Maine Yankee

321 OLD FERRY RD. • WISCASSET, ME 04578-4922

July 18, 2001
MN-01-029 RA-01-115

UNITED STATES NUCLEAR REGULATORY COMMISSION
Attention: Document Control Desk
Washington, DC 20555

Reference: (a) License No. DPR-36 (Docket No. 50-309)

Subject: Submittal of the Maine Yankee Defueled Safety Analysis Report Rev. 18

Gentlemen:

Pursuant to the requirements of 10 CFR §50.71 and 10 CFR §50.4, please find enclosed Revision 18 of the Maine Yankee Defueled Safety Analysis Report (DSAR) and ten copies. Separate copies are being supplied to the USNRC Region I office, the Maine Yankee Project Manager and the on-site NRC office.

Sincerely,

Thomas E Shipper

for Thomas L. Williamson, Director
Nuclear Safety and Regulatory Affairs

Enclosure

c: Mr. H. J. Miller, USNRC Region 1
Mr. M. K. Webb, USNRC OWFN
USNRC Maine Yankee Site Office
Mr. P. J. Dostie, State of Maine
w/o enclosure:
Mr. R. A. Gramm, USNRC OWFN
Mr. M. Roberts, USNRC Region 1
Ms. P. Craighead, Esq., State of Maine

A053
1/11

STATE OF MAINE

Then personally appeared before me, Thomas E. Shippee, who being duly sworn did state that he is acting for Director of Nuclear Safety and Regulatory Affairs at Maine Yankee Atomic Power Company, that he is duly authorized to execute and file the foregoing request in the name and on behalf of Maine Yankee Atomic Power Company, and that the statements therein are true to the best of his knowledge and belief.

Donna L. Pelletier

Notary Public

OFFICIAL SEAL
DONNA L. PELLETIER
Notary Public
STATE OF MAINE
My Commission Expires Dec. 11, 2006

MAINE YANKEE DEFUELED SAFETY ANALYSIS REPORT (DSAR), REV. 18 INSERTION INSTRUCTIONS

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transfer tube is closed by a manual isolation gate valve located in the spent fuel pool and by a blind flange located inside the containment.

Following the cessation of power operation and the removal of fuel from the vessel, the spent fuel inventory consists of a total of 1432 complete spent fuel assemblies, 2 partially consolidated assemblies, and 2 partially full failed fuel rod containers residing in the spent fuel pool. Additionally, the spent fuel pool contains a number of filters from prior pool vacuuming efforts, reactor startup neutron sources, Control Element Assemblies (CEAs), CEA fingers, Incore Instrumentation (ICI) thimbles, and a segment of ICI detector.

The spent fuel is stored in a single tier rectilinear array of free standing modules. Each fuel assembly, failed fuel rod container, or trash basket is contained within an individual cell. Cells are grouped together to form the free standing modules or racks. These high density spent fuel storage racks are designed for the passive reactivity control of the spent fuel in storage through the use of a fixed neutron absorber material and the spacing between assemblies. Design accommodations for cooling the spent fuel while in storage in the racks is provided through the use of flow holes in the bottom of each fuel storage cell. The spent fuel storage racks are designed to maintain the fuel in a coolable, subcritical geometry during accident conditions in the pool.

The spent fuel pool is filled with a borated water solution to provide a medium for cooling the spent fuel, shielding for workers and the public from normal and accidental radiation exposure, and as a means of controlling the spent fuel reactivity.

The spent fuel pool cooling system removes the spent fuel decay heat stored in the spent fuel pool by circulating the borated pool water through a heat exchanger. Each spent fuel pool cooling pump takes suction from the fuel pool, circulates the water through a heat exchanger, and returns it to the fuel pool below normal water level. Decay Heat Removal (DHR) water flows through the shell side of the heat exchanger and cools the tube side borated pool water. The DHR water is cooled by the DHR air-cooled heat exchangers located north of the BWST dikes. In the event that the DHR system is not available, fire protection system water may be substituted by manually connecting a hose to a flange connection on the shell side of the fuel pool heat exchanger.

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THE SITE AND ENVIRONS

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THE SITE AND ENVIRONS

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2.2 **Meteorology**

2.2.1 General

The Bailey Point site is located in the mid-coastal region of Maine. This coastal region is characterized by many inlets, bays, channels, harbors, rocky, islands, and promontories. The area adjacent to the site has many small forested hills.

The general climatic regime is maritime with its cool air moving in from the North Atlantic. Of special importance, from an engineering standpoint, are the extremes in annual snowfall for the coastal region; the occasional heavy rains, the coastal storm or "Nor'easter" with its resultant strong winds and heavy rain or snow; and sometimes glaze or "ice storms". These and other pertinent meteorological data which have been compiled by the TRC Corporation of Hartford, Connecticut, are presented in the following sections.

2.2.2 On-Site Meteorological Field Programs

An initial data collection program was undertaken at the site of the Maine Yankee Atomic Power Station to provide information on meteorological conditions for dispersion analyses for the original FSAR. Data for one year, from July 1, 1967 to June 30, 1968, were evaluated and formed the bases for those analyses. Appendix A contains a discussion of the July 1967 - June 1968 data collected from the initial monitoring program.

An upgraded on-site monitoring program which met the intent of Revision 0 to Regulatory Guide 1.23 was installed in late 1976. A description of this upgraded system is presented in Appendix A, along with wind and stability data summaries for one year of operation, from January 1, 1979 to December 31, 1979. This one-year period-of-record also forms the basis for the off-site dose accident analyses presented in Section 5 of this report.

2.2.3 Coastal Fog

Heavy fog is frequent and sometimes persistent along the coast, and may occur on one day in six during certain portions of the year. Data for the 11-year period (1951-1962) from the Brunswick Naval Air Station located 13 miles from the site indicate that 4.1% of the time (3,855 out of 95,073 observations), fog conditions exist. A fog condition is said to exist when the visibility is 0 to ½ mile.

2.2.7 Environmental Monitoring Program

2.2.7.1 Program Description

In the operation and decommissioning of Maine Yankee, small quantities of radioactive gases and liquids are released to the environment in a controlled manner in accordance with the applicable federal and state regulations. An environmental monitoring program was established to demonstrate the adequacy of safeguards inherent in the plant design, and the effectiveness of in-plant monitoring of controlled releases of radioactive materials.

The program consists of three phases, pre-operational, operational, and post-operational each having specific objectives. The pre-operational phase was started approximately two years prior to operation of Maine Yankee to establish background radiation levels and concentrations at selected locations, to determine variability between sample locations, and to observe any cyclical or seasonal trends in the environmental sample media. The operational phase of the program had the following objectives:

1. To assure those organizations responsible for public health and safety that concentrations of radionuclides in the environment resulting from plant operation met the applicable federal, state, and license regulations.

2. To make possible the prompt recognition of any significant increase in environment level of radioactivity related to plant operation.

3. To differentiate plant releases from other abnormal trends in environmental radiation due to fallout, other nuclear facilities and seasonal changes in natural background.

4. To obtain information on the critical radionuclides and pathways leading to the quantitative evaluation of the dose to man resulting from the operation of the plant if significant trends in positive environmental concentration are identified.

The post-operation phase is a continuation of the operational phase objectives with the intent to monitor the impact of the plant decommissioning process and the continued operation of the spent fuel pool.

A description of the radiological environmental monitoring program, in tabular form, is shown on Tables 2.2.8 and 2.2.9.

Environmental Radiological Sampling Locations (within 1 km)

Maine Yankee

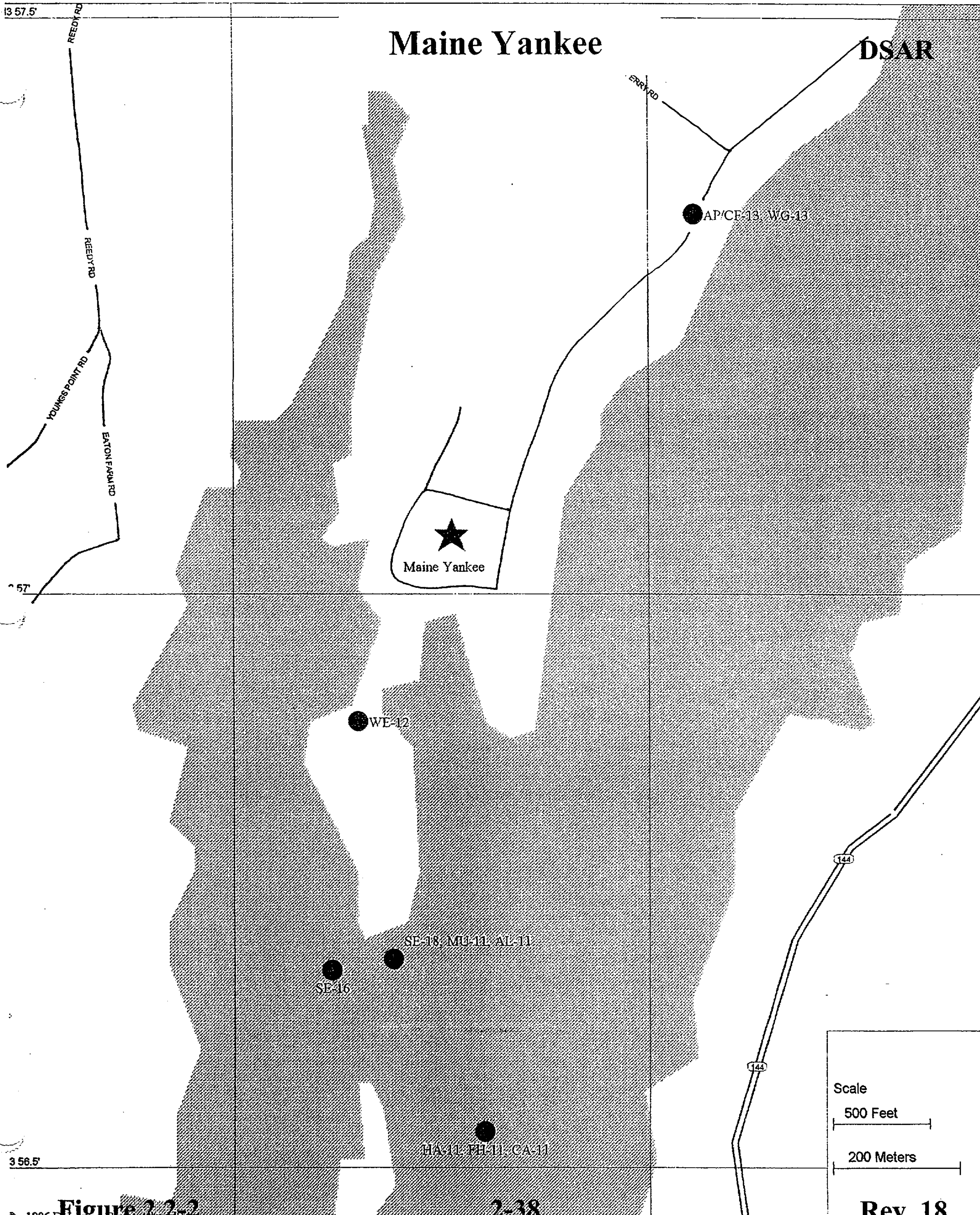

Figure 2.2-2

MAINE YANKEE ATOMIC POWER CO.	DELETED	FIGURES 2.2-3

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Environmental Radiological Sampling Locations (outside 1 km)

