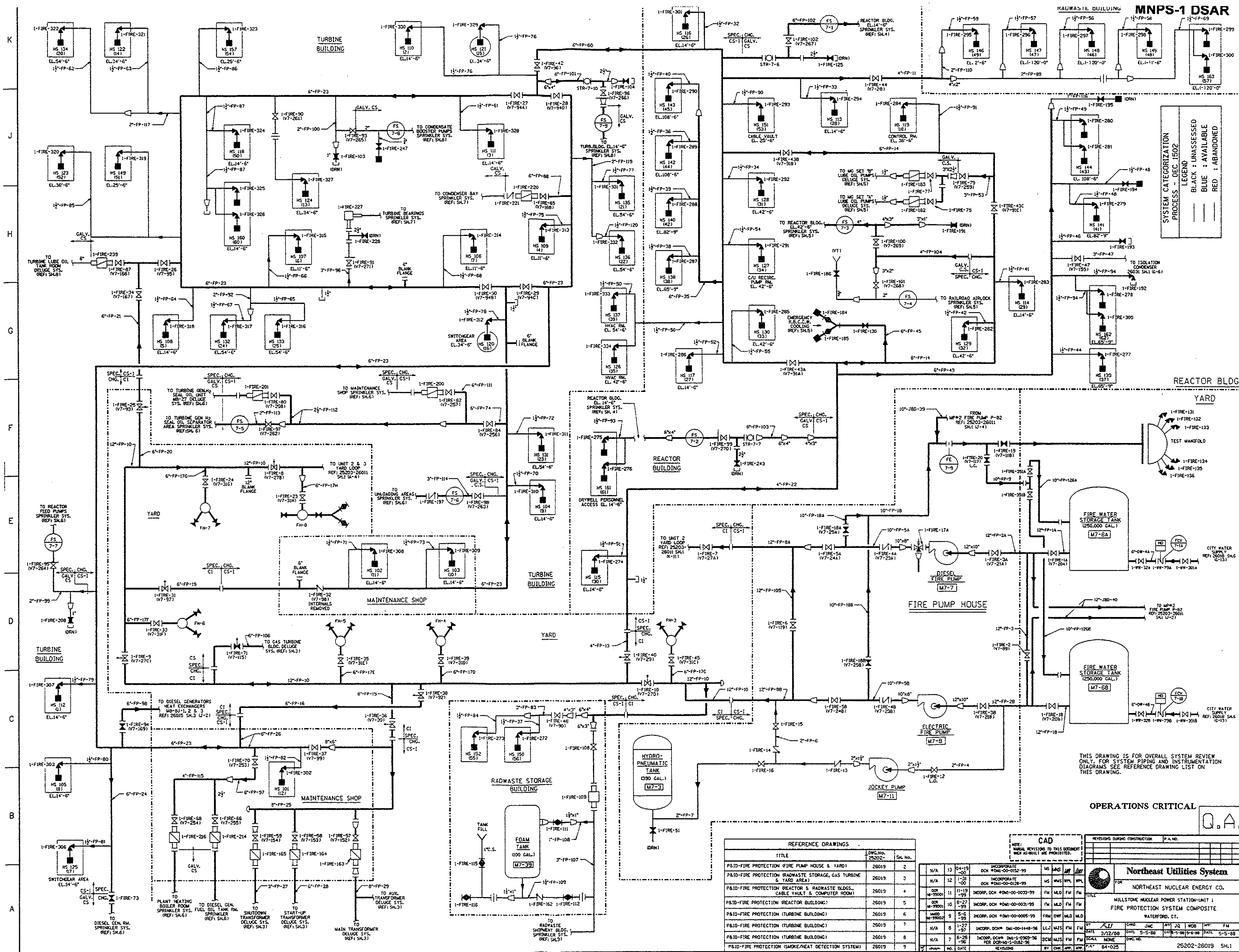

FIGURE 3.2-14 June 2000

CHAPTER 4

RADIOACTIVE WASTE MANAGEMENT

4.1 SOURCE TERMS

Sources of radioactivity that served as the original design bases for the waste management systems are provided in Reference 4.1-1, Demonstration of Compliance with 10 CFR 50, Appendix I. Parameters and assumptions used to determine the specified radioactivity of each isotope are provided. Also described are the mathematical models and bases for the values used to calculate the source terms.

Leakage rate from fluid systems containing radioactivity are provided in Reference 4.1-1. Estimates of radioactive release and the bases for the values are indicated.

4.1.1 References

4.1-1 Millstone Unit 1, Docket No. 50-245, Demonstration of Compliance 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 No. 1 is permanently shutdown and many installed components that are provided shielding, are not 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.

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 ultimate responsibility for implementing the radiation protection program lies with the Chief Nuclear Officer.

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 Northeast Nuclear Energy Company (NNECO) to maintain individual and plant personnel total radiation exposure ALARA. NNECO'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

The liquid waste management systems (LWMS) were originally designed to collect, store, process, and dispose of, or recycle, all radioactive or potentially radioactive liquid waste generated by the plant operation safely and economically.

The LWMS include shielding, equipment, valves, piping, instrumentation and controls, and are housed in the Radwaste Building. The LWMS are operated from a control room located in the Radwaste Building. The systems operate on a batch basis and consist of three systems to process and treat each of the various kinds of liquid radioactive waste encountered. Each system is made up of collector tanks for holding and sampling liquid waste prior to processing, decontamination process equipment, and post-processing sampling tanks where the liquids are sampled and monitored before being recycled back to the plant or discharged off-site.

Radioactive liquid wastes are classified according to their degree of chemical purity and radioactivity. The classifications used at Millstone Unit No. 1 are:

(1) <u>High Purity Wastes</u>

These are wastes with low conductivity and variable radioactivity. They are normally treated by filtration and ion exchange. Following treatment, they may be discharged.

These wastes come from piping and equipment containing high quality water. These wastes are collected in the drywell equipment drain sump, the Reactor Building equipment drain tank, and the Turbine Building equipment drain sump. The system for handling and treating the high purity water from equipment leakage and drainage is the waste collector system.

(2) <u>Low Purity Wastes</u>

These are wastes with high conductivity. These wastes are normally processed by vendor-supplied filtration/ion exchange beds and discharged. These wastes, when low enough in conductivity, can be processed through the waste demineralizer (ion exchanger).

Low purity wastes are collected primarily from floor drains and drains from the cooling water side of heat exchangers. They are collected from two Radwaste Building sumps, two Reactor Building floor sumps, the Turbine Building floor drain sump, and the drywell floor drain sump. The system that handles and processes these low purity liquid wastes is the floor drain system.

(3) Decontamination Wastes

These wastes contain detergents and arise from the equipment decontamination station and the personnel decontamination station. These wastes are kept separate from other radioactive liquid wastes and are collected and held in the decontamination solution tank and released through a cartridge type filter to Long Island Sound.

(4) Resin and Filter Slurry

These wastes are collected in the spent resin tank. The effluent is decanted and pumped to the radwaste floor drain sump. The slurries are pumped to the solidification facility using the transfer/solidification system.

4.4.1 Design Bases

The systems are designed to operate on a batch basis with sampling, monitoring and hold-up provisions to ensure compliance with the provisions of 10 CFR 20 and 10 CFR 50, Appendix I. Sufficient capacity is provided to accommodate wastes produced from normal activities at the plant.

The design provides for segregation in the collection and processing of the various kinds of liquid radioactive waste, and also provides cross-connections between the various processing systems and associated equipment for flexibility of operation.

The LWMS have sufficient volume retention and throughput capacity to ensure that no untreated wastes need be released.

The system design provides for decay of the initial radioactivity due to the wastes being held up in the various tanks of the LWMS. The design ensures that the decontamination and release of the most contaminated influent to the system will be in compliance with 10 CFR 20, and 10 CFR 50 Appendix I.

Several radwaste components have been determined to be critical to support the function of the Liquid Radwaste Effluent Radiation monitor. These critical Radwaste components are identified to be the isolation valves of the 5 dischargeable tanks ("A" and "B" Waste Sample tanks, "A" and "B" Floor Drain Sample tanks, and Decon Solution tank) and the 2 automatic radiation monitor isolation valves. The valves are designed to secure flow. The valves are periodically inspected to ensure that they operate in accordance with their design requirements. Periodic maintenance and inspections will be performed on these valves to ensure that the valves are capable of securing flow.

Although the LWMS are not safety-related, the below grade portion of the Radwaste Building has been seismically analyzed and the building entrances at grade level are protected against water entry from hurricane flooding. The Radwaste Building and equipment arrangement provide assurance that the Radwaste Building will form a radioactive waste boundary and prevent excessive radioactive material release. Seismic and quality group classifications are provided in Chapter 3.

The LWMS are designed to be operable from a single Radwaste Control Room. The status of most pumps and valves necessary to the functioning of the system is displayed in the Radwaste Control Room. Tank level information, temperatures and pressures important to the system operation are also displayed.

4.4.2 System Description

The LWMS utilize three systems to process all of the liquid radioactive wastes produced by plant operations. These systems are the waste collector system, the floor drain system, and the transfer/solidification system. The various types of liquid radioactive waste processed and treated have previously been described. The three processing systems are described below.

4.4.2.1 Waste Collector System

The waste collector system provides for the collection and processing of high purity liquid waste during all phases of plant operation. These are characterized by having low conductivity, moderate to high radioactivity and nearly neutral pH water.

Wastes from the equipment drains are pumped to the waste collector tank. That tank also collects the high purity, low conductivity wastes from the waste demineralizer drains and vents produced by the LWMS. These wastes are processed through a filter to remove solid particles, and then passed through one of two mixed bed waste demineralizers to further reduce the conductivity. Following these steps, the wastes are collected in one of the two waste sample tanks where, after the wastes are sampled and analyzed, they are normally discharged to the condensate storage tank (CST) for reuse in the plant. Wastes from the sample tanks can also be discharged to Long Island Sound, should plant conditions require.

4.4.2.2 Floor Drain System

4.4.2.2.1 Floor Drain Wastes

The low purity, high conductivity liquid wastes from floor drains are collected in sumps and pumped to one of the four floor drain collector tanks. The wastes are pumped to the vendor-supplied filtration/ion exchange skids or can be cross-tied to the waste collector system for processing.

The processed wastes are collected in one of two floor drain sample tanks. After the wastes are sampled and analyzed, they are either discharged from the plant via the discharge canal, or they are pumped to a floor drain collector tank for reprocessing depending upon sample results.