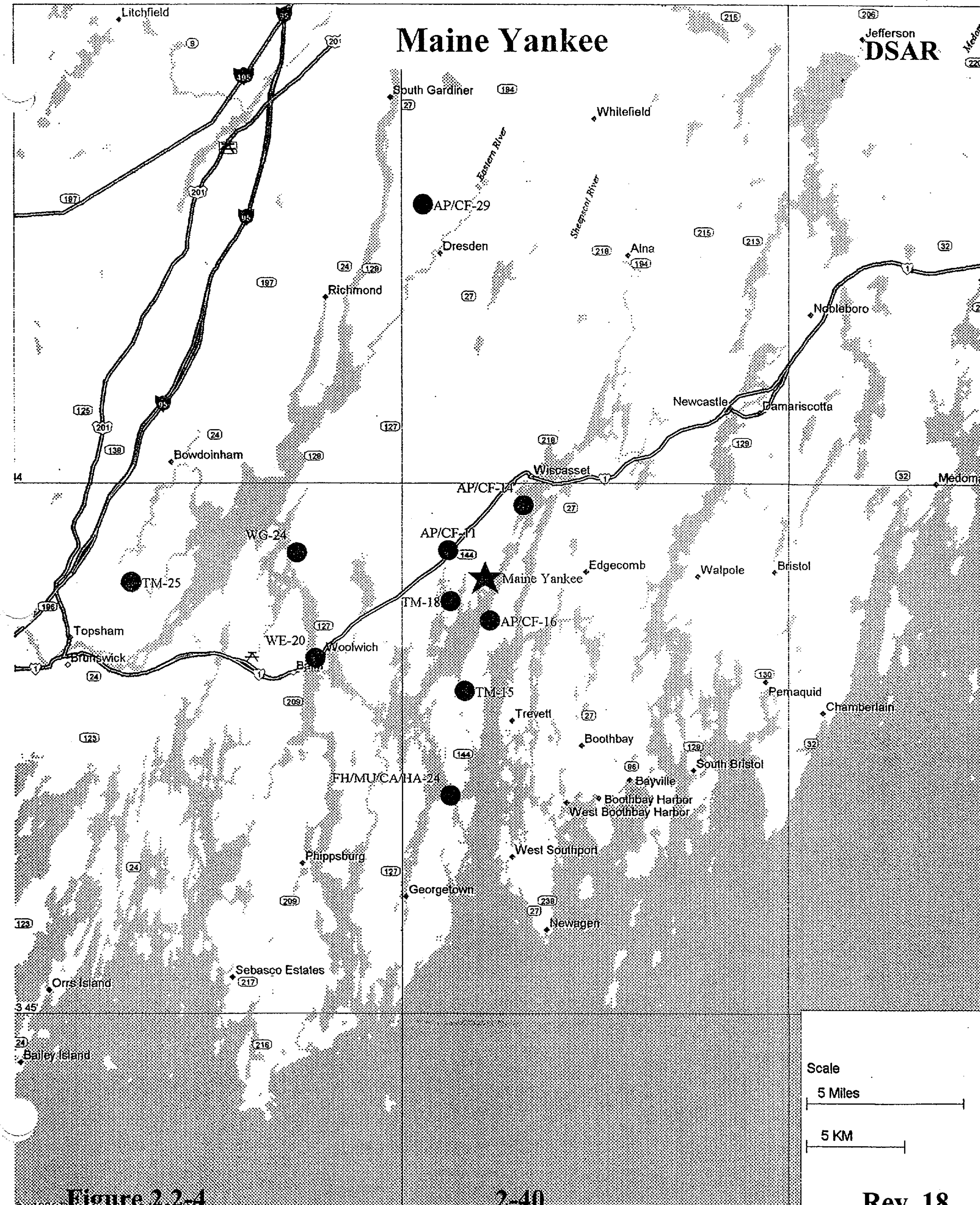

Figure 2.2-4

Direct Radiation Monitoring Locations (within 1 km)

Maine Yankee

DSAR

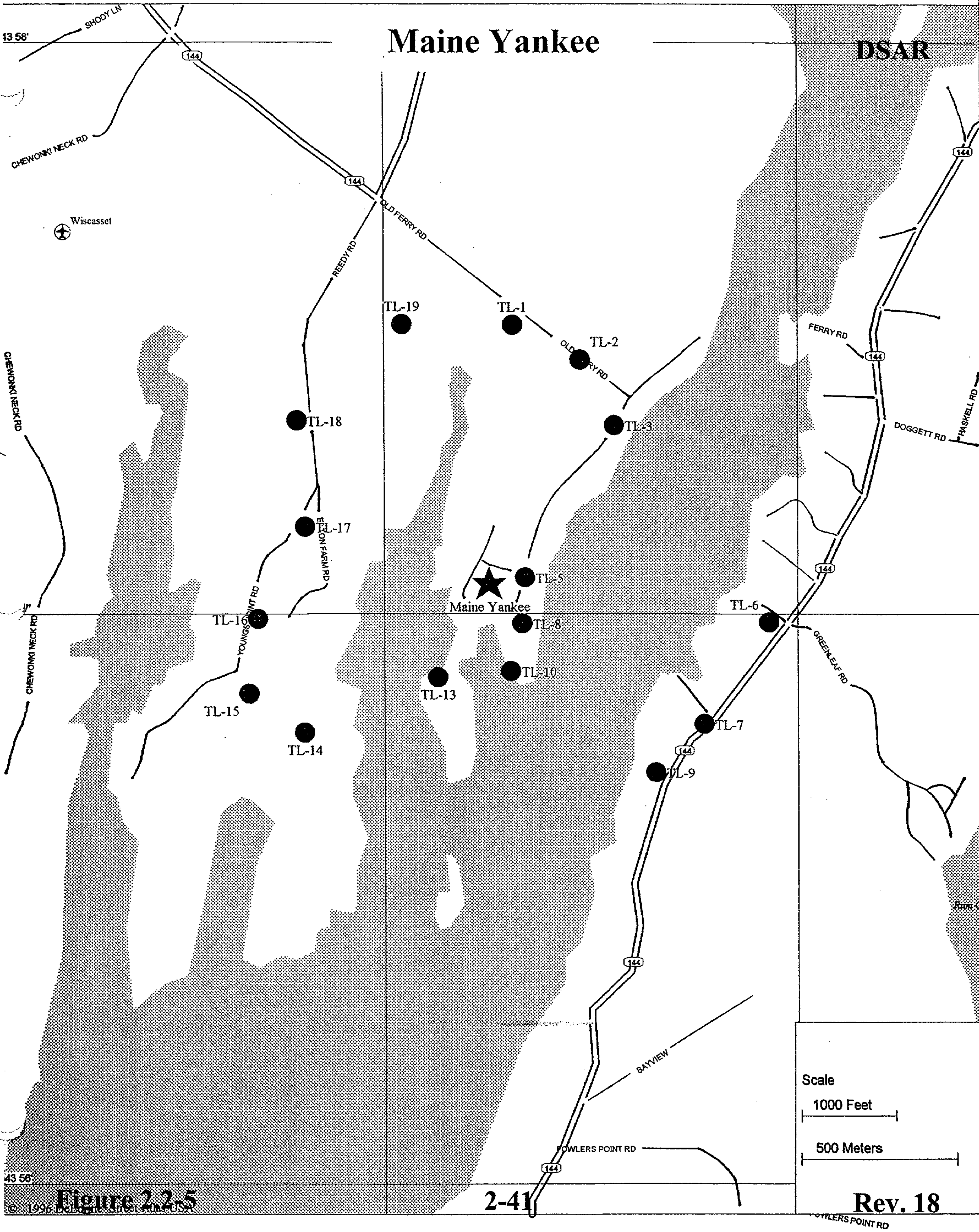

Figure 2.2-5

MAINE YANKEE ATOMIC POWER CO.	DELETED	FIGURES 2.2-6

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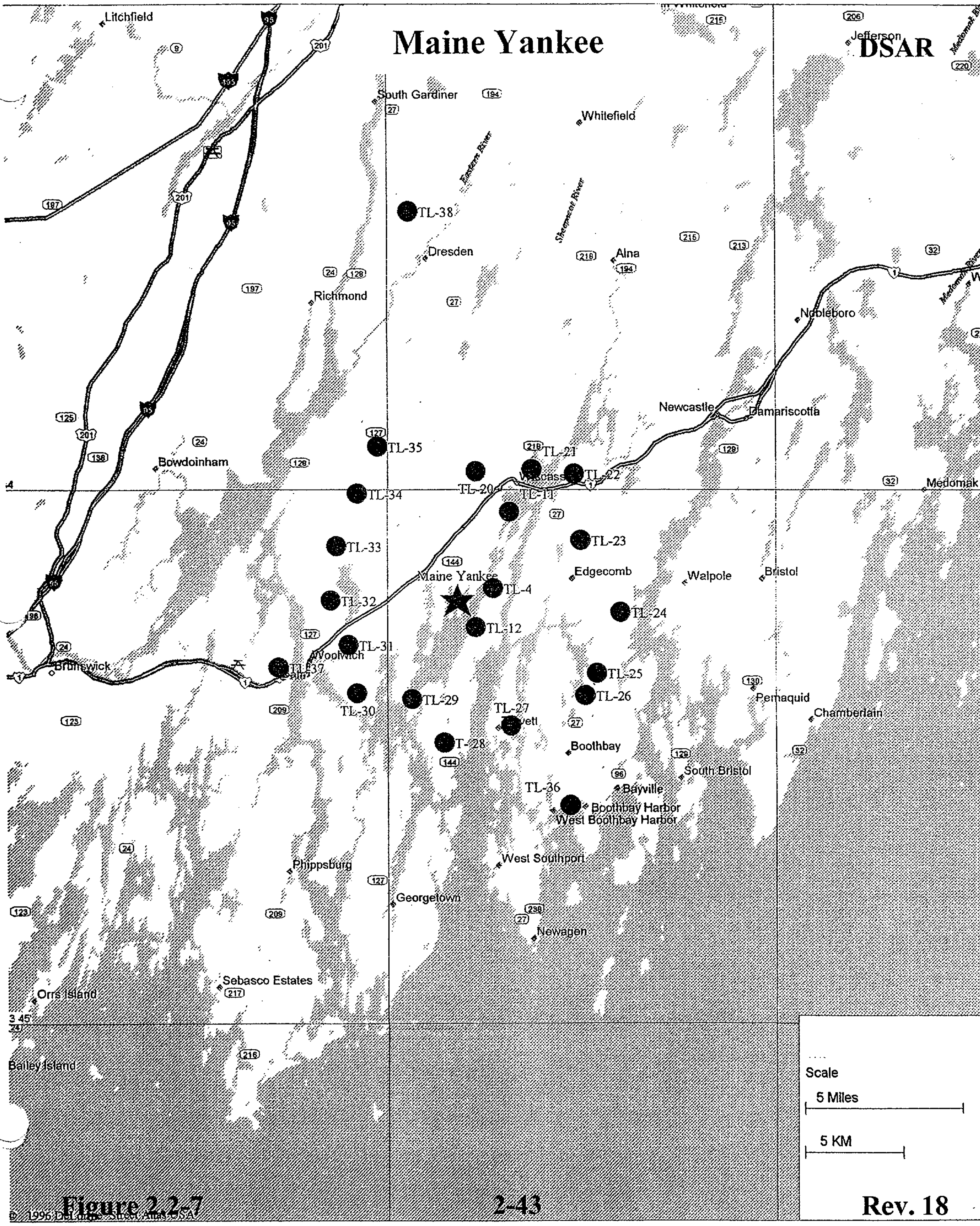
Direct Radiation Monitoring Locations (outside 1 km)



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SECTION 3.0
FACILITY DESIGN AND OPERATION

Section 3.0 discusses the design and operation of the structures, systems, and components required to safely store fuel. It also discusses supporting systems such as ventilation and auxiliary systems used to safely store fuel or support decontamination and decommissioning activities.

3.1 Design Criteria

3.1.1 Conformance with 10CFR 50 Appendix A General Design Criteria

In July of 1967, the Atomic Energy Commission issued the proposed general design criteria for nuclear power plants. These 70 criteria were issued for comment by the industry but had not yet been adopted as a regulatory requirement. Nevertheless, as the following discussion shows, the Maine Yankee plant has been designed and constructed in accordance with the intent of these criteria.

In the following paragraphs, each criterion is stated and its conformance indicated. On September 18, 1992, the USNRC confirmed that plants with construction permits issued prior to May 21, 1971 were evaluated on a plant-specific basis, and backfitting the current General Design Criteria of Appendix A to 10CFR 50 would provide little or no safety benefit. (Reference 1, SECY-92-223)

On August 7, 1997, Maine Yankee certified in accordance with 10CFR 50.82 that the company had permanently ceased power operation and that all fuel was removed from the reactor vessel (Reference 2, MN-97-89). With the docketing of these certifications, the Maine Yankee license no longer allowed operation of the reactor or placing of fuel in the reactor vessel. The AEC 1967 Design Criteria listed below are relevant to the permanently defueled plant condition.

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Group I - Overall Plant Requirements

Criterion 1 - Quality Standards

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

(This Criterion is directly analogous to Criterion 1 of 10CFR 50, Appendix A, 1971, except that the 1971 version addresses records retention. Maine Yankee record retention requirements are addressed in the Quality Assurance Program. Design, fabrication, and construction records are addressed by Criterion 5 of this section.)

Response:

Those systems and components which are essential to the prevention of accidents which could affect the public health and safety, or to mitigation of their consequences, have been designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Generally recognized codes and standards on design, materials, fabrication, and inspection have been used. These were supplemented to reflect current practices. The descriptions of the systems and components to which this criterion applies include the codes and other standards met by these systems. The quality assurance program is submitted to the regulator for review and approval of any proposed revisions which would result in a reduction of previous commitments.

References: Sections 3 and 4

Criterion 2 - Performance Standards

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

(This Criterion is directly analogous to Criterion 2 of 10CFR 50, Appendix A, 1971, except that the 1971 version addresses consideration of appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and the importance of the safety functions to be performed.)

Response:

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

On April 17, 1979, an earthquake of approximate magnitude 4 occurred about 10 kilometers west of the plant. On January 9, 1982, an earthquake of approximate magnitude 5.75 occurred in Central New Brunswick. Subsequently, the Licensee joined with the Regulator in a program to assess the seismic ruggedness of the Maine Yankee plant. This program is referred to as the Seismic Design Margins Program (SDMP). This program is fully described in NUREG/CR-4826, dated March 1987. On March 26, 1987, the Regulator issued a Safety Evaluation Report which concluded that all issues associated with the seismic design were considered resolved (Ref. 3).

References: Sections 2 and 3.

Criterion 3 - Fire Protection

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions, and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

(This Criterion is directly analogous to Criterion 3 of 10CFR 50, Appendix A, 1971, except that the 1971 version addresses fire protection and fire fighting considerations.)

Response:

As described in Section 3, the materials and layout used in the station design have been chosen to minimize the possibility and to mitigate the effects of fire. Sufficient fire protection systems and equipment have been provided to minimize the adverse effects of fire on structures, systems, and components important to safety taking into account the decommissioning plant condition and activities.

Criterion 5 - Record Requirement

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

(This Criterion, in combination with Criterion 1 of this section, is analogous with Criterion 1 of 10CFR 50, Appendix A, 1971.)

Response:

Records of design, fabrication, and construction of components are being maintained for the duration of the license. Design calculations are in the possession of Stone & Webster and Yankee Atomic. All other required design, fabrication, and construction information is in the possession of Maine Yankee.

References: Section 6 and the Quality Assurance Program

GROUP III - NUCLEAR AND RADIATION CONTROLS

Criterion 11 - Control Room

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Response:

This criterion is met only to the extent that radiation exposures of personnel in excess of 10CFR 20 limits cannot occur based on the available source term from any credible accident. Sufficient time is available to allow operators to restore cooling or makeup to the spent fuel pool and maintain exposures well below the limits of 10CFR 20. Controls for spent fuel pool cooling, makeup and purification are located near the equipment and are not in the control room.

References: Sections 3, 5, and 7

Criterion 12 - Instrumentation and Control Systems

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Response:

Instrumentation is provided as required to monitor and maintain significant variables. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation. In the permanently defueled condition, the principal variables to be monitored include fuel pool level, and temperature. Boron concentration is determined via sampling.

References: Sections 2, 3, and 4

Criterion 17 - Monitoring Radioactive Releases

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

(This Criterion is analogous to Criterion 64 of 10CFR50, App. A, 1971, except that the 1971 version addresses spaces containing components for recirculation of loss-of-coolant accident fluids.)

Response:

The means for monitoring radiation levels in the spent fuel pool and concentrations on-site is provided. The sensitivity and range of this equipment is adequate for operating and anticipated transients in the permanently defueled condition.

Effluent discharge paths where a potential release of radioactive material exists are monitored. Monitoring equipment has sufficient sensitivity and range for operating and anticipated transients in the permanently defueled condition.

Instrumentation is provided for monitoring radiation levels in the spent fuel pool area.

An environmental surveillance program has been established. The facility's environmental surveillance program provides effective monitoring of radioactive material released from the plant.

References: Section 4.

Criterion 18 - Monitoring Fuel and Waste Storage

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

Response:

The spent fuel is monitored by an area detector with audible and visual alarms activated at the detector location and in the control room. For waste handling operations or decontamination operations, portable detectors with audible alarms may be used to monitor radiation exposure.

The spent fuel pool is equipped with high and low liquid level alarms, and a high temperature alarm which will indicate loss of continuity in decay heat removal capacity of the fuel pool cooling system.

References: Section 3 and 4

GROUP VIII - FUEL AND WASTE STORAGE SYSTEM

Criterion 66 - Prevention of Fuel Storage Criticality

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Response:

Criticality is prevented by geometrically safe configurations.

Irradiated spent fuel is stored under water in a reinforced concrete pool, lined with stainless steel. Fuel assemblies are spaced and the racks are so fabricated that criticality is precluded. Although the water in the pool is generally borated, neither soluble boron nor control rods are required to keep even unirradiated fuel assemblies subcritical. The boron concentration required as a result of the analyzed "misplaced assembly" incident is discussed in section 3.3 and 5.2.

Reference: Section 3.3.

Criterion 67 - Fuel and Waste Storage Decay Heat

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Response:

The spent fuel pool is designed to maintain the water level safely above the spent fuel assemblies at all times. The fuel pool outlet pipe, which serves as the suction pipe to the fuel pool cooling pumps, is siphon protected to prevent draindown of the pool. The return piping is located such that siphoning by the cooling system is limited to no less than 10 feet above the active fuel. Additionally, a branch syphon break has been installed to increase this limit to no less than 19' above the active fuel.

Pool water level may be restored through diverse (offsite and onsite) supplies of fresh or demineralized water. Emergency makeup water and cooling is also available from the fire pond.

Reference: Section 3.

Criterion 68 - Fuel and Waste Storage Radiation Shielding

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10CFR 20.

(Also refer to Criterion 67 of this DSAR section.)

Response:

The radiation shielding (water) of the spent fuel pool is designed to provide adequate personnel protection. The waste storage and processing facilities are shielded as required to protect personnel from exposures in excess of regulatory requirements. In addition to shielding design, implementation of the Radiation Protection Plan requirements assures that doses to personnel performing work in these areas are maintained ALARA.

Reference: Section 3 and 4.

Criterion 69 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage

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

Response:

Fuel and waste storage areas are designed to preclude the inadvertent release of undue amounts of radioactivity. Considering the significantly diminished source terms in the permanently defueled condition, there are no credible accidents resulting in doses to the public approaching the 10CFR 100 limits. All spent fuel and waste storage systems are conservatively designed with ample margin to prevent the possibility of gross mechanical failure which could release significant amounts of radioactivity. Backup systems such as floor and trench drains are provided to collect potential leakages to preclude the release of radioactive materials to the environment.

Personnel are rigorously trained and administrative procedures are strictly followed to reduce the potential for human error. In addition, radiological limits on waste storage systems are established to assure that credible accident conditions will not result in doses to the public which approach 1 rem (whole body) exposure over a 2 hour-limited duration accident.

The consequences of a fuel handling incident are presented in Sections 3 and 5. In this analysis, it is demonstrated that undue amounts of radioactivity are not released to the public.

References: Sections 3, 4, 5 and 7.

GROUP IX - PLANT EFFLUENTS

Criterion 70 - Control of Release of Radioactivity to the Environment

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10CFR 20 requirements for normal operation and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Response:

The plant radioactive waste control systems (which include the liquid, gaseous and solid radwaste systems) are designed to limit the off-site radiation exposure during normal operation to levels below limits set forth in 10CFR 20.

References: Sections 4.

3.1.2 Classification of Structures, Systems, and Components

Structure Classification

The plant structures and process systems are classified according to their function and the degree of integrity required to protect the public from uncontrolled releases of radioactive byproducts. Structural design criteria include two classes of buildings:

1. Class I Structures

Class I structures were designed in accordance with the "Building Code Requirements for Reinforced Concrete," ACI-318-63, including increases allowed for stresses produced by earthquake loads in combination with other appropriate loads. Where steel is utilized, it is designed in accordance with AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings." A Class I structure will maintain its integrity during the hypothetical earthquake where the combination of the normal operating loads and the seismic stresses do not exceed 90% of the yield strength. These structures are primarily constructed with massive reinforced concrete and the earthquake loads are not a major factor in the design.

Class I structures and equipment are designed to remain functional during an operational basis earthquake (ground acceleration 0.05g) and maintain fuel pool integrity during the more severe design basis of the hypothetical earthquake (ground acceleration 0.10g). In addition, some of the Class I structures are designed so that damage will not result from tornado winds or missiles.

The structures, systems and components which have been designed to Class I seismic requirements are listed below:

PAB (only to the extent it may affect the Fuel Building or other SFPI SSC's)

Containment Building (reinforced concrete substructure and superstructure)

Fuel Building (reinforced concrete structure and steel superstructure)
- Fuel assemblies

Fuel Pool Liner
Valve FP-21
Blind flange on Containment side of Fuel Transfer Tube
Fuel Transfer Tube
Spent fuel storage racks
Fuel building - yard crane steel support structure]
Fuel building - yard crane (CR-3)
Fuel handling platform and hoist (CR-9)
Flow limiters on the liner leakage detection system
Spent Fuel Pool Cooling Loop Suction Piping (from the pool wall up to and including the siphon protection)

2. Class II Structures

Class II structures are usually designed to the requirements of the Uniform Building Code. Class II structures were not designed for OBE and DBE. They were designed for dead load plus live load plus wind in accordance with AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings" and "Building Code Requirements for Reinforced Concrete", ACI-318-63, Part IV-A, utilizing working stress design methods.

Where Class I structures are connected directly to the Class II structures, the interaction between the structures is taken into account in the design of both. In addition, shake space between all adjacent Class I structures is provided for in the design. This rattle space is a minimum of 3 inches which is conservative with respect to the actual seismic requirements to prevent impact between buildings in the event of a disturbance.

<u>Dismantlement of Seismically Designed Structures, Systems and Components</u>

For SSCs formerly designed to Seismic Class I requirements but not credited for performing a Seismic Class I function in the defueled condition, the following criteria apply prior to performing dismantlement operations:

a. Declassification of components shall be performed in accordance with appropriate engineering and design procedures and processes.
b. When declassifying an SSC, a 10CFR 50.59 evaluation shall be performed if:
    1) the safety classification is described in the FSAR, or

2) it's failure in a seismic event could affect a Seismic Class I component described in the FSAR in such a manner as to cause an unanalyzed incident or an accident with offsite doses exceeding 230 mrem (whole body) or 260 mrem (organ dose).

c. When declassifying an SSC, a 10CFR 50.54 evaluation shall be performed if the classification is described in the OQAP.

d. SSCs shall be designed to seismic Class I requirements if, during a seismic event, its failure has the potential to drain the fuel pool water level lower than 10 feet above the active fuel.

3.1.2.1 Structures, Systems and Components Important to the Defueled Condition (ITDC)

General

On August 7, 1997, Maine Yankee certified per 10 CFR 50.82 that the company had permanently ceased power operation and that all irradiated fuel had been permanently removed from the reactor vessel (Reference 2). This is a permanent, non-revocable certification that changed Maine Yankee's licensing basis by no longer allowing fuel in the reactor vessel and no longer allowing power operation.

The license basis for the majority of Structures, Systems and Components (SSCs) associated with nuclear safety has been changed. Those SSCs which only performed a reactor safety function (i.e., SSCs which do not support a spent fuel or radiation protection safety function) need no longer be maintained under nuclear grade controls.

SSC classification involves a determination that an SSC is, or is not, safety-related[1]. SSCs classified as safety-related are treated differently by regulation than other SSCs.[2]

For a plant undergoing decommissioning, the only SSCs which meet the definition of safety-related[3] are the fuel transfer tube, and spent fuel cooling loop suction piping (from the pool wall

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1. Safety related SSCs are those relied upon to remain functional during and following design basis events to ensure: a) the integrity of the reactor coolant pressure boundary; b) the capability to shut down the reactor and maintain it in a safe shutdown condition; and c) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR 100.

2. 10 CFR 50 Appendix B notes that "The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of..." SSCs.

3. The first two parts of the safety-related definition (reactor coolant pressure boundary, and capability to achieve and maintain safe shutdown) do not apply to a decommissioning plant, given the license restrictions of 10 CFR 50.82. The third part of the safety-related definition (accident consequences comparable to 10 CFR 100 guidelines) also does not apply. At Maine Yankee, the consequences associated with the design/license basis events applicable to decommissioning are nearly three orders of magnitude lower than Part 100 guidelines and lower than the EPA protective action guide limit.

to and including the siphon protection), blind flange on Containment side of fuel transfer tube, valve FP-21. This results in two areas of interest:

1) Maine Yankee's "nuclear grade" processes are based largely upon quality assurance (10 CFR 50 Appendix B) requirements. Reclassifying all SSCs as non-safety related could lead to the elimination of most management controls in situations where maintaining rigorous management controls is intended,

2) Maine Yankee recognizes that certain functions remain important to safety in the defueled condition.

It is necessary to reclassify SSCs in order to proceed with decommissioning. Strictly following regulatory requirements in reclassification results in elimination of most of the current management controls, which is contrary to management's intent. Thus, in order to provide an enhanced engineering controls above that mandated by regulatory requirements, an artificial classification system termed "Important to the Defueled Condition" (ITDC) is introduced.

The following concerns are addressed within this classification:

- SSCs which support a fuel safety or radiation protection safety function, and
- Identification of enhanced management and engineering controls are maintained on SSCs classified as ITDC.

It is noted that SSCs which are not defined as within the ITDC classification, or otherwise designated as safety class, are eliminated from the license basis. It is not the intention of implementing the ITDC classification to reclassify components previously defined as NNS as ITDC. Additionally, there may be other SSCs to which a level of enhanced quality or engineering oversight has been applied, but do not meet the intent of the ITDC classification.

The following criteria are used to determine which SSCs are designated as ITDC:

Criterion 1. The SSC is essential to the normal operation of the storage, control, or maintenance of the spent nuclear fuel or the monitoring of radioactive effluent.