4.4.2.2.2 Decontamination Wastes

Liquid wastes from the equipment decontamination area and the personnel cleanup areas are pumped to the decontamination solution tank. From there it is pumped through a cartridge type filter, and if within established limits, discharged directly to Long Island Sound.

4.4.2.3 Transfer/Solidification System

Waste from the spent resin tank is pumped via the transfer/solidification system to the shipping bay to be dewatered in a high integrity container or solidified for off-site transport and disposal.

4.4.3 Radioactive Releases

The treatment, and discharge of process liquid wastes is discussed in Reference 4.4-1. Parameters and assumptions used in calculating releases of radioactive materials in liquid effluents, and their bases is also provided in this reference.

A current summation of all releases and current off-site dose estimates resulting from liquid effluents are provided in Reference 4.4-2. The methods and criteria for determining the contents of these two reports are contained in Reference 4.4-3. Reference 4.4-3 provides sampling and analysis programs and meets the intent of 10 CFR 20 and 10 CFR 50. In addition, it outlines the information to be submitted to the NRC in References 4.4-2 and 4.4-4.

4.4.4 References

4.4-1 Millstone Unit No. 1, Docket No. 50-245, Demonstration of Compliance with 10 CFR 50, Appendix I

4.4-2 Millstone Nuclear Power Station Unit No. 1, Docket No. 50-245, Annual Radioactive Effluents Report.

4.4-3 Millstone Unit No. 1, Docket No. 50-245, Radiological Effluent Monitoring and Off-Site Dose Calculation Manual and Process Control Program

4.4-4 Millstone Unit No. 1, Docket No. 50-245, Annual Radiological Environmental Operating Report

4.5 SOLID WASTE MANAGEMENT SYSTEM

The Millstone Unit No. 1 solid waste management system is capable of handling, processing and transferring wet solids, spent resin slurry, filter sludge slurry and cleanup filter slurry to high integrity containers or a mobile solidification system. The solid waste operators contract the de-watering and solidification of wet solids to outside vendors.

The plant has no capability for processing concentrated waste solutions to a solidified product. These services are contracted to vendors.

Dry Activated Waste (DAW) is processed and stored in appropriate boxes 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 the solid waste management system is to provide facilities for processing, packaging and handling solid dry and wet wastes, and to allow for radioactive decay and/or temporary storage prior to shipment off site and subsequent disposal.

The solid radwaste system is not a safety-related system. However, the system design ensures compliance with the following regulations and Regulatory Guides:

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

(2) 10 CFR 50, Appendix I, See Reference 4.5-1

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

(4) 10 CFR 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 system is designed to accommodate the following radioactive wastes, which are typical for BWR power plants:

(1) 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 boxes 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.

(2) Wet solid wastes, which consist of filter sludges and spent resins - filter and resin sludges are typically collected in a tank, dewatered and decanted in the liquid radwaste facility and periodically routed to the solid radwaste facility. The wet solid wastes are pumped to a high integrity container or solidification container for shipment and disposal. When the transfer is complete, the dewatering fill head is used to dewater the wet solid waste, or a vendor solidification or dewatering service is used to prepare the waste for off-site shipment.

(3) Concentrated liquid wastes for solidification - vendors supply equipment to the site, receive the concentrated waste solution or slurry, and solidify it in containers suitable for off-site waste disposal.

Summaries of solid waste shipments, types, volumes, and radionuclide composition are given in Reference 4.5-2.

4.5.3 References

4.5-1 Millstone Unit No. 1, Docket No. 50-245, Demonstration of Compliance with 10 CFR 50, Appendix I

4.5-2 Millstone Nuclear Power Station Unit No. 1, Docket No. 50-245, Annual Radioactive Effluents Report

4.6 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING

4.6.1 Design

4.6.1.1 Design Basis

The process and 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, and Regulatory Guides (RG) 1.21, 1.97, 4.15 and 8.8.

The system functional requirements are:

(1) Monitor the room air at the refueling floor area to provide prompt indication of a gross release of radioactive material.

(2) Monitor the radiation level in the radioactive liquid discharge effluents. If the setpoints are exceeded, close the valves in the discharge line.

(3) Monitor the radiation level of the reactor building closed cooling water (RBCCW) system, and provide an alarm if the setpoints are exceeded.

(4) Monitor the radiation level of the reactor building service water system, and provide an alarm if the setpoints are exceeded.

4.6.1.2 System Design Description

The system consists of local radiation detection equipment and assemblies about the plant with indication, recording, alarms and trips displayed in the main control room. The system also includes the sampling stations needed to obtain samples for instrument calibration and verification purposes at specified locations. These systems, as described below, are summarized in Table 4.6-1

4.6.1.2.1 Refuel Floor Area Monitors

The reactor building ventilation includes refuel floor area monitors with capability to continuously measure the radioactivity in the building and provide continuous indication in the main control room. High-radiation levels or loss of signal failures are alarmed in the main control room.

Two G-M monitoring channels located on the refueling floor also monitor the building air above the fuel pool. These channels provide continuous indication of the gross radioactivity levels in the building or exhaust ducts. Refueling floor monitors provide indication only.

The high level setpoints in the reactor building are set sufficiently above background radiation level to avoid spurious alarms, but low enough to respond to the radiation level resulting from abnormal conditions. The trip logic will actuate alarms based upon one upscale or two downscale trips.

The monitors at the refueling floor provide indication of a refueling accident and are set to alarm at less than or equal to 100 mr/hr on high radiation.

4.6.1.2.2 Stack Gas Monitors

The stack gas radiation monitor is designed with the capability to monitor, indicate and record the discharge of gaseous radioactivity from the plant. Capability for sampling of halogen and particulate activity is provided. Annunciation in the main control room occurs if setpoints are exceeded.

Although the monitor cannot determine the individual activity level of the radionuclides in the stack gas, it provides the overall level and a basis for correlation with laboratory analyses of filter and grab sample activities. A sample is drawn from the stack through isokinetic sample probes located approximately two thirds of the way up the 375 foot stack. The General Electric (GE) normal range stack gas monitor is located in the stack gas sample room located in a building at the base of the stack. The sample flows through a two-part assembly. One part is an particulate filter and the other an activated charcoal cartridge for iodine collection. The sample continues into the two shielded radioactive gas detection chambers (fixed volume) and then returns to the stack via the sample pump. The in-line particulate and charcoal filters are periodically removed for detailed radiological quantitative analysis.

Readout consists of a seven decade meter display and a two-pen recorder. The scale is calibrated in counts per second.

4.6.1.2.3 Process Liquids Monitors

The process liquid monitoring system monitors, measures and records the activity levels in the major process streams. The monitors will alarm in the main control room when radiation levels reach the setpoints.

The following process and effluent streams are monitored by one channel of scintillation counter instrumentation:

(1) RBCCW
(2) Reactor building service water system
(3) Radioactive waste building liquid waste discharge water to environs

At each installation, the detector is located in a shielded sampler that is positioned on a section of the process liquid piping or on a side taken for monitoring purposes. If radiation is detected in the RBCCW or service water systems by the monitors, an alarm will alert operators that a leak from the processing system carrying radioactive fluids may have occurred, so that timely corrective actions can be taken.

Liquid radioactive waste is discharged from the station via the liquid effluent line to the discharge canal and to Long Island Sound. These liquids are combined with the contents of the plant discharge quarry such that releases will result in concentrations that are less than allowable limits at the plant discharge. The radioactive waste monitor on this discharge line will alarm if the established high-high radiation setpoint is exceeded and close the discharge valve in the line, prior to reaching allowable release limits.

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 areas where radioactive material may be present, stored, handled, or inadvertently introduced. The system also provides information concerning radiation at selected locations within the plant.

4.6.2.2 System Description

The area radiation monitoring system detects, measures, and indicates ambient gamma radiation dose rates at various locations in the plant. It provides audible and visual alarms in the Main 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.

Each area radiation monitoring channel consists of a sensor and a converter unit as follows:

(1) A gamma-sensitive detector.

(2) A dc log radiation monitor complete with fail-safe operational alarm, electrical test circuitry, appropriate high and low voltage power supplies and control and trip contacts. All channels are provided with a calibration unit and an input into a multi-point strip chart recorder.

Each channel has a high and low instrument trouble trip alarm adjustable over the entire scale. These trip functions operate indicator lights that "seal in" on trip and require a manual reset. These trips also operate annunciators in the Main Control Room.

Channels where local alarm on high level trip is desired are provided with an auxiliary unit which provides for the local alarm and 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 Nos. 1 and 2, Post TMI Requirements - Response to NUREG-0737," Docket Nos. 50-213, 50-245, 50-336.

TABLE 4.6-1

PROCESS AND EFFLUENT RADIATION MONITORS

Monitor	Detector	Range	Trip Function
Stack Gas-Normal	(2) Gamma Scintillator	0.1 to 10^6	None
Stack Gas-High	(2) G-M	Approx. 10^{-3} to 10^5 uCi/cc (Xe 133)	Isolates Normal Range Monitor and Starts Auxiliary Pump when Stack Activity exceeds 10^{-1}uCi/cc
Refuel Floor Area Radiation Monitors	(2) G-M	1 to 10^4 mR/hr 1.0 to 10^3 mR/hr	None
Liquid Waste Effluent	(1) Gamma Scintillator	0.1 to 10^6 cps	Isolates Discharge Valve
RBCCW	(1) Gamma Scintillator	0.1 to 10^6 cps	None
Service Water	(1) Gamma Scintillator	0.1 to 10^6 cps	None

TABLE 4.6-2

AREA RADIATION MONITORING SYSTEM
SENSOR AND CONVERTER LOCATIONS FOR MILLSTONE UNIT NO. 1

STATION NUMBER	SENSOR AND CONVERTER LOCATION	RANGE mR/hr
	REACTOR BUILDING	
1	West Refuel Floor	$0.01\text{-}10^2$
2	West Refuel Floor Hi Range	$10.0\text{-}10^6$
3	East Refuel Floor	$0.01\text{-}10^2$
	RADWASTE BUILDING	
24	Radwaste Building control Room	$0.01\text{-}10^2$
25	Filter Sludge Pump Cubicle	$0.01\text{-}10^3$
26	Decon. Solution Pump cubicle	$0.01\text{-}10^2$
27	Sample Pump Area (Radwaste0	$0.01\text{-}10^2$
28	Radwaste Storage Building	$1.00\text{-}10^4$
29	Radwaste Storage Building	$1.00\text{-}10^4$
30	Radwaste Storage Building (Compactor)	$0.10\text{-}10^3$
	OTHER	
36	Stack Sample Room	$10.0\text{-}10^6$

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CHAPTER 5

ACCIDENT ANALYSIS

5.1 INTRODUCTION

In July of 1998, Northeast Nuclear Energy Company (NNECO) certified to the NRC that Millstone Unit No. 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 No. 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 Main 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 <u>References</u>

5.1-1 "Millstone Nuclear Power Station, Unit No. 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 No. 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.

The 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:

<u>Regarding the impactor movement and the target:</u>

- 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.

<u>Regarding the impact mechanism transmission:</u>

- 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 No. 1 spent fuel pool for the analyses performed to date.

5.2.2 Radiological Consequences

Since NNECO has certified to the NRC that there is a permanent cessation of operations of Millstone Unit No. 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 (References 5.2-2 and 5.2-3). 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

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Guide 1.25. Doses were calculated using the TACT-III, ORIGEN-2, and ELISA computer codes. Ch. 3

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 No. 1 and Millstone Unit No. 2 control rooms. The results of this dose assessment is as follows: Ch. 3

Thyroid dose to the Millstone Unit No. 1 Control Room	9.37E-02 REM
Thyroid dose to the Millstone Unit No. 2 Control Room	7.65E-02 REM
Whole-body dose (calculated as TEDE) to the Millstone Unit No. 1 Control Room	1.07E-01 REM
Whole-body dose to (calculated as TEDE) the Millstone Unit No. 2 Control Room	8.67E-02 REM
Beta skin dose to the Millstone Unit No. 1 Control Room	2.66E+01 REM
Beta skin dose to the Millstone Unit No. 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 No. 3 control room since the atmospheric dispersion factor (X/Q) is approximately 50 times less that the X/Q to the Millstone Unit No. 2 control room. Therefore, the dose to the Millstone Unit No. 3 control room would be approximately 50 times less than the Millstone Unit No. 2 control room dose. Ch. 3

5.2.3 References

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

5.2-2 Calculation No. XX-XXX-44RA, "Radiological Assessment of a Fuel Handling Accident at MP-1," Revision 1, October 21, 1998.

5.2-3 Calculation Package NUC-197, "MP1 Defueled State - Radiological Analysis of a Fuel Handling Accident," Duke Engineering and Services, October 11, 1999. Ch. 3

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: • 85 percent Elemental • 15 percent Organic	Regulatory Guide 1.25 See Ref. 5.2-3.
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: • 30 percent Noble Gases • 12 percent Iodines	Regulatory Guide 1.25 & conservative assumption
9.	• No credit taken for secondary containment • SGTS not in operation • Puff release is an unfiltered ground release	Technical Specifications
10.	Breathing rate = 3.47×10^{-4} m^3/sec	Regulatory Guide 1.25
11.	Ground level dispersion factor (X/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 No. 50-245
12.	Decay Time for fuel = 3.8 years	Based on the MP1 shutdown on November 4, 1995.

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CHAPTER 6

CONDUCT OF OPERATIONS

6.1 ORGANIZATIONAL STRUCTURE

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

The majority owners, holding 100 percent of the Millstone Unit No. 1 nuclear plant, are the Northeast Utilities System (NU) subsidiaries; Connecticut Light and Power Company (CL&P), and Western Massachusetts Electric Company (WMECO).

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 No. 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 Northeast Utilities Quality Assurance Program (NUQAP) 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 10CFR50.59 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 Northeast Nuclear Energy Company (NNECO), a NU company, 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 Northeast Utilities Quality Assurance Program (NUQAP) Topical Report

Northeast Utilities has developed and implemented a comprehensive Quality Assurance Program to ensure conformance with established regulatory requirements as set forth by the Nuclear Regulatory Commission (NRC), and accepted industry standards. The participants in the NUQAP 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 NUQAP 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 (SAR).

This NUQAP is also established, maintained and executed with regard to Radioactive Material Transport Packages as allowed by 10 CFR 71.101(f). In addition, the NUQAP 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 Nos. 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 Nos. 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 NUQAP Topical Report (Reference 6.1-1).

6.5.2 Independent Review

Independent review of activities affecting the unit's safety is performed by the Nuclear Safety Assessment Board is as described in the NUQAP 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 NUQAP Topical Report (Reference 6.1-1).

CHAPTER 7

DECOMMISSIONING

7.1 SUMMARY OF ACTIVITIES

Millstone Unit No. 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 Northeast Utilities Board of Trustees 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 10CFR50.82 (a)(1)(i) & (ii), was filed on July 21, 1998 [Reference 7.1-1], at which time the 10CFR50 license no longer authorized operation of the reactor or placement of fuel in the reactor vessel.

The mission of Northeast Nuclear Energy Company (NNECO) 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 No. 1 decommissioning activities be delayed and performed concurrently with the decommissioning of Millstone Unit Nos. 2 and 3. Other considerations may dictate early scheduling of certain decommissioning activities. Therefore, the approach to decommissioning Millstone Unit No. 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 No. 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, NNECO 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 No. 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).

Under any eventuality such as unavailability of a LLW disposal site, temporary shortfall in decommissioning funding, or other unforeseen circumstances, 10CFR50.82 requires NNECO to maintain the capability to suspend decontamination and dismantlement.

7.1.1 Decommissioning Approach

NNECO is planning on decommissioning Millstone Unit No. 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 is constructed and the transfer of spent fuel from the spent fuel pool (SFP) is completed before a SAFSTOR period. The SAFSTOR period ends with decontamination and dismantlement of any remaining systems, structures, and components commence in coordination with Millstone Unit No. 2 and Millstone Unit No. 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 and the 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

Planning includes the preparation of licensing and design basis change documents and the post shutdown decommissioning activities report (PSDAR)[Reference 7.1-3]. Additionally, 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.

General planning and preparation for decommissioning include the following activities:

- Review and revise plant licensing basis documents as necessary, consistent with cessation of power operations. These documents include the defueled safety analysis report and the technical specifications.

- Develop a decommissioning organizational structure and select project staff.

- Identify the Millstone Unit No. 1 systems shared by Millstone Unit Nos. 2 and 3 and revise the designs and the operation of these systems to isolate Millstone Unit No. 1 from Millstone Unit Nos. 2 and 3.

- Review and reclassify systems, structures, and components consistent with cessation of power operations.

- Review and revise plant programs and procedures as necessary to be consistent with cessation of power operations.

- Prepare a plan for the SFP cleanup.

- Design and implement a SFP cooling system which is isolated from the remainder of the plant.

- 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 is undertaken during which radiological, regulated and hazardous wastes are identified, categorized, and quantified. Surveys are conducted to establish the contamination and radiation levels throughout the Millstone Unit No. 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 No. 1 may be performed, although a detailed survey of the environs would likely be deferred pending decommissioning of Millstone Unit Nos. 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 is undertaken as a part of the site characterization. Using the results of this analysis, these components are classified in accordance with 10CFR61 and form the basis for the detailed plans for their packaging and disposal. Material which is found to be greater than Class C (GTCC) is stored with the spent fuel or in an ISFSI.

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 10CFR50.2 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 10CFR61.55."

The initial major decommissioning activities are the removal of the drywell head and removal of the reactor vessel internals by segmentation. The drywell head is sectioned and sent to a metal processor. The internals comprising the core shroud, core support structure, fuel guide plate, and upper portions of the control rod guide tubes may be GTCC waste which is segmented, packaged into fuel bundle sized containers, and transferred to the SFP or ISFSI for storage and eventual disposal with the fuel. Using this approach all internals are packaged and disposed of independent of the reactor vessel. When the internals segmentation effort is completed, the reactor vessel is drained and any remaining debris removed. Without the internals present, several options are available for later removal and disposal of the reactor vessel: segmentation, sectioning into larger 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. PSDAR
    2. A change to technical specifications reflecting the defueled condition
    3. A detailed site-specific decommissioning cost estimate pursuant to 10CFR50.82
    4. A license termination plan pursuant to 10CFR50.82
    5. A spent fuel management program, pursuant to 10CFR50.54(bb)

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

- Millstone Unit No. 1 systems shared with the other Millstone units are separated by modification or reconfigured to permit operation by Millstone Unit Nos. 2 and 3.

- 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 No. 1 and Millstone Unit No. 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 No. 1 license termination is constrained by the need to terminate the Part 50 license coincident with that of Millstone Unit No. 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.

NNECO is required to prepare a License Termination Plan (LTP) for Millstone Unit No. 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 10CFR20 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 No. 1 area of the Millstone site will be undertaken when the 10CFR Part 50 license for Millstone Unit No. 1 is terminated. This event may coincide with Millstone Unit Nos. 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 No. 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. NNECO estimates the low-level waste burial volume for Millstone Unit No. 1, will be at or below this value for the modified SAFSTOR alternative. NNECO'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, NNECO estimates that the volume for Millstone Unit No. 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 No. 1 plant. The current DOE estimate for startup of the federal waste management system is 2010. For planning purposes, NNECO 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. NNECO 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 No. 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 NNECO 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 NNECO 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, NNECO 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 NNECO will be at or below this value.

7.2.1 <u>Nuclear Worker</u>

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 wold 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.

NNECO 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 use 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 RADIOATION 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 No. 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 NUREGCR-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.

NNECO 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 No. 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.

Attachment 2 to B18128
Millstone Unit 1
Safety Evaluation Summaries

Safety Evaluation Number: S1-EV-98-0018
Document Number: TMOD1-98-006
Title: Installation of Temporary EDG Fuel Oil Storage Tank

Description of Activity:
This temporary modification will provide new above ground fuel oil storage for the EDG to support Unit 1 spent fuel pool (SFP) cooling and the Unit 2 Appendix R and SBO programs. Unit 1 will abandon the EDG underground storage tanks in place and fill the tanks with an inert solid material. The associated fuel transfer piping will be cut and disconnected from the remaining system.

Reason for Activity:
The existing underground fuel oil storage tank must be emptied prior to exceeding thirty years life as required by the State of Connecticut Department of Environment Protection.