Criterion 2. The SSC is essential in preventing postulated accidents or incidents involving the storage, control, or maintenance of the spent nuclear fuel or the monitoring of radioactive effluent.

Criterion 3. The SSC historical classification is the direct result of an outstanding commitment to the USNRC which remains essential to storage, control, or maintenance of the spent nuclear fuel; or the monitoring of radioactive effluent.

Criterion 4. The SSC satisfies a requirement based in regulations which remain essential to storage, control, or maintenance of the spent nuclear fuel; or the monitoring of radioactive effluent as defined in the Maine Yankee licensing basis. This includes any SSC which is independently required by the Limiting Conditions of Operation (Section 3) of the Technical Specifications.[4]

A positive response to any criterion indicates that an SSC is ITDC.

Authorizations, Restrictions and Limitations on use of the SSC reclassification criteria.

The SSC reclassification criteria will be used as a basis to change various Maine Yankee processes, provided that the change involves an SSC that is non-ITDC and, provided that plant procedures contain an acceptable method for approving the change. The following kinds of "software" changes associated with non-ITDC SSCs are allowed:

- SSC classifications
- drawings,
- calculations,
- procedures
- nonconforming items and corrective actions
- external industry operating experience reports
- commitments
- open work orders (in process at the time the decision was made to decommission the plant)
- the application of 10 CFR 50 Appendix B criteria provided it does not represent a reduction in commitment.

Use of these criteria does not authorize:

a. Activities creating new hazards or initiators not already recognized as part of the current license basis (e.g., decontamination or decommissioning of major components defined in 10 CFR 50.82)

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

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4. The ITDC evaluation assures that the appropriate regulatory change mechanism is used for effecting the change.

c. Changes to Technical Specification requirements applicable to the current mode of operation.
d. Changes to regulations, license conditions, rules, and permits until such time that relief is granted from the regulating authority. However, it may provide the basis for requesting relief from the regulations, license conditions, rules, and permits.
e. Changes to commitments. Application of the commitment change process is required to change commitments.
f. Changes to the OQAP. However, it may provide the basis for initiating a change to the OQAP.
g. Changes to the ODCM. However, it may provide the basis for initiating a change to the ODCM.
h. Changes to the Emergency Plan. However, it may provide the basis for initiating a change to the Emergency Plan.
I. Changes to the Security Plan. However, it may provide the basis for initiating a change to the Security Plan.
j. Changes to the Fire Protection Plan. However, it may provide the basis for initiating a change to the Fire Protection Plan.
k. Changes to the Radiation Protection Program. However, it may provide the basis for initiating a change to the Radiation Protection Program.

Boundaries and Interfaces for ITDC SSCs

SSCs identified as ITDC that require "availability", must meet the following criterion:

"A system, subsystem, train, component, or device is "available" or will have "availability" when it is capable of performing its specified function(s)."

Implicit in this definition is the assumption that the necessary attendant instrumentation, controls, power sources or equipment, or other auxiliary equipment that are required to support the available SSC, are capable of performing their support function, as necessary.

Engineered Requirements for ITDC SSCs

A higher level of quality is maintained for ITDC components to assure that the capability exists to reliably meet the performance expectations and requirements. The controlled list of ITDC components is governed by plant procedures. ITDC components are not safety-related components and are not required to satisfy 10CFR50 Appendix B requirements. Although not required by regulation, the following criteria is developed and applied, as appropriate, to ITDC SSCs to assure continued reliability:

a. Design Control

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

b. Procurement Document Control

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

c. Instructions, Procedures, and Drawings

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

d. Control of Purchased Material, Equipment, and Services

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

e. Inspection

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

f. Handling, Storage and Shipping

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

g. Test Control

Surveillance testing will be established for SSCs to ensure that the SSCs perform satisfactorily commensurate with the importance of its intended safety function.

h. Measuring and Test Equipment

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

I. Corrective Action

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

3.1.2.2 Wind, Missile, and Tornado Loadings

The Maine Yankee facility is capable of withstanding the effects of severe winds or tornadoes without loss of capability of the safety systems to perform their safety functions. Section 2.2.6 discusses wind and tornado data for this region. The design tornado has a rotational velocity of 300 mph, a velocity of advance of 60 mph, and an external vacuum of 3 psig developed in 5 seconds. Thus the total effective velocity is 360 mph. Missiles may travel with the tornado equivalent to (1) a utility pole 35 ft long, 14 inches in diameter, weighing 50 lb/cu. ft. (1850 lbs total), and traveling 150 mph, or, (2) a 1-ton automobile traveling at 150 mph.

The structures housing spent fuel and adjacent structures are designed to resist the combined effects of tornado wind load, pressure drop and missile loads to produce the most critical loading condition. The original design code for the spent fuel pool and other Class I structures was ACI 318-63. The allowable stresses for shear and flexure are defined by the criteria in sections 1600 and 1700 of ACI 318-63. This specifically included the appropriate capacity reduction factor and allowable stresses. The Fuel Building is designed for protection against wind and tornado as follows:

Reinforced concrete structure

Steel superstructure

Fuel building - yard crane steel support structure, portion within spent fuel pool building only

The 6 foot thick reinforced concrete walls, which extend from 12' 6" below ground grade to 26' above ground grade, are designed to withstand the effects of tornado and missiles. A very substantial degree of added tornado and missile protection is afforded by the below-grade construction of the fuel pool. The nominal grade level (20' elev) is approximately at the same elevation as the top of the active fuel. The steel framing above the pool is designed for tornado loadings such that it will not fall into the pool and damage fuel assemblies.

The loss of water from the storage pool under the effect of a tornado would result in a maximum calculated loss of five feet of water, primarily due to vortexing. Given that the normal level of the spent fuel pool is at the 44 ft. elevation and approximately 23 ft. above the active fuel, this water loss is not significant as it is bounded by the siphoning incident.

The fire pump structure is also designed to withstand the effects of wind and tornado. The fire pumps, located at ground grade near the fire protection reservoir are tornado protected and screened from the full effects of the tornado by the dike. This is a very reliable makeup source of water to the pool.

3.1.2.3 Water Level (Flood) Design

3.1.2.3.1 Hurricane

An investigation was made to predict the probable maximum flood level which could occur at the site of the Maine Yankee Atomic Power Station on the Sheepscot River estuary when the probable maximum hurricane is taken as the design basis meteorological event. The investigation is based upon the parameters of the probable maximum hurricane as defined by U.S. Weather Bureau Report HUR 7-97, Interim Report - Meteorological Characteristics of Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States and discussed in section 2.2.6.

This investigation shows that the maximum water levels at the Maine Yankee Power Station due to the probable maximum hurricane are predicted to be at Elevation 19.9 feet and Elevation 21.4 feet on the plant site and screen well structure, respectively. These levels are based upon the simultaneous occurrence of the maximum storm surge, maximum predicted astronomical tide, an initial rise in mean sea level, estuarine amplification, the probable maximum flood in the Sheepscot watershed, maximum waves in Montsweag Bay and existence of a channel restriction at the former Cowseagan Narrows Causeway.

3.1.2.3.2 Snow

Section 2.2.5.1, Table 2.2.5 shows average snowfall statistics for Portland which are considered to be representative of the site. Structural loading and capacity reduction factors for buildings required in the defueled condition provide ample margin to accommodate snow loading.

3.1.2.3.3 Ice, Glaze, and Temperature Extremes

Ice loading is discussed in Sections 2.2.5 and 2.3.4 of this report. Glaze and ice storms usually occur in the months October through April with an average frequency of 1 to 3 storms per year. Ice thicknesses of .25 inches or .5 inches will likely occur every year whereas .75 inches is likely to occur at least every three years. Structural loading and capacity reduction factors for buildings required in the defueled condition provide ample margin to accommodate ice loading.

As discussed in section 2.2.4, the average January temperature is about 22°F with between 10 and 20 days of sub-zero temperatures occurring yearly. Temperature data representative for the site is provided in Table 2.2.2. During extended periods of freezing temperatures, it is possible that freeze damage could occur to water filled piping in buildings no longer maintained due to the defueled condition. The primary concern regarding freezing is the affect on the integrity of the pool. Pool water freezing is not possible provided that the aggregate decay heat load of the stored assemblies is reasonably high. Assurance is provided through routine operator rounds which monitor and log the temperature of the pool water. Adjacent buildings and rooms containing significant water sources which could potentially affect the safe storage of fuel are either temperature controlled to preclude freezing, or the water source is appropriately heat traced or drained, as necessary. This includes adequate administrative or design controls for protecting the integrity of the fuel transfer tube.

3.1.2.4 Seismic Design

3.1.2.4.1 Design Basis

All structures and elements of the plant are designed in accordance with sound engineering practice and are considered capable of withstanding seismic forces corresponding to a ground acceleration of at least 0.03g, in addition to normal loads, without damage or loss of function.

In addition, all structures and components of the plant which are important from the standpoint of nuclear safety and damage which could affect the health and safety of the public, i.e., "Class I" portion, are designed to meet the following criteria:

1. The design earthquake is based on a ground acceleration of 0.05g, and this portion of the plant shall be capable of operating through such earthquake.
2. The hypothetical earthquake is based on a ground acceleration of 0.1g, and this portion of the plant shall be capable of performing its intended safety function under an earthquake.
3. A spectrum analysis is used, with appropriate conservative damping factors.

3.1.2.4.2 Design Data

While 0.04g was found to be the maximum probable ground acceleration at the base of the structures on the site, it was decided to round this upward to 0.05g horizontal for design purposes.

The values for seismic design of Class I structures and components are as stated under "Design Bases." In Class I structures and components, stresses due to normal loads plus the design earthquake do not exceed those design values permitted in the applicable codes, while stresses due to normal loads plus the hypothetical earthquake do not exceed the yield stress of the affected materials. Earthquake stresses are based on a horizontal ground acceleration and a vertical ground acceleration of two-thirds of the horizontal, with the two acting simultaneously.

Design for Class I structures and components used the "response spectrum" approach in the analysis of the dynamic loads imparted by earthquakes. The seismic design is based on the acceleration response spectrum curves shown in Figure 3.1-1 for the design earthquake and Figure 3.1-2 for the hypothetical earthquake. The curves are derived from the "Housner Spectrum" normalized to 0.05g for the design earthquake and 0.10g for the hypothetical earthquake.

The design response spectra (Figures 3.1-1 and 3.1-2) are a specification of the level of seismic design acceleration, or displacement, as a function of natural period of vibration and damping level.

The response spectrum analysis is applied to all category I structures and components and groups thereof whose responses may be interdependent, considering their natural period and using appropriate damping factors as listed on Table 3.1.1.

Class I structures and components are designed in the following general manner:

1. An analysis is made to determine the natural periods of vibration of the structure using equivalent lump mass systems or distributed mass systems as is considered appropriate. In these analyses, periods and mode shapes are determined for each lumped mass mode. These data then define participation factors for each structure. Where structures are supported on their own foundations, foundation displacements are considered in determining natural periods and participation factors. It should be noted, however, that Class I structures at this site are founded on granite gneiss. Accordingly, foundation yielding will be very small and may in many cases be neglected without introducing significant error.

2. The earthquake design acceleration value for the specific natural period of the structure or component being considered is determined from Figure 3.1-1 using appropriate damping factors. The horizontal component of the ground acceleration is taken directly and the vertical component is taken as two-thirds of the horizontal value. These components are considered as acting simultaneously.

3. For certain structures, and especially for vibratory systems of a highly complex nature, such as a piping system, use of the maximum response value (peak of the curve) corresponding to the appropriate damping factor may be elected in performing the stress analysis of the system.

4. A tabulation of typical damping factors which are used for various vibratory systems important to nuclear safety is presented in Table 3.1.1. Conservative values are shown for various materials, methods of construction, and location with respect to the ground.

5. The design is then checked to verify that stresses are within acceptable limits for the hypothetical earthquake using Figure 3.1-2.

Refer to section 3.1.2 for a listing of SSCs required to be designed to Seismic Class I requirements in the defueled condition. SSCs which were previously designed to Class I criteria but are not listed in section 3.1.2, are not credited for performing a safety function (Class I) in the permanently defueled condition, and therefore, are no longer required to be designed to Class I requirements.

3.1.2.4.3 Seismic Design and Qualification

As of March 24, 1986, new Class I systems, structures, and components will be designed and qualified to the seismic demands as defined by a 0.18g NUREG/CR-0098 50th percentile Ground Response Spectra (GRS). Analytical qualification will be to the SEP allowable stress levels and damping values listed in Reference 4 (i.e., for piping, damping = 3% or PVRC, allowable stress = 2.4Sh, no OBE) up to any interface with existing structures, systems, and components. Seismic adequacy may also be demonstrated through similarity by comparison to the documented performance of equipment in natural earthquakes (Reference 5), or simulated earthquakes on testing machines.

<u>Section 3.1 References:</u>

1. Memorandum; SECY-92-223-Resolution of Deviations Identified During The Systematic Evaluation Program; S J. Chilk, Secretary, USNRC to J.M. Taylor, Sept. 18, 1992.