Safety Evaluation Summary:
The proposed change maintains the ability of the EDG to support the safety functions required with the plant permanently defueled. Specifically, the capability of the EDG to support SFP decay heat removal assuming the limiting single active failure, (a Loss of Normal Power) is maintained, since the EDG can supply the necessary onsite AC power for 5 days using onsite fuel inventory only. The change in quantity of onsite fuel simply reflects the reduced EDG loading since the ECCS loads are no longer required. In addition, power is also available to supply SFP cooling from the Unit 2 EDG via bus 14H. The large thermal capacitance of the SFP also provides a significant time buffer for operators to take action to mitigate multiple beyond design basis failures.

For a Unit 2 Appendix R event, the proposed change ensures that the Unit 1 EDG can supply the necessary power to Unit 2 via the bus 14H crosstie while maintaining the Unit 1 SFP temperature less than 140 F for the required period of 72 hours. For a Unit 2 station blackout scenario, the Unit 1 EDG can supply the necessary power for a period in excess of the eight hours. The proposed change is safe and does not constitute an Unreviewed Safety Question

Safety Evaluation Number: S1-EV-99-0001
Document Number: SPROC EN99-1-1
Title: Emergency Diesel Generator Governor Droop Adjustment

Description of Activity:
SPROC EN99-1-1 has been written to determine the correct Emergency Diesel Generator governor droop setting.

Reason for Activity:
Previous testing performed under the control of SPROC 97-1-35 shows that the current governor droop setting is 1.5%. Industry standards and Woodward Governor Co. recommendations dictate that this setting should be between 3 and 5%. Failure to have the governor droop adjusted correctly causes the operator to have poor control when loading and unloading, and makes the EDG vulnerable to load changes as a result of grid frequency changes, with the risk of overload.

Safety Evaluation Summary:
This test will be performed with the plant in the "COLD SHUTDOWN" mode and with all fuel in the Spent Fuel Pool. The EDG will not be "OPERABLE" during the test. In this operating mode, there is no requirement for the safety bus to be re-energized in 10 seconds following a Loss of Normal Power. However, in the event that this procedure is performed while MP2 is in Modes 1 or 2 and the requirements of the station blackout program apply, the operators must be able to restore power within one hour.

During the performance of this procedure, the EDG will be considered inoperable and will not be credited for the mitigation of a cold shutdown accident. The off-site power sources will be monitored in accordance with the requirements of OM 2, "Shutdown Risk Management," and OP 240 "Conduct of Outages". The availability of those offsite power sources is not affected by this test. The EDG can be restored from testing within 1 hour in the event of a Loss of Normal Power. The Unit's (and Station's) ability to respond to an LNP event is not affected by this test. Therefore, performance of this procedure is concluded to be safe and does not involve an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0002
Document Number: SPROC OP99-1-02
Title: Emergency Diesel Generator Automatic Voltage Regulator Droop Adjustment

Description of Activity:
SPROC OP99-1-02 was written to provide steps for the adjustment of the Emergency Diesel Generator Automatic Voltage Regulator (AVR).

Reason for Activity:
During the performance of SPROC 97-1-35, it was determined that the EDG AVR, is set at 11% droop. Industry standards dictate that this setting should be between 3 and 5%. The effect of a droop setting of 11% is to make the AVR slow to respond to voltage transients, putting unnecessary stress on the generator and power potential transformers and to limit the reactive loading capability of the AVR during surveillance testing.

Safety Evaluation Summary:
This test will be performed with the plant in the "COLD SHUTDOWN" mode and with all fuel in the Spent Fuel Pool. The EDG will not be "OPERABLE" during the test. In this operating mode, there is no requirement for the safety bus to be re-energized in 10 seconds following a Loss of Normal Power. However, in the event that this procedure is performed while MP2 is in Modes 1 or 2 and the requirements of the station blackout program apply, the operators must be able to restore power within one hour.

The change to the AVR droop setting does not affect the response of the AVR when the EDG is operating as a result of an emergency and the adjustment procedure is safe to perform. During the performance of this procedure, the EDG will be considered inoperable and will not be credited for the mitigation of a cold shutdown accident. The off-site power sources will be monitored in accordance with the requirements of OM 2, "Shutdown Risk Management," and OP 240 "Conduct of Outages". The availability of those offsite power sources is not affected by this test. The EDG can be restored from testing within 1 hour in the event of a Loss of Normal Power. The Unit's (and Stations's) ability to respond to an LNP event is not affected by this test. Therefore, this procedure for EDG AVR droop adjustment is safe and does not involve an Unreviewed Safety Question (USQ).

Safety Evaluation Number: S1-EV-99-0003
Document Number: TMOD 1-98-012
Title: Relocation of NaOCl Injection for Service Water Piping

Description of Activity:
This temporary modification reroutes the Sodium Hypochlorite injection from its current position at the suction bells of each of the service water pumps to a single location on the outlet nozzle of the service water strainer. The temporary modification will be implemented by running PVC piping from the Sodium Hypochlorite Room to the intake structure through the common wall into the strainer pit and into the outlet nozzle of the service water strainer. Injecting Sodium Hypochlorite at this point routes the chlorine into the service water strainer header downstream of the service water strainer.

Reason for Activity:
This temporary modification is being implemented to eliminate the inadvertent discharge of chlorinated water via the service water strainer blowdown line, idle service water pump suction, and the service water pump seal water lines. Changing these chlorination paths improves NPDES Permit compliance at Millstone.

Safety Evaluation Summary:
The proposed modification moves the Sodium Hypochlorite injection point from the service water pump suction bells to the service water strainer outlet nozzle. It does not affect the operation or capability of the service water system in this mode of operation. No increase in radiological dose to the public is possible even if the new Sodium Hypochlorite piping catastrophically failed. Hypochlorite injection and service water are neither event initiators nor mitigating systems for any credible accident for the plant in cold shutdown. No increased probability of any accidents or malfunctions or their consequences is possible as a result of this modification. Therefore, this change is safe and does not involve an Unreviewed Safety Question.

Safety Evaluation Number:	S1-EV-99-0004
Document Number:	TRMCR 99-1-1 TE M1-EV-00-0001
Title:	Unit 1 Technical Requirements Manual Section #7.6 "Unit 1 SSCs That Interface With Unit 2"

<u>Description of Activity:</u>
This change adds Section 7.6 to the Technical Requirements Manual (TRM). This section identifies Unit 1 SSCs that are required to support Unit 2 Licensing Bases operational requirements.

<u>Reason for Activity:</u>
Operations Department Manual 1-OM-10.4 lists all Unit 1 SSCs that support Unit 2, however this procedure may be revised without PORC review and approval. Formalizing those SSCs that are required to support the Unit 2 Licensing Bases operational requirements in the TRM will ensure that all changes receive the appropriate level of review prior to being changed.

<u>Safety Evaluation Summary:</u>
This safety evaluation concludes that no Unreviewed Safety Question is involved and that the change is safe. The change involved, an addition to the Unit 1 Technical Requirements Manual and its associated Technical Evaluation, to identify operational aspects of SSCs that are required to support the Unit 2 Licensing Bases. The proposed activities do not increase the probability of occurrence or consequences of accidents or malfunctions previously evaluated, or create the possibility of an accident or malfunction of a different type. The proposed activities will not reduce the Margin of Safety as described in the Unit 1 or Unit 2 Technical Specifications. The proposed activities are safe and do not increase the risk to the public.

Safety Evaluation Summary: S1-EV-99-0005
Document Number: ONP 541
Title: Supplying MP2 with Emergency Power

Description of Activity:
This Off Normal Procedure is intended to provide Unit 1 operators with the action steps necessary to supply Unit 2 with emergency power for Unit 2 Appendix R and Station Blackout events. The subject procedure only covers the transfer of power from Unit 1 to Unit 2. As part of this procedure, the fuel pool cooling system may have to be secured to supply adequate power to Unit 2 in an Appendix R scenario for a short period of time.

Reason for Activity:
This procedure replaces existing off normal procedures which addressed operating conditions.

Safety Evaluation Summary:
This evaluation concludes that in the process of executing the steps of this procedure under the applicable controls and conditions in place, no unsafe conditions will exist and no Unreviewed Safety Question is created. Transfer of power from Unit 1 normal and emergency power sources to Unit 2 does not create an unsafe condition for the Unit's irradiated fuel in the spent fuel pool.

Safety Evaluation Number: S1-EV-99-0006
Document Number: DCR M1-99001
Title: MP1 Fire Water Tanks and Recirculation Line Replacement; Domestic Water Tank Demolition

Description of Activity:
This activity involves the following:
- demolition and replacement of the existing Fire Water Tanks M7-6A & 6B
- the removal of the existing 10" recirculation line between Fire Water Tank M7-6A and the Fire Water Pumps and the addition of a new recirculation line between both tanks and the fire water pumps
- the demolition and removal of the abandoned Domestic Water Storage Tank M7-10

Reason for Activity:
Ultrasonic testing of the existing Fire Water Storage Tanks (FWST) has shown degradation and unacceptable wall thinning. The removal of the abandoned Domestic Water Storage Tank will provide additional space for the FWST activities.

Safety Evaluation Summary:
The Fire Water System and the Domestic Water System do not perform any nuclear safety related function. The Fire Water System and Domestic Water System are not capable of, or connected to any system which is capable of acting as an accident initiator.

The Domestic Water Storage Tank is currently isolated from the Domestic Water System. The water supply to Unit 1 is provided via the Unit 2 cross-tie and system surges are controlled through the Unit 2 tank.

Replacement of the Fire Water Storage Tanks M7-6A and M7-6B will not change the flow characteristics from the tanks to the Fire Water Pumps or reduce the available fire water volume. The FWST implementation method will ensure one FWST (with Technical Requirements Manual specified minimum volume of 200,000 gallons) and all three Fire Water Pumps remain available (within TRM requirements) at all times. Site Fire Plan SFP-31 will be invoked during implementation as the backup Fire Water System to maintain the Fire Water System available for fire suppression and other backup sources as currently defined in the SAR.

This Safety Evaluation finds that the replacement of the Fire Water Tanks, modification to the fire water recirculation piping, and the demolition and removal of the Domestic Water Storage tank are safe and do not represent an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0007
Document Number: TMOD 1-99-3
Title: Installation of Temporary Fire Water Jockey Pump

Description of Activity:
The temporary modification will install an electric driven temporary pump to take suction from the "B" fire water tank (at tank drain valve 1-FIRE-56B) and discharge to the fire water distribution system (at hydrant 2-FIRE-7). The pump will maintain fire protection system pressure, during the time that the jockey pump is removed from service for fire system piping replacement.

Reason for the Activity:
During the replacement of the suction piping from the "B" fire water tank, the electric fire water jockey pump (M7-11) will not be able to take suction from the "A" Fire Water Tank. The installation of the portable pump will maintain the system pressurized, which will prevent unnecessary cycling of the main fire water pumps.

Safety Evaluation Summary:
The temporary pump installation will maintain the fire water system pressurized, and so will prevent unnecessary cycling of the main fire pumps to compensate for normal system leakage. During actual system demand, such as that which is required to supply one of the fire suppression systems, at least one of the main fire water pumps would start on low distribution header pressure to provide the necessary system flow. The jockey pump is interlocked off when either of the MP1 main fire pumps start, and therefore never functions as a sole source of fire water.

A malfunction in which the temporary jockey pump fails to function would result in unnecessary cycling of the main fire water pumps; however, the fire water system would still remain capable of supplying fire water and cooling/makeup water for those systems which utilize fire water as a backup source. The ability of the fire water system to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located will not be compromised by this temporary installation.

A malfunction in the pressure boundary of the temporary jockey pump would not result in loss of the fire water system, since the temporary installation is between the inoperable "B" fire water tank and a system hydrant. The loop hydrant will be equipped with an FPQA check valve to prevent backflow from the hydrant if the temp mod connections fail. This will prevent any substantial flow of fire water out of any breach in the temporary jockey pump pressure boundary.

The pump being utilized will be equipped with a relief valve, which will prevent over pressurizing the fire water system piping or other components. The temporary installation was also reviewed to ensure that the equipment to be located adjacent to the fire water tanks, does not impose adverse affects on safety related equipment.

The temporary installation is safe, and does not result in an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0008
Document Number: FSARCR 99-MP1-7
Title: Update of the Fire Protection Organization in Chapter 9, Remove Job Titles in Chapter 13

Description of Activity:
This proposed change to the UFSAR rewrites Section 9.5.1.5, "Fire Protection Organization," deletes Figure 9.5-1, "Fire Protection Organizational Chart," removes job titles and organizations' names from Section 13.4.3, "Audits," and removes job titles and revises organizational functions from Section 13.5, "Plant Procedures," and Section 13.5.2, "Operating and Maintenance Procedures."

Reason for Activity:
Information contained in the cited FSAR Sections is not consistent with respect to the Fire Protection Manual, the NUQAP Topical Report, and Technical Specifications.

Safety Evaluation Summary:
The proposed changes in the subject FSARCR related to Fire Protection are consistent with all aspects of the Millstone fire protection program, including the Fire Protection Program Manual and the Fire Protection Program Procedure WC7. The proposed changes in the subject FSARCR related to Chapter 13 of the UFSAR are administrative in nature and are consistent with the details in the licensing basis organization. The changes do not constitute an Unreviewed Safety Question and are Safe. The changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire because of Unit 1's permanently defueled condition and certification to no longer operate.

The malfunctions evaluated involve personnel associated with responsibilities in the fire protection organization which could only have an indirect affect on SSCs important to safety and the defueled condition. The probability of occurrence and consequences of these types of malfunction would not increase because these changes are administrative in nature and are already incorporated in the SAR by reference of the Fire Protection Program Manual. For similar reasons, the proposed changes would not create a different type of malfunction important to safety than previously evaluated in the SAR.

The scope of the proposed changes also has no affect on the Unit 1 fire hazards analysis or upon any Unit 1 accident analysis. Implementation of the proposed changes, therefore, would not increase the probability or consequences of accidents discussed in the SAR, and would not create the possibility of an accident of a different type than discussed in the SAR.

Safety Evaluation Number: S1-EV-99-0013
Document Number: MSEE DCN DM1-00-0061-99
Title: Rework Section of Screen wash Supply to E Bay Trough

Description of Activity:
The Design Change Notice (DCN) provides the necessary technical details needed for the replacement of the ¾" backwash pipe nipple and fitting leading to screen wash E north. In addition, line numbers are being assigned to other pipe nipples to facilitate processing of the current as well as future work orders as necessary.

Reason for Activity:
The subject DCN provides the necessary technical details needed for the replacement of the ¾" backwash pipe nipple and fitting leading to screen wash E north trough which is corroded to the point that it is leaking. The proposed change affects the Unit 1 UFSAR by virtue of having line numbers assigned to the screen wash ¾ in. pipe nipples shown on P&ID 25202-26015 sh. 8 which is included in Unit 1 UFSAR as Fig. 09.2-01 sh. 6. The line numbers are being added to facilitate processing of the current, as well as the future work orders, as necessary.

Safety Evaluation Summary:
The replacement piping meets the original material requirements, follows the same routing and does not perform any safety function. The proposed replacement of the ¾" backwash pipe nipple and fitting leading to screen wash E north and assignment of the new line numbers to other pipe nipples in the screen wash system is safe and does not constitute an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0014
Document Number: N/A
Title: Clearance to allow the "Diesel Generator LOCA Start Bypass" switch to be transferred from the "NORMAL" position to the "BYPASS" position

Description of Activity:
A clearance will be written to allow the "Diesel Generator LOCA Start Bypass" switch to be transferred from the "NORMAL" position to the "BYPASS" position. Additionally, the clearance will track the switch position change while MP1 remains in the defueled condition.

Reason for Activity:
This change will block a spurious LOCA signal from starting the EDG. In the current mode of operation, with the reactor defueled, low pressure core and containment cooling subsystems are not required to be functional. Therefore, no genuine LOCA start signal can be initiated by this instrument loop.

Safety Evaluation Summary:
A clearance will be issued to change the position of the "Diesel Generator LOCA Start Bypass" switch from the "NORMAL" position to the "BYPASS" position. Start signals caused by instrument drift or similar erroneous conditions will be blocked from initiating an unplanned EDG start. Since unplanned starts are not preceded by activities to prelube the engine bearings, potential wear can be reduced and availability increased by eliminating the possibility of spurious start signals.

The EDG LOCA start circuit is provided to send a start signal to the EDG, should LOCA conditions occur. Thus the EDG would be ready, operating unloaded, at rated voltage and frequency, in the event of a subsequent loss of normal power. Under the current plant operating conditions, with the reactor defueled, no genuine LOCA start signal can be generated; therefore, preventing the EDG from starting as a result of a LOCA signal, cannot have any impact on the safe operation of the plant.

In view of the above, it is concluded that this change is safe and does not constitute an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0016
Document Number: TMOD 1-99-005
Title: Temporary Removal of East and West Fuel Prep Machine Mechanical Stops

Description of Activity:
The Temporary Modification will measure the existing placement of the Fuel Prep Machine mechanical stops and then remove them to allow recovery of unirradiated fuel assemblies. An additional air supply isolation valve will be mounted on the operator platform of both Fuel Prep Machines.

Reason for Activity:
This modification is required to support the special procedure needed to remove the unirradiated fuel assemblies currently stored in the Unit 1 Spent Fuel Pool. The modification will bypass the minimum depth limit by removing the mechanical stops from the East and West Fuel Preparation Machines. The removal of these stops will allow the Fuel Prep Machine to be used in the new fuel mode of operation.

The additional air supply will give the operator the ability to rapidly stop the Fuel Prep Machine in the event of a directional control valve malfunction.

Safety Evaluation Summary:
The process of removing the new fuel will utilize existing radiological controls and special nuclear material checks to prevent the unintentional removal of an irradiated fuel assembly. This temporary modification supports the project by removing the mechanical stops and then by installing a redundant air isolation valve. In the event of an equipment malfunction, such as a directional control valve failure, the carriage can be stopped and corrective actions evaluated.

Prior to implementing the new fuel recovery process, administrative tagging will limit the use of the "elevator" to unirradiated fuel assemblies by warning operators that the stops have been removed. Using existing administrative controls for the removal of new fuel will preclude the possibility of removing material other than new fuel. The stops exist and are removable by design to provide positive control and require a planned evolution to enable the "elevator" to go from the spent fuel mode to new fuel mode of operation. This is safe and does not constitute an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0017
Document Number: OP 310
Title: Fuel Pool System

Description of Activity:
The proposed change to procedure OP 310, "Fuel Pool System" accomplishes the following objectives:

- Provides guidelines allowing the spent fuel pool cooling system to be shutdown for controlled outages in order to perform maintenance on the common spent fuel pool cooling system valves and components. The procedure will instruct the operators to monitor fuel pool temperature and level during system outages and restore the system prior to reaching 125F.
- Provides the valve lineup for additional sources of available make-up water to the spent fuel pool to replace evaporative losses.
- Provides consistent methods of swapping fuel pool cooling water pumps by first throttling the pump discharge valve prior to starting the cooling water pump.
- Performs writer's guide upgrades and format enhancements.

Reason for Activity:
Procedure OP 310, "Fuel Pool System," is being revised to provide the guidance for securing the spent fuel pool cooling system. The current UFSAR system description states that the SFP cooling system is continuously operated to support decay heat removal from irradiated fuel assemblies. There is a desire to allow greater flexibility in securing forced cooling based on the lower decay heat loads which now exist in the SFP.

Safety Evaluation Summary:
The licensed spent fuel pool heat load for a maximum normal end-of-cycle full core offload is 22.54 x 10^6 BTU/hr based on a maximum possible inventory of 4412 fuel assemblies. However, following permanent cessation of operations and the extended shutdown, the actual spent fuel pool decay heat load was calculated to be 1.781 x 10^6 BTU/hr as of January 1999, based on the decay heat produced from the current inventory of 2885 fuel assemblies and 1 partial assembly consisting of 19 fuel rods.

With this heat load, the time to heat up the bulk pool water from 100F to 140F is 2 ½ days under worst case conditions.

The proposed change does not perform any physical modifications to the system. The spent fuel pool cooling system is designed for a single active failure without the loss of the system's function. Since the system is comprised of manual valves and pumps controls, operator actions are necessary to restore the system function following the single failure. Therefore, with the system manually secured, operator action is still required to restore the system when cooling is necessary.