2. MY Letter to the NRC MN-97-89 "Certification of Cessation of Power Operation and Permanent Removal of Fuel from the Reactor, dated August 7, 1997.

3. Letter: "Seismic Design Margins Program," P.M. Sears, USNRC to J.B. Randazza, Maine Yankee Atomic Power Company, March 26, 1987; NMY 87-029

4. USNRC Letter to MYAPCO, dated March 26, 1987.

5. USNRC Letter to MYAPCO, dated February 19, 1987, (Generic Letter 87-02).

TABLE 3.1.1
EARTHQUAKE DAMPING FACTORS

	Percent of Critical Damping	
	DESIGN EARTHQUAKE	HYPOTHETICAL EARTHQUAKE
Reactor Containment	2.0	5.0
Reinforced concrete structure, other than containment, founded on soil or rock	2.0	5.0
Reinforced concrete supporting structure, not founded on soil or rock	2.0	5.0
Steel-framed structures, including supporting structures and foundations		
Bolted or Riveted	3.0	5.0
Welded	1.0	2.0
Reactor vessel, internals and control rod drives		
Welded Assemblies	1.0	1.0
Bolted Assemblies	3.0	3.0
Mechanical equipment, including pumps, fans and similar items	2.0	2.0
Piping Systems	1.0	2.0

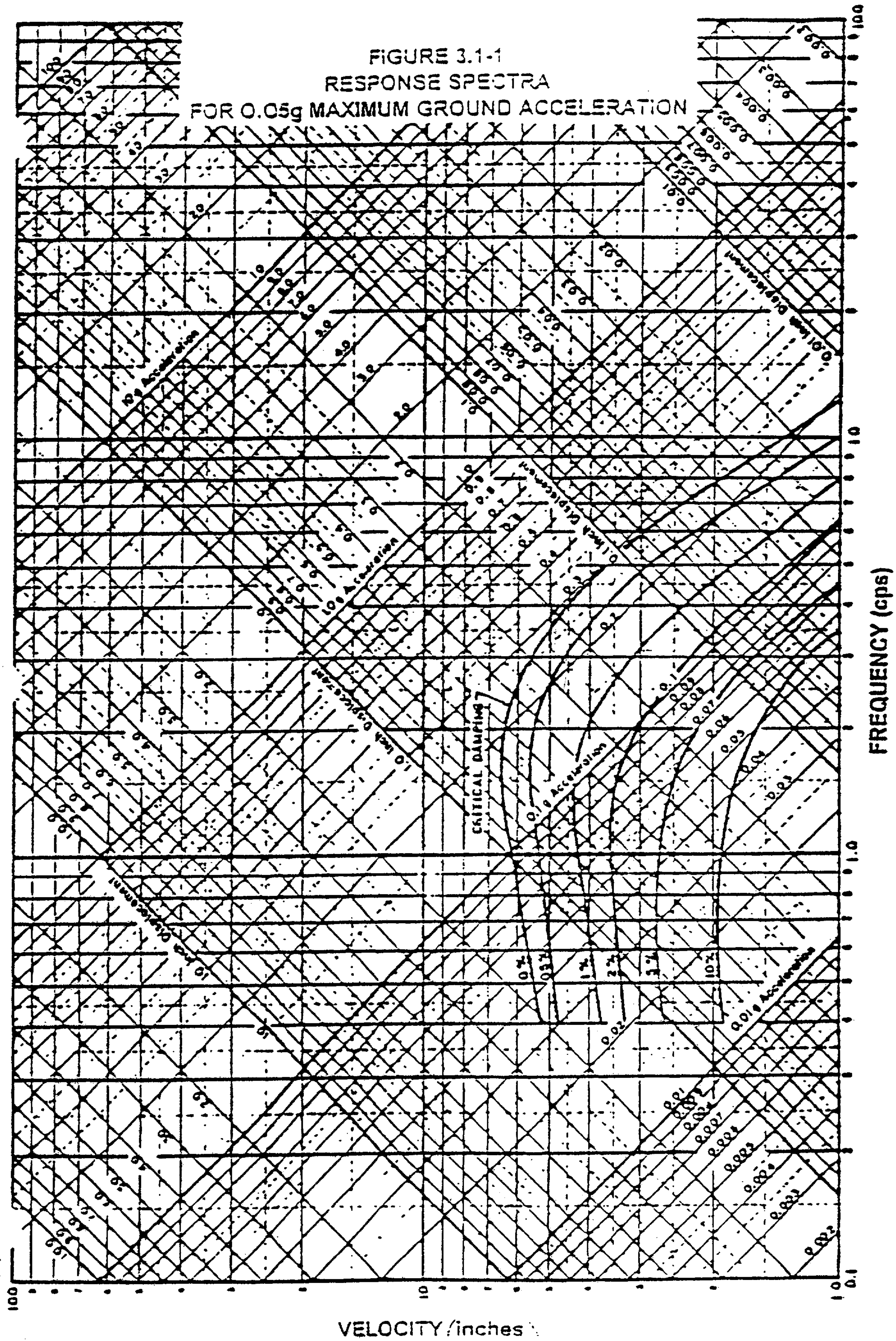


FIGURE 3.1-1
RESPONSE SPECTRA
FOR 0.05g MAXIMUM GROUND ACCELERATION

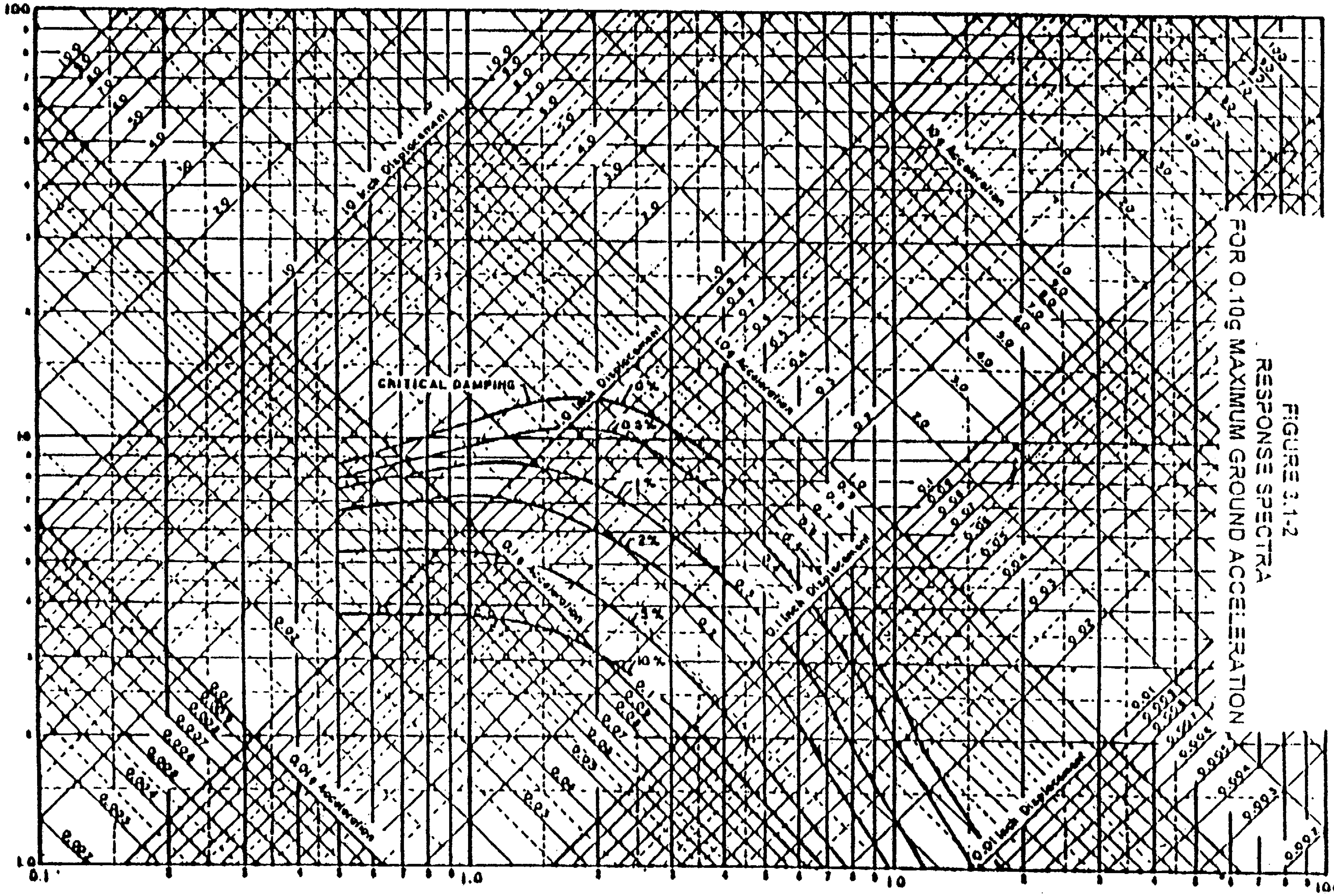


FIGURE 3.1-2
RESPONSE SPECTRA
FOR 0.10g MAXIMUM GROUND ACCELERATION

3.2 Structures

3.2.1 Fuel Building

3.2.1.1 General

The principal function of the fuel building is to provide a location for the safe storage of new and spent fuel assemblies. The building houses a new-fuel unloading area, a new-fuel storage room, a spent fuel pool and the necessary cranes required for the handling of the fuel assemblies. The spent fuel pool cooling system heat exchanger, the fuel pool cooling pumps and the fuel pool purification pump are located on Elevation 21'0". The spent fuel pool support systems are also located in the fuel building. The fuel building arrangement is shown on Figures 3.2-1 and 3.2-2.

3.2.1.2 Fuel Unloading Area

New fuel was shipped to the site in two-element shipping casks. A five-ton overhead crane in the fuel building was used to unload the shipping casks. The spent fuel pool purification system filters are located in shielded cubicles below the fuel unloading area. Shield slabs are removed from the fuel unloading floor to replace expended filter cartridge elements.

3.2.1.3 New Fuel Storage Area

The new-fuel storage room is designed for storage of 160 fuel assemblies. The fuel room is located over the spent fuel pool cooling pumps and heat exchanger. The fuel rack consists of guide sleeves symmetrically located on the floor at Elevation 31 ft. 1-1/2 in. and through the ceiling of the new-fuel room at Elevation 44 ft. 6 in. The fuel room floor has a drain opening located over the spent fuel pool cooling equipment cubicle. The floor opening prevents flooding of the new-fuel storage area. The spent fuel pool new-fuel elevator winch is also located in the new-fuel storage room.

3.2.1.4 Spent Fuel Pool

Cooling of the spent fuel assemblies during the radioactive decay period is accomplished in a stainless steel lined reinforced concrete pool filled with borated water. Space is provided in the pool to place the spent fuel shipping cask. The pool is serviced by means of the yard crane, as well as a moveable platform with hoist. A new-fuel area adjoins the spent fuel pool. The pool is designed to safely resist the hypothetical earthquake or tornado, as well as the applied loads of the water and fuel.

The pool has a reinforced concrete floor founded on rock and sidewalls 6 feet thick which extend from 12 feet 6 inches below ground grade to 26 feet above ground grade. The concrete is reinforced with #11 bars at 12 inch center to center spacing with a yield strength of 40,000 psi. The concrete has a 28 day minimum compressive strength of 3,000 psi. The reinforced spent fuel pool was originally designed in accordance with ACI-318-63 to resist the appropriate dead, live, hydrostatic and maximum hypothetical seismic loadings. The structure was reanalyzed, in support of EDCR 92-111, to demonstrate the acceptability of installing the new high density spent fuel storage racks.

As part of the preliminary decommissioning activities, the structural evaluations have been performed which demonstrate the adequacy of the SFP concrete and liner to withstand the effects of dead, live and hydrostatic forces in conjunction with an elevated pool water temperature of 212°F. Complete details of this evaluation are contained in References 3.2-1 and 3.2-2.

The pool is completely lined with plates of stainless steel which have test channels behind each weld. The test channels are piped to the spent resin pit sump through four-1 inch tell tale pipes, each with a flow limiter at the end of the pipe. In the event of a malfunction of a liner weld, the leakage through each telltale is limited to less than 2.5 gpm.

The liner is designed as a ASME Section III, Division 2, Paragraph CC-3720, Liner, Table CC-3720-1, Service Category, Membrane. The plate material is ASTM A240, Type 304 stainless steel. Liner Anchors are designed to ASME Section III, Division 2, Paragraph CC-3730 and are constructed of ASTM A-36 steel. The weld rods used to weld the vertical stiffener flanges to the liner wall liner were ASTM E309 (carbon to stainless steel) with a minimum tensile strength of 81,000 psi.

The fuel transfer tube was originally designed as safety class 2; however, since the containment integrity design basis is not applicable in the defueled condition, it has been reclassified as safety class 3. It consists of a 36-inch OD, 3/8 inch thick, ASTM A312 TP304, stainless steel pipe installed inside a 40-inch OD stainless steel sleeve as shown in detail on Figure 3.2-13. The inner pipe acts as the transfer tube and connects the containment refueling canal with the spent fuel pool and is welded to the fuel pool stainless steel liner. The outer pipe is fitted with bellows expansion joints, backed up by a packed slip joint to compensate for any differential movement.

Structural steel supports a superstructure of protected metal siding which encloses the pool. The steel framing above the pool is designed for earthquake and tornado to prevent it from falling into the pool and damaging fuel assemblies. The masonry wall at the south end of the fuel building is not designed for certain wind or earthquake loadings, and, therefore, an evaluation of the consequences of a wall collapse was performed. The analysis demonstrated adequate spent fuel pool cooling capability and structural rack integrity.

3.2.1.5 Fuel Storage Racks

The new and spent fuel pool structures including fuel racks are designed to withstand the anticipated earthquake loadings as Class I structures in accordance with the guidance of Regulatory Guide 1.29. Analyses show that the racks will perform their intended function under both seismic and load drop loadings in accordance with Regulatory Guide 1.124 and NUREG-0800. The design ensures that during the event, rack-to-rack and rack-to-wall interaction is appropriately considered. Structural material used in the rack design is ASME Section II, SA 240, Type 304 stainless steel. The design considered thermal loads induced by an operating temperature of 154°F. Subsequently an evaluation was performed which documented the acceptability of the racks at a temperature of 212°F. The ANSYS version 4.4A program was used for all computer aided mechanical analysis.

The design considered impact loads from a 2500 lb. (submerged weight) fuel element dropped from 18 inches above a module, a fuel element hangup during removal, and the load induced if an assembly hit the top of a rack while moving at the maximum horizontal velocity of the crane. Subsequent reanalysis considered impact loads from a 2000 lb. (submerged weight) fuel element dropped from 22.5 inches above a module. Subcriticality and a coolable geometry are maintained and damage to the stored fuel is minimized.

The racks consist of individual storage cells joined into a rack module. The racks are a single tier, rectilinear array of free standing modules, not anchored to the pool walls, floor or adjoining racks. Each rack module is provided with adjustable support feet. Each fuel rack is a folded metal plate assembly of 14 gage metal, approximately 180 inches high, 117 inches wide and 128 inches deep. The folded metal plate assembly is welded to a baseplate, which is supported by adjustable supported feet. Region I contains 5 racks, spaced on a minimum of 10.5 inch centers. Region II contains 18 racks spaced on a minimum of 9 inch centers. Spent fuel storage racks may be moved]
only in accordance with written procedures which ensures that no rack modules are moved over fuel assemblies.

3.2.2 Storage Buildings

3.2.2.1 Underground RCA Storage Bunker

The underground RCA storage bunker, also referred to as the high rad bunker, is located within the protected area about 120 feet northwest of containment. The bunker is a reinforced concrete structure and is partially buried below yard grade. The bunker is 27.5 feet by 16 feet and approximately 12 feet high. The top of the bunker is about 5 feet higher than the surrounding yard grade. The bunker is divided into five internal compartments each separated by 18 inch thick concrete walls. The exterior bunker walls vary from 12 to 18 inches in thickness. The roof of the bunker consists of six removable 18 inch thick concrete roof plugs.

A floor drain system directs any liquids collected in the bunker to a sump. Liquids collected in the sump may be pumped to the spent resin pit sump in the nearby RCA storage building through an underground pipe or temporary hose. The spent resin pit sump discharges to the Liquid Waste Holdup Tank (TK-109) where any liquids would be collected for processing.

The bunker provides temporary storage for radioactive wastes before they are moved to the low level waste storage building or shipped for processing and disposal. The outside yard crane is available to move waste containers in and out of the bunker. Waste may be stored in the bunker to allow for some decay before being placed in the low level waste storage building or until arrangements are made for permanent off-site disposal.

For purposes of decommissioning the high rad bunker has been abandoned. However, the Decommissioning Operations Contractor may use the facility for short term waste storage. Its description will be retained until the facility is no longer in use.

3.2.2.2 Radiation Controlled Area (RCA) Storage Building

The RCA storage building, adjacent to the fuel building houses the Duratek waste processing skid, a decontamination area and a waste solidification area. The waste solidification area is no longer operable. The Duratek skid has been modified to support processing of waste liquid from decommissioning activities. Tank 109 (TK-109) has been converted from resin hold up to liquid waste hold up. TK-109 is used to feed the Duratek processing system. The decontamination area provides an area to support fuel pool cask decontamination as well as general equipment decontamination. The area is tied into the Fuel Building exhaust ventilation system and is kept under a slight negative pressure.]

3.2.2.3 Low Specific Activity (LSA) Storage Building

This building houses the LSA compactor and served as the storage building for LSA containers to keep them free of damage or deterioration. It extends from the south wall of the RCA building to the containment, but is not structurally attached to either. The building is designed non-nuclear safety. The LSA compactor compresses LSA material. It is vented into the RCA filtered ventilation system. The LSA sump pump discharges to the spent resin pit sump in the RCA Building.

3.2.2.4 Warehouse

This building is located outside the RCA. The building is used to store replacement components and equipment.

]

3.2.3 Service Building

The Service Building has been abandoned in its entirety by Maine Yankee and no longer serves any function important to the safe storage of fuel.

3.2.3.1 Control Room Area

The main control room, located in the Gate House, contains the controls and instrumentation necessary to monitor and control various areas and equipment required for safe storage of spent fuel. The main control room is designed to be available at all times. The decayed source term in the defueled condition, in conjunction with the location and design of the control area, provides sufficient protection to ensure that control room personnel will not be subjected to doses which would exceed 10CFR20 limits. Equipment in this area has been designed to minimize the possibility of a condition which could lead to possible inaccessibility or evacuation. In the event that this area becomes inaccessible, the controls for spent fuel pool cooling and makeup, water treatment and waste disposal, are located at control stations remote from the main control room.

3.2.4 Turbine Building

The turbine building housed the secondary plant components and systems. Though the turbine building is not a Class I seismic structure, it is designed for wind forces which are greater than the seismic forces as determined from a combined seismic analysis of the service building, control room and the turbine hall. The turbine hall will remain intact during the DBE.

The Turbine hall is considered partially abandoned with certain restrictions. This structure provides support on the north and east exterior walls for SFPI control cabling. Demolition activities are allowed while ensuring interfaces with available SSCs (front office hallway structure, administration building structure, fire suppression systems, and designated orange marked equipment on the north and east interior and exterior walls) are not interfered with.

3.2.5 Primary Auxiliary Building (PAB)

All exterior and interior concrete walls and slabs are designed to safely resist the hypothetical earthquake.

The PAB has been partially abandoned with certain restrictions. This assessment of the PAB will allow for modifications and limited demolition activities to this structure as long as 1) containment closure can be established to meet the requirements of the DSAR, 2) the integrity and seismic stability of the Spent Fuel Pool and support equipment is not challenged and 3) all effluents are via a monitored and approved release path.

3.2.6 Service Water Intake Structure

The Service Water Intake Structure has been abandoned and serves no function relating to the safe storage of spent fuel.

3.2.7 Fire Pump Building

The fire pump house is shown on Figure 3.2-12. It is located near the water storage pond and houses the equipment servicing the liquid portion of the fire protection system. The building houses two fire pumps, a pressure maintenance pump and a hydro-pneumatic tank. A diesel engine drives
one pump while the other is motor-driven. The building houses the diesel fuel tank, the batteries and control board required for the diesel operation.

3.2.8 Masonry Walls

Masonry walls in the vicinity of safety-related equipment have been evaluated to determine whether they will withstand all postulated design loads (seismic, tornado, hurricane, attached equipment, etc.). Table 3.2.1 lists the walls evaluated and notes those which are fully qualified and those which are assumed to fail, but such failure will not result in unacceptable damage to equipment. Bulletin 80-11 Category Definitions:

Category 3: Walls whose collapse will not affect safety-related equipment.

This Table has been revised to address only those walls that are required in the permanently defueled condition.

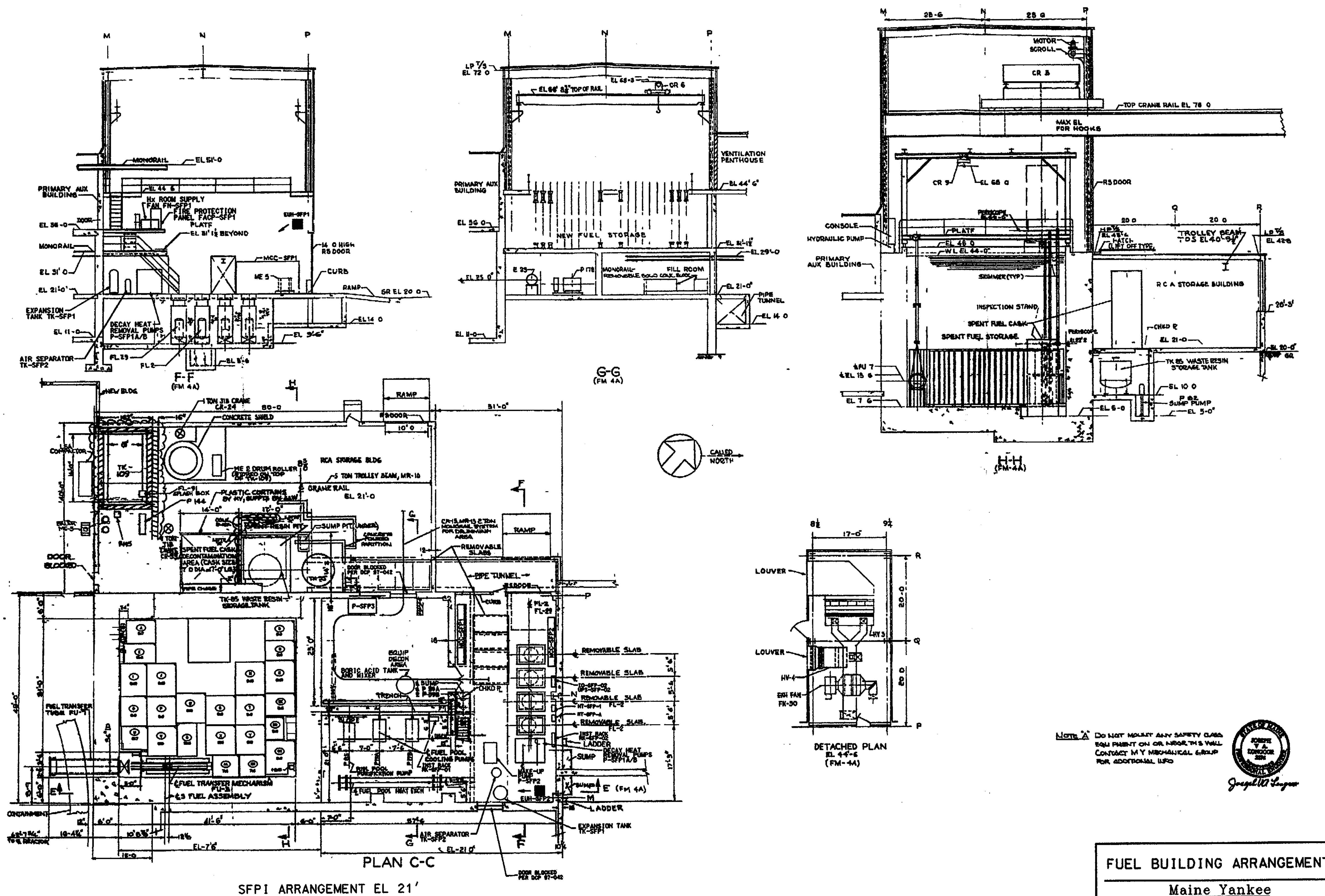
Table 3.2.1

BUILDING	WALL ID	ROOM LOCATION	REFERENCE DRAWING	WALL LOCATION	EQUIPMENT PROTECTED	CATEGORY
Fuel	FB 44 1	Spent Fuel Pool Elev 44'-6"	11550-FA-12A	South Wall Spent Fuel Building	Spent Fuel Purification and Cooling Return Lines and spent fuel storage racks.	3

Section 3.2 - References:

1. YNSD Letter to R. Fraser from D.L. Magnarelli/W.E. Henries, "Review of AES Analysis of SFP for Elevated Temperatures," dated Nov. 25, 1997.

2. YNSD Calculation No. MYC-2001, "Analysis of SFP Structure for Elevated Pool Water Temperature," dated November 25, 1997.