When activities such as maintenance on common isolation valves or system testing require spent fuel pool cooling system outages, operators will continue to have the reliable means of detecting a rise in spent fuel pool temperature or a reduction of pool inventory. This change is safe and does not constitute an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0018
Document Number: N/A
Title: Cross Contamination of Millstone Unit 1 RBCCW System

Description of Activity:
The Unit 1 reactor building closed cooling water (RBCCW) system is contaminated with Cs-137 with about 9E-7 μCi/ml of radioactivity concentration. NRC I&E Bulletin 80-10 requires a safety evaluation for the operation of any nonradioactive system which becomes contaminated. Given the concentration and total inventory of contamination, and the potential for routine or accidental releases of radioactivity to the environment, the evaluation must document that the system can be safely operated.

Reason for Activity:
Minor leaks into the RBCCW system from the RCS have occurred changing it from a non-radioactive to a contaminated system. Although this system was designed as a nonradioactive system, a system radiation monitor was installed because of the possibility of contamination and the potential for release to the environment with a subsequent leak to the service water system (UFSAR Section 9.2.3).

Safety Evaluation Summary:
The Unit 1 RBCCW system is recognized in UFSAR Section 9.2.3.2 as having the potential to become contaminated, but by design was intended to be operated as non-contaminated. Based upon the existence of a system rad-monitor with alarm set-points above current system background contamination levels, routine weekly Chemistry monitoring for changes in these levels, and Operations procedures in place to quickly apply appropriate corrective actions in the event of increased contamination, operating the RBCCW up to or achieving a level of 1E-5μCi/ml is safe.

Continuing operation of the RBCCW as a contaminated system is safe and is not an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-0019
Document Number: TMOD 1-99-2
Title: Diesel Generator Service Water Annubar Temporary Abandonment

Description of Activity:
This temporary modification replaces the flow annubar which measures Service Water (SW) flow to the MP1 Emergency Diesel Generator. This annubar will be replaced with a blind flange.

Reason for Activity:
The annubar needs to be replaced, because it has a through-wall leak which adversely affects the availability of the EDG. No direct replacement annubar is available so a blind flange will be put in place of the annubar to maintain Service Water system integrity and EDG reliability.

Safety Evaluation Summary:
The installation of this temporary modification to replace the annubar which measures SW flow to the EDG with a blind flange is safe. There are no malfunctions added by this installation and no increase in the consequences or likelihood of existing malfunctions. The replacement blind flange is of the same quality level and pressure rating as the annubar and the SW piping and the flange is lighter than the annubar which it replaces. Therefore, there are no seismic concerns and no increased chance of a leak in the SW system. No accidents are prevented or mitigated by the SW system and no new accidents are introduced by this replacement. The SW annubar was installed when the plant was operating to enhance the operability of the EDG by allowing SW flow to be throttled to ensure adequate flow was maintained to the EDG. It is not used in the current permanent shutdown condition to take any action since the cooling loads on the SW system are substantially reduced from when the plant was operating or preparing to operate. This change is safe and does not constitute an Unreviewed Safety Question.

Safety Evaluation Number:	S1-EV-99-0020
Document Number:	OD MP1-0005-97
Title:	Elevated Fuel Bundles in Spent Fuel Pool Racks due to Misaligned Channel Fasteners

Description of Activity:

This safety evaluation appraises (1) the current configuration of the fuel assemblies that are not fully seated in their storage racks, (2) the evolution of removing unirradiated fuel assemblies from the Spent Fuel Pool with respect to the raised fuel assemblies, (3) the ability to manipulate the raised fuel assemblies in order to fully seat them, and (4) the ability to move irradiated fuel assemblies with respect to raised assemblies in the spent fuel pool.

The existing Operability Determination, NRC commitment, and procedure are being revised to state:

> *"No unaffected, irradiated fuel assemblies will be moved within the Spent Fuel Pool until the raised fuel assemblies have been fully seated. In the event that an assembly cannot be fully seated, no unaffected, irradiated fuel assemblies will be moved until it is determined that either the assembly configuration has been previously evaluated, or an evaluation of the configuration is performed".*

Reason for Activity:

This safety evaluation addresses the removal of the un-irradiated fuel assemblies from the Spent Fuel Pool while the raised fuel assemblies are present, and the reseating of the raised fuel assemblies.

Safety Evaluation Summary:

This safety evaluation addresses two aspects of the Spent Fuel Pool with respect to the raised fuel assemblies:

- Does the current configuration of the Spent Fuel Pool with the raised fuel assemblies involve an unreviewed safety question.
- Does the activity of moving fuel assemblies while the raised fuel assemblies in the Spent Fuel Pool exist involve an unreviewed safety question.

Based on the conclusions of this safety evaluation, the current configuration of the Spent Fuel Pool with respect to the raised assemblies, and the movement of fuel with respect to the raised fuel assemblies is safe and is not an unreviewed safety question.

Safety Evaluation Number: S1-EV-99-0021
Document Number: SPROC MNT 99-1-02
Title: Head Insulation Package Disassembly

Description of Activity:
The SPROC MNT 99-1-02, titled "Head Insulation Package Disassembly" was written to control the maintenance activities necessary to disassemble and dispose of the Unit 1 reactor vessel head insulation package.

Reason for Activity:
The disassembly and disposal of the reactor vessel head insulation package is necessary to create working space on the refueling floor in support of the removal of unirradiated fuel from the site and the various decommissioning efforts related to the refueling floor and spent fuel pool.

Safety Evaluation Summary:
The proposed disassembly of the head insulation package is both safe and not an Unreviewed Safety Question. The basis is that the head insulation package is not required in the present defueled mode of operation and it is unlikely that any operation will be conducted in the future. In addition, a method of tracking the disassembly of the head insulation package has been initiated such that restoration is assured if the system were to be made operable. The work activity itself will be conducted adjacent to the spent fuel pool in a fashion that will preclude the introduction of any objects into the spent fuel pool that would challenge the existing safety analysis. Loads related to this work being carried by the reactor building overhead crane are not allowed over the spent fuel pool.

Safety Evaluation Number: S1-EV-99-0022
Document Number: SPROC OPS 99-1-05
Title: SFP Decay Heat Measurement Test

Description of Activity:
This Safety Evaluation addressed one-time procedure SPROC OPS 99-1-05, Spent Fuel Pool (SFP) Decay Heat Measurement Test. This procedure controls the measurement of SFP water temperatures and structural component temperatures as a function of time by securing the SFP cooling system. The SFP will be allowed to increase in temperature from approximately 85°F to a maximum of 105°F at which time data collection will be discontinued and the SFP Cooling System will be restarted in accordance with procedure OP 310, Spent Fuel Pool Operation.

Reason for Activity:
The purpose of SFP temperature data collection is to determine the actual SFP heatup rate prior to designing modifications to the SFP cooling system in support of an independent SFP island.

Safety Evaluation Summary:
This safety evaluation is based on SE S1-EV-99-0017, Fuel Pool System, which modified guidelines for operation of SFP cooling. The SPROC OPS 99-1-05 will be based on the new guidelines in Procedure OP 310.

The proposed SPROC will enable the Unit 1 Spent Fuel Pool Cooling system to be shutdown for the collection of temperature data.

The proposed SPROC does not perform any physical modifications to the system. The spent fuel pool cooling system is designed for a single active failure without the loss of the system's function. Since the system is comprised of manual valves and pump controls, operator actions are necessary to restore the system function following the single failure. Therefore, with the system manually secured, operator action is still required to restore the system when cooling is necessary. Operations will retain the same level of control of spent fuel pool inventory and temperature as existed prior to this test.

When activities (such as testing or data gathering) which require spent fuel pool cooling system outages are performed, operators will continue to have the reliable means of detecting a rise in spent fuel pool temperature or a reduction of pool inventory.

This test was determined to be safe and does not constitute an Unreviewed Safety Question.

Safety Evaluation Number: S1-EV-99-023
Document Number: SPROC MNT-99-1-04
Title: Temporary Fuel Pool Filter Operation

Description of Activity:
This safety evaluation evaluates the installation, operation, maintenance, and removal of a temporary filtration system in the MP1 Spent Fuel Pool (SFP). This filtration system is manufactured by Tri-Nuclear corporation and will not be moved over fuel in the Spent Fuel Pool.

Reason for Activity:
The installation of the temporary filtration system in the MP1 Spent Fuel Pool is necessary due to upcoming work in the SFP. The filtration system will maintain the water clarity within the SFP as well as allow for vacuuming activities. The work in the SFP includes removal of the unirradiated fuel assemblies, and processing of components stored on the SFP.

Safety Evaluation Summary:
This safety evaluation assesses the installation, operation, maintenance and removal of a temporary filtration system within the MP1 Spent Fuel Pool. This safety evaluation concludes that this activity does not constitute an unreviewed safety question and is safe. The temporary filtration system will be located in the Northeast corner of the SFP suspended by slings from a support beam which bridges the corner of and rests on top of the SFP. The suspension method of installation is necessary to place the skid in a lower dose area in order to monitor the increase in activity in the filters. This temporary filter skid will not be moved over any spent fuel.

Safety Evaluation Number: S1-EV-99-024
Document Number: TRMCR 99-1-4
Title: Revise Fire Protection Technical Requirements

Description of Activity:
The following changes to the Technical Requirements Manual (TRM) are proposed:

1. Reformatted and renumbered all Fire Protection TRM Sections.
2. Broke out Fire Protection Systems bases section to separately address the bases for sections A through D.
3. Removed fire watch requirements for inoperable fire protection systems, except for the Fire Pump House Halon 1301 system.
4. Added requirements for notification of Site Fire Protection, as well as Unit 2 for inoperable fire dampers, or penetrations in a common Unit 1/Unit 2 barrier fire.
5. Deleted exemption from conducting fire system valve cycling (1-FIRE-29 & 1-FIRE-30), hose station inspections, or fire detector functional instrument tests.
6. Deleted action to place reactor in hot standby within 6 hours and in cold shutdown or refuel condition within 24 hours of loss of fire water distribution system. Eliminated corresponding requirement 3.1.A.4.a to establish a backup fire water distribution system within 24 hours, in favor of provision of an alternate water supply if two or more fire pumps or both water supplies inoperable, as well as cessation of cutting, welding, and grinding.
7. Revised administrative controls section to include the Fire Brigade Advisor and delete reference to the safe shutdown crew.
8. Corrected typographic errors in table 3.1.3 to reflect proper numbers of "Total Available" and "Minimum Instruments Required" for Turbine Building 14'-6" elevation panel 1-FDS-4.
9. Changed applicability statements from "when safety related equipment in the area is operable" to "when equipment in the area is operable".
10. Eliminated the definition of Fire Suppression System

Reason for Activity:
The TRM section was updated to reflect current procedural requirements for format and to reflect the defueled condition.