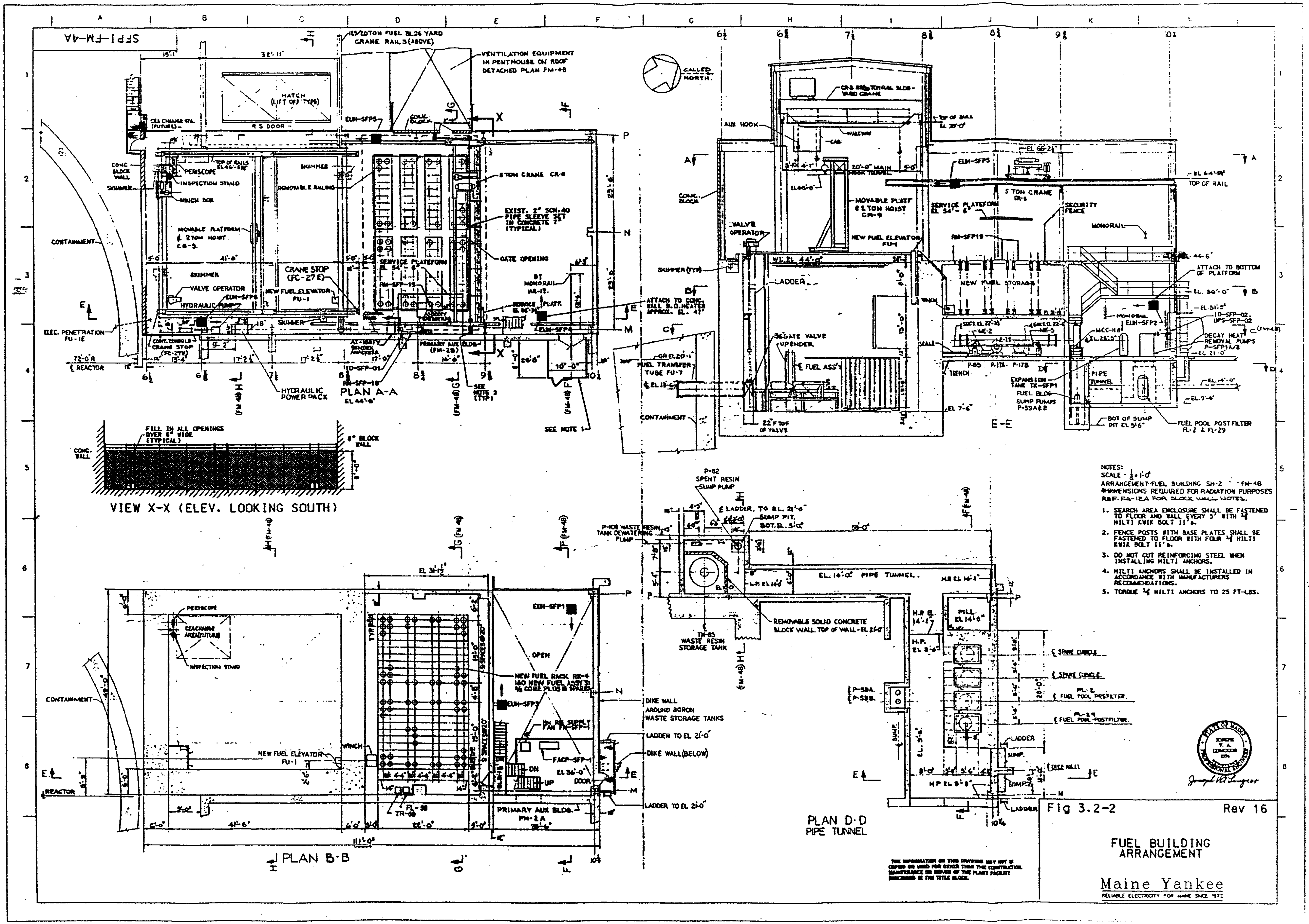


FUEL BUILDING ARRANGEMENT

Maine Yankee

REV.	DATE	BY	CHKD	APPD
0	05/23/01	MMM		

DRAWING NUMBER FIG 3.2-1, REV. 17



SFPI-FM-4A
125/20 TON FUEL BLDG YARD CRANE RAILS (ABOVE)
VENTILATION EQUIPMENT IN PENTHOUSE ON ROOF DETACHED PLAN FM-4B
HATCH (LIFT OFF TYPE)
R.S. DOOR
EUH-SFP5
PERISCOPE
INSPECTION STAND
SKIMMER
REMOVABLE RAILING
5 TON CRANE CR-6
EXIST. 2" SCH.40 PIPE SLEEVE SET IN CONCRETE 7" (TYPICAL)
MOVABLE PLATFORM & 2 TON HOIST CR-9
GATE OPENING
CONTAINMENT
VALVE OPERATOR
HYDRAULIC PUMP
CRANE STOP (FC-27E)
NEW FUEL ELEVATOR FU-1
SERVICE PLATFORM EL 54'-6"
MONORAIL
ATTACH TO CONC. WALL B.O. HEATER APPROX. EL. 47'
ELEC. PENETRATION FU-1E
CRANE STOP (FC-27E)
REACTOR
PRIMARY AUX BLDG (FM-2B)
HYDRAULIC POWER PACK
PLAN A-A
EL 44'-6"
SEE NOTE 2 (TYP)
SEE NOTE 1
FILL IN ALL OPENINGS OVER 6" WIDE (TYPICAL)
CONC. WALL
8" BLOCK WALL
VIEW X-X (ELEV. LOOKING SOUTH)
CALLED NORTH
AUX HOOK
CAB
WALKWAY
TOP OF RAIL
20'-0" MAIN HOOK TRAVEL
MOVABLE PLATF & 2 TON HOIST CR-9
SERVICE PLATFORM EL 54'-6"
5 TON CRANE CR-6
SECURITY FENCE
MONORAIL
NEW FUEL ELEVATOR FU-1
CONC. BLOCK
VALVE OPERATOR
SKIMMER (TYP)
W.L. EL 44'-0"
LADDER
NEW FUEL STORAGE
ATTACH TO BOTTOM OF PLATFORM
DECAY HEAT REMOVAL PUMPS P-SFP1A/B
GATE VALVE UPENDER
FUEL ASS'Y
FUEL TRANSFER TUBE FU-7
CONTAINMENT
22'F TOP OF VALVE
EXPANSION TANK TK-SFP1
FUEL BLDG SUMP PUMPS P-39A&B
PIPE TUNNEL
BOT OF SUMP PIT EL 5'-6"
FUEL POOL POSTFILTER FL-2 & FL-29
E-E
P-82 SPENT RESIN SUMP PUMP
LADDER TO EL 21'-0"
SUMP PIT
P-108 WASTE RESIN TANK DEWATERING PUMP
EL. 14'-0" PIPE TUNNEL
REMOVABLE SOLID CONCRETE BLOCK WALL TOP OF WALL-EL 21'-0"
TK-85 WASTE RESIN STORAGE TANK
SPARE CUBICLE
FUEL POOL PREFILTER
FUEL POOL POSTFILTER
DIKE WALL
PLAN D-D
PIPE TUNNEL
PERISCOPE
CEACHAMA AREA (FUTURE)
INSPECTION STAND
CONTAINMENT
NEW FUEL ELEVATOR FU-1
WINCH
OPEN
EUH-SFP1
NEW FUEL RACK RK-4 140 NEW FUEL ASSY'S 1/2 CORE PLUS 8 SPARES
DIKE WALL AROUND BORON WASTE STORAGE TANKS
LADDER TO EL 21'-0"
DIKE WALL (BELOW)
PRIMARY AUX BLDG. FM-2A
PLAN B-B
NOTES:
SCALE - 1/8" = 1'-0"
ARRANGEMENT-FUEL BUILDING SH-2 - FM-4B
DIMENSIONS REQUIRED FOR RADIATION PURPOSES
REF. FA-12A FOR BLOCK WALL NOTES.
1. SEARCH AREA ENCLOSURE SHALL BE FASTENED TO FLOOR AND WALL EVERY 3' WITH 3/8 HILTI KWIK BOLT II's.
2. FENCE POSTS WITH BASE PLATES SHALL BE FASTENED TO FLOOR WITH FOUR 1/4 HILTI KWIK BOLT II's.
3. DO NOT CUT REINFORCING STEEL WHEN INSTALLING HILTI ANCHORS.
4. HILTI ANCHORS SHALL BE INSTALLED IN ACCORDANCE WITH MANUFACTURERS RECOMMENDATIONS.
5. TORQUE 1/4 HILTI ANCHORS TO 25 FT-LBS.
Fig 3.2-2
Rev 16
FUEL BUILDING ARRANGEMENT
Maine Yankee
RELIABLE ELECTRICITY FOR MAINE SINCE 1972