Safety Evaluation Summary:
This revision to the TRM does not involve any Unreviewed Safety Questions. The revision does not affect the ability to safely shutdown the plant, nor will it result in a reduction in the effectiveness of fire protection of facilities,systems, and equipment which could result in a radiological hazard as required by 10CFR50.48f.

Safety Evaluation Number: S1-EV-99-0026
Document Number: TRMCR 99-1-6
Title: Revise Unit 1 TRM Section 7.6, "Unit 1 SSCs that Interface with Unit 2" to include Unit 1 SSCs that Interface With Unit 3.

Description of Activity:
Unit 1 TRM Change Request 99-1-6 revises Unit 1 TRM Section 7.6 as follows:

- Reformats the TRM
- Deletes the index section of TRM 7.6.
- Renumbers the TRM
- Retitles the TRM to "Unit 1 SSCs that Interface With Units 2 or 3."
- Adds Table 3 and associated text to the TRM Section that identifies Unit 1 SSCs that are required to support Unit 3 Licensing Basis operational requirements.

The first three bulletized items are format changes that are editorial in nature and do not affect the intent of this TRM. The substantive part of this proposed change is the fourth bullet. Table 3 lists the Unit 1 SSC description, system number, and a description of the associated function supporting Unit 3 operation. Table 3 does not interpret the SAR requirements.

Reason for Activity:
Operations Department Manual 1-OM-10.4 lists all Unit 1 SSCs that support Unit 3; however this procedure may be revised without PORC review and approval. Formalizing those SSCs that are required to support the Unit 3 Licensing basis operational requirements in the TRM will ensure that all changes receive the appropriate level of review prior to being changed.

Safety Evaluation Summary:
The proposed TRM change does not increase the probability of human errors in actions required to mitigate an event, the failure of equipment required to mitigate an event, or the failure of equipment that could mitigate an event. The consequences of failing to report to Unit 3 personnel an SSC whose ability to meet the requirements of Unit 3 is suspect are identical to the existing consequences. There is no increased risk to the public.

This safety evaluation concludes that no Unreviewed Safety Question is involved and that the change is safe. The change involved an addition to the Unit 1 Technical Requirements Manual. Its associated Technical Evaluation identify operational aspects of SSCs that are required to support the Unit 2 Licensing Bases. The proposed activities do not increase the probability of occurrence or consequences of Accidents previously evaluated or create the possibility of an Accident of a different type. The proposed activities are safe and do not increase the risk to the public.

Safety Evaluation Number:	S1-EV-99-0030
Document Number:	TSCR 99-1-2 DSARCR 99-1-14
Title:	Revise Bases for Proposed LCO 3.1.1, "Fuel Storage Pool Water Level" and revise DSAR to reflect accident analysis submitted in support of TS Amendment 106.

Description of Activity:
This change incorporates information issued by the NRC in their SER for TS Amendment 106. The proposed change will revise the Bases for LCO 3.1.1 that was submitted to the NRC when the Defueled Technical Specifications were submitted. The change will also revise the DSAR to reflect the submittals and the NRC SER issued with Amendment 106. The bases for LCO 3.1.1 is revised to more accurately reflect the underlying safety analysis.

Reason for Activity:
During the review of Amendment 106 at the NRC, a concern was identified with the methodology utilized during performance of the fuel handling accident calculation. As a result of this concern, additional submittals to the NRC were made that included an assumption that deviated from the proposed Bases that had been previously submitted. Therefore, the Bases that had been proposed no longer accurately reflect the licensing basis of the facility as submitted to the NRC. This change will revise the previously proposed Bases page to reflect the latest submittals to the NRC, and also be utilized to incorporate the appropriate detail into the DSAR.

Safety Evaluation Summary:
This change is being made to incorporate the information previously submitted to the NRC in support of their review of Amendment 106 to the Technical Specifications into the Bases and DSAR for Unit 1. The change does not affect any procedures, practices, hardware, or other activities that are conducted on the unit. It merely corrects a calculation initial condition that was described in the Bases originally proposed with revised LCO 3.1.1, and modifies the DSAR to include the information provided to, and accepted by, the NRC.

Safety Evaluation Number: S1-EV-99-032
Document Number: MP1 NUQAP Ch. 99-02
Title: MP1 Northeast Utilities Quality Assurance Program

Description of Activity:
The following changes to the U1 NUQAP are proposed:
1. Title for VP - Site Services was changed to VP - Nuclear Work Services.
2. Title for VP - Human Services was changed to VP - Human Services - Nuclear
3. VP - Human Services - Nuclear is responsible for employee concerns program and human services. The reporting requirement for these functions was changed from the President and CEO of NNECO to the Senior Vice President and CNO - Millstone.
4. Deleted references to generic titles for Officers and Managers no longer defined in the Technical Specifications.
5. Changed references in SORC composition from Designated Member of Unit 2 and Unit 3 PORC to designated Member of Unit 2/3 PORC.

Reason for Activity:
1. Title changed to reflect official corporate title.
2. Title changed to facilitate organizational changes.
3. Reporting requirement changed to facilitate organizational changes
4. Appendix F contains requirements that were once part of the Technical Specifications. Thus, various Officers and Managers were given Generic Titles. Plant specific titles have been provided for the officers and managers, thus, the references to designated manager/officers are superfluous.
5. The Units 2 and 3 PORCs have been combined into a single committee. This change was made to agree with Change 8 to Revision 21 of the Units 2 and 3 NUQAP.

Safety Evaluation Summary:
The proposed changes to the Millstone Unit 1 NUQAP involve the Millstone organizational structure and responsibilities. They do not involve any physical changes to any structures, systems, or components, nor do they involve any changes in the manner in which a structure, system, or component is operated. The proposed changes do not involve an Unreviewed Safety Question.

Safety Evaluation Number:	E1-EV-99-0002
Document Number:	M1-EV-99-0003 FSARCR 99-MP1-3
Title:	Reclassification of SSCs to support the Defueled Condition and Revise the FSAR to a Defueled SAR

Description of Activity:
The proposed change to the Unit 1 UFSAR is composed of the following activities:

- Removal of items that are no longer applicable due to the defueled condition.
- Clarification of the existing UFSAR language to more accurately reflect the existing design and licensing basis.
- Consolidation and integration of information from the UFSAR into a revised format of 6 chapters and retitled as the "Defueled Safety Analysis Report."
- Addition of a discussion of decommissioning activities and environmental impacts in Chapter 7 of the DSAR based on NUREG/CR-0672 and the Unit 1 Post Shutdown Decommissioning Activities Report (PSDAR).

Reason for the Activity:
On July 21, 1998, Northeast Nuclear Energy Company informed the NRC that Millstone Unit 1 had permanently ceased operations and that the fuel had been permanently removed from the reactor vessel. Pursuant to 10 CFR 50.82(a)(2), the certification in the letter modified the Unit 1 license to permanently withdraw the authority to operate the unit. As a result, Unit 1 is only authorized to possess special nuclear material and is prohibited from placing or retaining fuel in the reactor vessel. As such, structures, systems, and components (SSC) solely used to perform a nuclear safety function or power generation function in a power generating facility are no longer applicable due to the license modifications.

Safety Evaluation Summary:
The proposed activity revised SSC classifications, modifies and deletes some programs, and revises all UFSAR sections in order to reflect the certification that reactor operations at Unit 1 have permanently ceased and that fuel has been permanently removed from the reactor vessel.

The proposed reclassification of SSCs, modification and cancellation of many programs, and the resulting changes in the UFSAR are safe and do not constitute an Unreviewed Safety Question. These activities do not involve an increase in the probability or consequences of a malfunction or of an accident previously evaluated in the SAR. The possibility of a new or different kind of malfunction or accident previously evaluated is not created. These activities do not involve a reduction in the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation Number: S2-EV-98-0198
Document Number: MMOD M2 98045
Title: Unit 2 Appendix R Radio Communications

Description of Activity:
This design change installs an antenna on the outside, and a new base station inside of the Unit 1 cable vault to facilitate use of the 800 MHz radios during postulated Unit 2 Appendix R fires. A new phone and power cable, powered from Unit 1 power, will be routed to one of the two current desktop radio handsets in the control room. Both of these handsets currently serve the existing base station located in the Condensate Polishing Facility. During a loss of power or communications due to the postulated fire, the radio handset will be used and communication will be available.

Reason for Activity:
The design change consists of a set of modifications necessary to restore communications from the control room to the regular 800 MHz radio system during a postulated fire in the Turbine Building.

Safety Evaluation Summary:
The modification to add a communication base station for Appendix R safe shutdown considerations ensures the plant is in compliance with its licensing and design basis, including 10CFR50 Appendix R and the Unit 1 and 2 FSARs. The proposed change bounded by this safety evaluation is the installation of a base station in the Unit 1 cable vault for Unit 2 safe shutdown communications. This design change does not affect the availability or safety function of any devices credited in either Unit 1 or Unit 2 FSAR accident analyses. Also, this change does not compromise the ability to achieve and maintain safe shutdown. This change will provide an alternate communication station to achieve safe shutdown during a Unit 2 fire. Therefore, the change is safe and does not constitute an Unreviewed Safety Question.

Safety Evaluation Number: S3-EV-99-0066
Document Number: MMOD M3-99018
Title: Replacement of Emergency Notification and Response System

Description of Activity:
This modification replaces the existing Emergency Notification and Response System (ENRS) and Site Emergency Response Organization (SERO) Notification System with an automated emergency notification and response system consisting of radiopaging, telephone, audio message and fax delivery computer system. ENRS is a NON-QA system, comprised of redundant telephone servers and client terminals located in Unit 2 and Unit 3 control rooms and simulator, and Emergency Operations Facility (EOF). The Unit 2 client terminal will be shared with Unit 1 Operations.

Reason for Activity:
The Emergency Notification and Response System is not Y2K compliant and replacement parts are no longer supported by the manufacturer. In addition, the Site Emergency Response Organization Notification System performance has been less than adequate.

Safety Evaluation Summary:
The ENRS and SERO Notification System is an automated emergency notification and response system consisting of radiopaging, telephone, audio message and fax delivery computer system to perform SERO, and the State and Local Agency emergency notification. The system satisfies the 15 minute State and Local Agency notification requirement of NUREG-0654. ENRS is a NON-QA system, Y2K compliant and configured with a defense-in-depth design approach and high reliability.

ENRS is independent of the Reactor Protection System and Safeguards Systems of each unit and will not interact with, or impact the operation or design basis functions of safety related equipment or systems. Failure of ENRS components or system would not affect plant systems or components, create plant transients, or initiate accidents, and will not prevent safety systems from performing accident mitigating functions. These modifications do not introduce any accident initiators, alter any fission product barriers or reduce the ability of any safety system to perform its accident mitigation function, and thus will not cause a dose release to the public. These modifications will not degrade the performance of safety systems and will not impact the margin of safety as defined in the Technical Specifications. Therefore, the proposed changes are safe and do not represent an Unreviewed Safety Question.

Safety Evaluation Number:	SG-EV-98-0003
Document Number;	REM/ODCM Rev 11
Title:	Revision 11 to the Radiological Effluent Monitoring and Off-Site Dose Calculation Manual (REM/ODCM): Part II (ODCM) Changes to Pages E-1 and E-2

<u>Description of Activity:</u>
This change will allow Unit 1 to discharge when dilution flow from Unit 1 is not adequate by allowing credit for dilution flow from Unit 2 or Unit 3.

<u>Reason for Activity:</u>
Because Unit 1 is permanently shutdown, the circulating water pumps will not be available during a liquid radwaste discharge. The REM/ODCM currently requires radioactivity concentrations in the quarry to be maintained below 10% of the limits in 10CFR20 using only dilution flows from Unit 1. The legal limit for the radioactivity concentration of water released to the environment is applicable at the quarry cut where the water is discharged into Long Island Sound. Therefore, credit for dilution flows from Unit 2 or Unit 3, which are being released into the quarry at the time of the Unit 1 discharge into the quarry, would ensure that the limit on discharges into Long Island Sound is not exceeded.

<u>Safety Evaluation Summary:</u>
This change will allow Unit 1 to discharge liquid radwaste using the Unit 2 or Unit 3 diluting water when dilution flow from Unit 1 is not adequate, by itself, to maintain radioactivity concentrations in water being discharged to the quarry below 10% of the 10CFR20 limits. Unit 1 discharge could exceed the concentrations at the point of discharge into the quarry; however, radioactivity concentrations in water being discharged to Long Island Sound will be below the limit after mixing with Unit 2 or Unit 3 dilution flow. This is consistent with other requirements in the Unit 1 Technical Specifications. Operation Procedure SP671.5 has been revised to ensure the same degree of surveillance of dilution water flow when crediting other unit flow as when discharging with Unit 1 dilution flow. It requires confirmation of adequate dilution flow from another unit prior to discharge, notification of the other unit at the start of a discharge, and notification of Unit 1 by the other unit if there is a reduction in the credited flow.

Requirements in the REM/ODCM which ensure that discharged radioactivity concentration remains well below the limits are not being changed. These include:

- A limit on discharge flow so that the concentrations in discharge to the Long Island Sound are at 10% of the 10CFR20 limits.
- A radiation monitor trip setpoint at 20% of the 10CFR20 limits.
- Administrative controls to ensure that the required dilution flow is maintained.

For these reasons, this change does not constitute an unreviewed radiological environmental impact and is not a Unreviewed Safety Question.

Safety Evaluation Number: SG-EV-98-0005
Document Number: MMOD M1-98010
M1-EV-98-0021
Title: Millstone Site Radio Communication

Description of Activity:
This modification will expand the users of the 800 MHz radio system to include Security and Emergency Planning and improve radio communication for Fire Protection. This modification will install base stations in the primary and secondary security stations, Tech Support Station, Condensate Polishing Facility, the Fire Protection Station, and the EOF.

Reason for Activity:
Expand the availability and use of the 800 MHz radio system.

Safety Evaluation Summary:
Technical Evaluation M1-EV-98-0021 reviews the potential impact of electromagnetic interference (EMI) from the 800 MHz radio system on plant equipment. The use of the 800 MHz trunked radio system within the Unit 1 power block in accordance with the guidelines provided in Technical Evaluation M1-EV-98-0021 will not adversely affect plant equipment. This safety evaluation concludes that the use of the 800 MHz radio system is safe and does not constitute an Unreviewed Safety Question.

The use of the 800 MHz trunked radio system within the Unit 1 Power Block is safe because (a) most plant equipment is not adversely impacted by EMI and (b) radio use is prohibited in the areas where plant equipment susceptible to EMI from radios is located. With these restrictions in place, the use of the 800 MHz portables will have no impact on plant equipment operation.

The proposed change does not conflict with any safety requirements required by the Emergency Plan or the Security Plan. To prevent false indications, inadvertent equipment actuation or equipment malfunctions, transmission from the portable radios is administratively controlled through plant access training, on the job training, and postings to prevent their use in areas where known sensitive equipment is located. The proposed change will not increase the probability of a malfunction or the occurrence of an accident. It will not increase the consequences of a malfunction or an accident, or create the possibility of a different type of malfunction or accident. The proposed activity will not modify any plant control system, such that plant transients will be more likely, or alter the performance of any safety system required to prevent or mitigate accidents or anticipated operational occurrences, which are analyzed in the safety analysis. The proposed change will not impact any assumptions contained in the accident analyses or reduce the margin of safety as defined in the bases for any technical specification. The proposed change is safe and there are no Unreviewed Safety Questions.

Safety Evaluation Number: SG-EV-99-0005
Document Number; TE MG-EV-99-0003
Title: Site Modification for Red Barn Area Access

Description of the Activity:
The activity being evaluated is the addition of vehicle barriers along the south side of the Site protected area fence line, extending from the Refueling Outage Building westward to Long Island Sound, and the subsequent removal of the existing barriers blocking access to the Red Barn area.

Reason for the Activity:
The reason for the activity is to re-open the Red Barn area to vehicular traffic while maintaining the same level of plant security without adversely impacting the design and licensing basis of the site or any individual unit.

Safety Evaluation Summary:
The only impact to the site, and Unit 1 in particular, from installation of the jersey barriers along the southern boundary of the site is a slight increase in flood levels should a Probable Maximum Precipitation event occur. This floods the southern and eastern sides of Unit 1 by up to 2.4 additional inches of rainwater. The credited safety-related system is the Emergency Diesel Generator, which is not impacted by the increased flood levels. Thus, there is no impact on the site or an individual unit as a result of this proposed new vehicle barrier configuration. This increase in flood level does not increase the probability of occurrence or consequences of an accident or malfunction of any equipment important to safety. As such, the proposed new vehicle barrier configuration is safe and does not impact the design or licensing basis of the site or any individual unit. The proposed change is safe and there are no unreviewed safety questions.

Attachment 3 to B18128
Millstone Unit 1
Commitment Changes

The following three commitments which were maintained as potentially relevant to the defueled condition have been evaluated and changed using the guidance of NEI 99-04 as implemented by Millstone 1 procedure U1 RAC 06 "Regulatory Commitment Management."

1. **RCR-01958**

   From NU Letter B14703 "Millstone Nuclear Power Station, Unit No. 2, Reply to a Notice of Violation, NRC Inspection Report No. 50-336/93-18," dated 1/5/94

   Original Commitment

   Two additional engineers/senior engineers will be added to the Independent Safety Engineering Group (ISEG) organization, such that the ISEG function will be extended beyond the technical specification requirements for Millstone Unit No. 3 and expand the voluntary ISEG functions at Millstone Unit Nos. 1 and 2 as well. This action was scheduled for completion by March 1, 1994.

   Change

   The voluntary commitment for an ISEG organization supporting Unit 1 was cancelled.

   Discussion

   The ISEG function as envisioned in NUREGs 0660 and 0737 and implemented in the Unit 2/3 Quality Assurance Plan was to review "unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of Unit design and operating experience information . . . which may indicate areas for improving unit safety"

   For Unit 1, the potential for an adverse impact on the health and safety of the public is very limited and is relatively fixed. The unit operating characteristics, industry operating experience, and other sources are not generally applicable to permanently shut down units. When necessary this information will be assessed and acted upon by the existing organization.

2. **RCR-39803**

   From NRC Letter "NRC Combined Inspection 50-245/97-01; 50-336/97-01; 50-423/97-01 and Notice of Violation" dated 4/11/97

   Original Commitment text:

   Procedural controls were put in place to ensure that no fuel assemblies are transferred within the spent fuel pool until all fuel is fully seated.

   Change

   This commitment has been revised to state:

   No unaffected, irradiated fuel assemblies will be moved within the spent fuel pool until the raised fuel assemblies have been fully seated. In the event that an assembly cannot be fully seated, no unaffected, irradiated fuel assemblies will be moved until it is determined that either the assembly configuration has been previously evaluated, or an evaluation of the configuration is performed.

Discussion
As originally worded, the commitment did not allow for removal of the new (unirradiated) fuel, nor the movement of any of the raised fuel assemblies in order to get them seated properly. This change allows the fuel movement to occur but ensures that potential consequences are evaluated prior to fuel movement. This change was evaluated under safety evaluation S1-EV-99-0020.

3. **RCR-42714**
From LER 97-037-00, dated 10/30/1997

Original commitment
NNECO commits to provide an updated submittal to the NRC in accordance with the guidelines of I&E Bulletin 80-10 prior to restart from the current refueling outage.

Change
This commitment has been cancelled.

Discussion
I&E Bulletin 80-10 required plants in operation at the time to (1) identify non-radioactive systems that could become contaminated, and (2) establish a program to monitor those systems. The bulletin goes on to to provide guidance as to the NRC expectations if one of these systems become contaminated. The Bulletin required a 60 day response which was submitted on July 3, 1980.

LER 97-037-00 documented a breakdown in the program and committed to update the 80-10 response prior to restart. Since the unit is permanently shutdown, the due date will never occur. A station procedure was developed that documents the program has been established for the station to both (1) meet the intent of I&E bulletin 80-10 and (2) to prevent future program breakdowns. The procedure lists the MP1, 2 & 3 systems that are contaminated or have the potential to be contaminated and provides guidance for the proper response should a system become cross-contaminated. On a going forward basis, this procedure satisfies the intent of I&E Bulletin 80-10 and the commitment to update the earlier response would serve no useful purpose