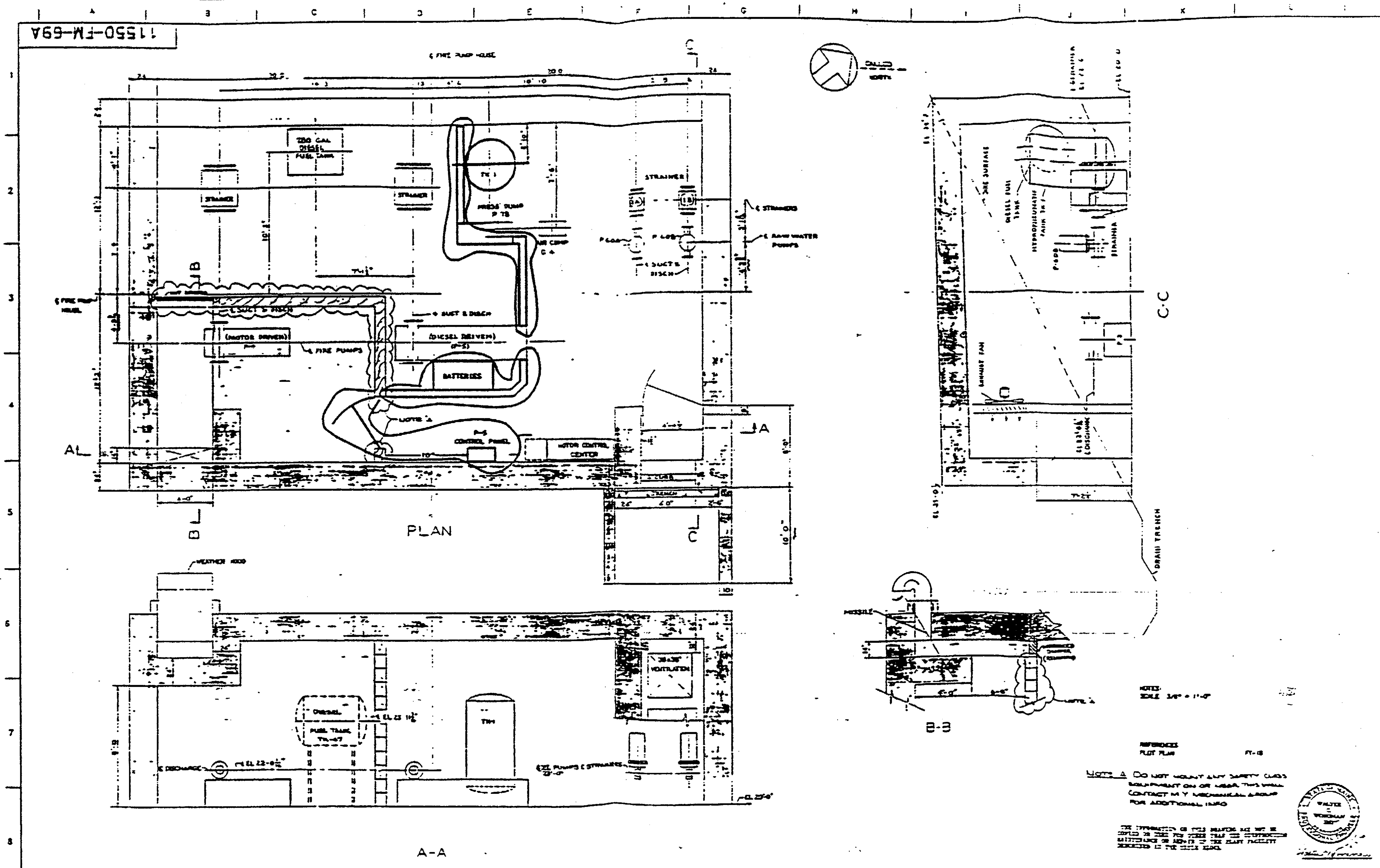


Fig. 3.2-12 Rev. 14

FIRE PUMP HOUSE

Maine Yankee

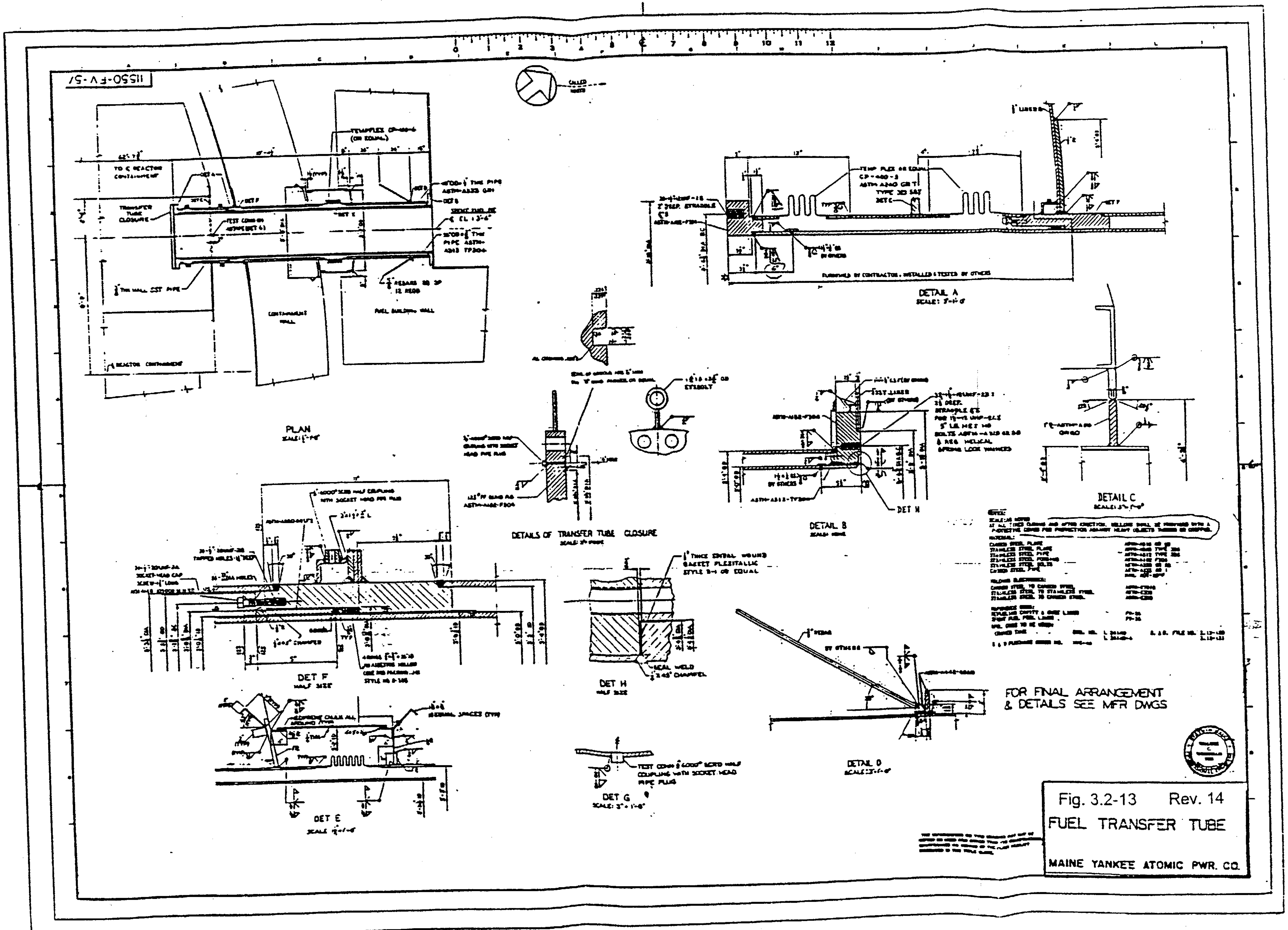


Fig. 3.2-13 Rev. 14

FUEL TRANSFER TUBE

MAINE YANKEE ATOMIC PWR. CO.

3.3 **Systems**

3.3.1 Fuel Storage

3.3.1.1 Fuel Storage Design Basis

Design Criteria

The fuel storage design criteria are as follows:
1. Criticality in new and spent fuel storage is prevented by physical design features or processes. A geometrically safe configuration is emphasized over procedural controls.
2. Appropriate shielding is provided to meet the requirements of 10CFR 20.
3. The fuel building is continuously monitored by area specific detectors. Audible and visual alarms are activated at the detector locations and in the Control Room for radiation levels in excess of predetermined limits.
4. Spent fuel storage systems are:
   a. designed to prevent or mitigate accidents which could lead to the release of significant amounts of radioactivity affecting the public health and safety,
   b. designed, fabricated and erected to withstand, the additional forces that might be imposed by natural phenomena.

The spent fuel pool cooling, make-up, and purification system design criteria is as follows:
1. Prevent damage to spent fuel which could cause radioactive release to plant operating areas or the public environs.
2. Maintain adequate level of pool water to provide shielding for radiation protection.
3. Maintain clarity and purity of the borated water in the spent fuel pool.
4. The spent fuel pool is equipped with a high and low liquid level alarm, and a high temperature alarm which will indicate loss of continuity in decay heat removal.

Design Basis

The fuel storage and handling design basis is as follows:

a. Store 1568 complete spent fuel assemblies. In addition, other irradiated components such as CEAs are stored in the fuel assemblies. The total maximum decay heat load to be stored in the pool is $\leq 5.96 \times 10^6$ BTU/hr (BTP ASB 9-2 as of 10/15/97). This assures that the calculated time-to-boil and boil-
off rates are adequate to provide ample time for operators to take remedial actions in the event of a loss of forced cooling incident.]

Rev. 18]

b. The design basis for preventing criticality is that, considering the possible variations, there is 95% probability at a 95% confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 for flooded conditions without credit for borated water.

c. A minimum boron concentration of 670 ppm was assumed in the analysis for the postulated misplaced fuel assembly. A higher administrative limit is required by technical specifications. The minimum concentration was determined to be required to maintain a K_{eff} less than .95 for the most reactive accident condition. This concentration is required whenever fuel assemblies are stored in the pool and a spent fuel pool assembly placement verification has not been performed since the last movement of fuel assemblies in the spent fuel pool. The analysis considered various misposition configurations including a fuel assembly laying horizontally on the top of the spent fuel racks, a fuel assembly placed adjacent to or on the outside corner of the fuel racks and a fuel assembly placed in the wrong rack.

d. Maintain fuel cladding integrity in the event forced cooling is lost and cooling occurs by SFP boiling (212°F) at the water surface. The cooling water make-up rate exceeds evaporative losses. Makeup supplies are available in the event of an extended loss of offsite power incident.

e. The new and spent fuel pool structures, including fuel racks, are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pool is lined with stainless steel to ensure against loss of water.

f. The spent fuel racks are designed to Seismic Class I requirements, capable of sustaining a temperature of 212°F, and able to withstand the dropping of a 2500 lb. (submerged weight) assembly from a height of 18" above the top of the racks without incurring damage which could result in criticality. A subsequent evaluation has also shown that the racks are able to withstand the dropping of a 2000 lb. (submerged weight) fuel assembly from a height of 22.5 inches above the top of the racks without incurring damage which could result in criticality or spent fuel cooling concerns.

g. The spent fuel pool concrete structure is a Seismic Class I structure and capable of sustaining a service temperature of 212°F. The steel framing above the pool is a Seismic Class I structure and is tornado resistant such that it will not fall into the pool and damage fuel assemblies (Note: The tornado resistant design evaluated full tornado wind pressures on the bare structural steel frame).

h. The dose rates at the surface of the SFP from spent fuel assemblies do not exceed 2.5 mrem/hr in fuel building passageways during normal storage (i.e., 50 mrem/hr during fuel handling operations). Dose rates at the outside surface of the walls adjacent to the spent fuel area do not exceed the maximum radiation zone level for the area.

i. Siphon protection is provided to preclude inadvertent draining of the SFP to a level lower than approximately 10 feet above the top of the active fuel. Additional margin (to approximately 19') was provided by the installation of a branch syphon break.

j. The minimum spent fuel pool water bulk temperature shall be 40°F.

k. Spent fuel storage racks may be moved only in accordance with written procedures which ensures that no rack modules are moved over fuel assemblies.

3.3.1.2 Fuel Pool Storage System Description

The fuel pool storage and handling system is designed to prevent or mitigate the consequences of accidents associated with the storage of fuel. A description of the fuel and irradiated components is presented below followed by the structures, systems, and components supporting the safe storage and handling of the spent fuel and irradiated components. A description of the spent fuel handling system equipment is contained in section 3.3.2.

3.3.1.2.1 Fuel Assemblies (Typical)

Figure 3.3-1 shows the details of the fuel assembly. Except for the two consolidated assemblies, all assemblies consist of five guide tubes, nine fuel spacer grids, upper and lower end fittings, fuel rods and burnable absorber rods, if required. The number and type of burnable absorber rods in each type of assembly were cycle dependent. The structural frame of the assembly consists of the guide tubes, spacer grids and end fittings. The four outer guide tubes are mechanically attached to the stainless steel end fittings and the spacer grids are welded or mechanically attached to all five guide tubes.

A sleeve is provided in the upper region of the guide tube. The sleeve is made of stainless steel and is chrome plated on the ID. The sleeve is positively located within the guide tubes and components of the upper end fitting. The lower end fitting is a cast stainless steel structure. The lower end fitting contains flow holes for the fuel rods, and positioning holes for the guide tubes.

A hold-down device is incorporated into the upper end fitting and features a hold-down plate which acts on the underside of the fuel alignment plate (Figure 3.3-1). The hold-down plate is loaded by coil springs which are located at each of the guide tube posts. The Inconel springs are positioned at the upper end of the assembly so that the spring load combines with the assembly weight in counteracting the upward hydraulic forces. The springs are sized and spring preload selected, such that a sufficient net downward force will be maintained for all normal and anticipated transient flow and temperatures conditions.

The fuel assembly upper end fitting is a cast stainless steel structure. It serves as an attachment for the guide tubes and as the lifting fixture.

The fuel rod spacer grids (Figure 3.3-2) maintain the fuel rod pitch over the full length of the fuel rods. Typical grids are fabricated from preformed strips interlocked in an egg crate fashion and welded together. Each fuel rod is supported by springs and arches opposite the springs. The springs press the rod against the arches to restrict relative motion between the grids and the fuel rods. The springs and arch positions may be reversed from grid to grid to provide additional restriction to relative motion. The perimeter strips also contain springs and arches in addition to special features to prevent hang up of grids during a refueling operation.

There are two types of rods contained within the typical assembly: fuel rods and discrete burnable absorber rods. A brief description of each type of rod is contained below.

Fuel Rod

The fuel rods consist of UO_2 pellets, a compression spring and in the ABB-CE and EXXON design spacer discs, all encapsulated within a Zircaloy-4 or ZIRLO tube. The UO_2 pellets are dished at both ends to accommodate the effects of thermal expansion and swelling.

The fuel cladding is slightly cold worked Zircaloy-4 or ZIRLO tubing. The diametrical gap between the pellet and clad ID is chosen to meet design criteria regarding clad stresses and strains, and transfer of heat from the pellets. The compression spring located at the top of the fuel pellet column maintains the column in its proper position during handling and shipping. It also provides support for the clad in the plenum region to prevent local buckling.

In the ABB-CE and EXXON designs, an alumina spacer is located at the upper end of the fuel pellet stack to insulate the plenum region and to prevent UO_2 chips from entering the plenum region. In ABB-CE fuel, an alumina spacer is located at the bottom of the fuel pellet stack to reduce lower end cap temperature. The plenum above the pellet column provides space for axial thermal expansion of the fuel column and for accommodation of fission gas.

Discrete Burnable Absorber Rods Description

Discrete burnable absorber rods are mechanically similar to fuel rods except that they each contain a column of burnable poison pellets instead of fuel pellets. The poison material is alumina with uniformly dispersed boron carbide particles. The poison material is a 122.7 inch long column approximately centered within the 136 to 137 inch long active fuel region.

Components stored within the fuel assembly guide tubes in the spent fuel pool are: control element assembly and neutron source assembly. A description of each is provided below.

Control Element Assembly

The CEA (shown in Figure 3.3-3) comprised of five Inconel tubes 0.948 inches in diameter. Each tube (finger) is sealed by welded end caps. A gas expansion space is provided to limit maximum tubes stress due to internal pressure developed by the release of helium gas and moisture from the boron carbide. The overall length of the control elements assembly is approximately 161-5/16 inches. Four fingers are assembled in a square array around the centrally located fifth finger. The fingers are joined by an upper end fitting.

Most CEAs are full-strength and comprise five active fingers containing B_4C pellets. The four outer fingers on all CEAs and all five fingers on most CEAs have silver-indium-cadmium filled tips. Some CEAs are designed as part-strength CEAs, utilizing stainless steel pellets in place of B_4C in selected fingers. The full strength and part-strength CEAs are described in Figure 3.3-4.

Neutron Source Assemblies

Between two and four neutron source assemblies were installed in the reactor for each cycle. Each source may include both a startup and sustainer source, or just a sustainer source. The sources are held in vacant CEA guide tubes by means of protrusions which fit into slots cut into the top of the upper end fitting posts.

3.3.1.2.2 Fuel Storage System Description

This section identifies the SSCs which comprise the fuel storage system. A description of each is contained below.

- Fuel pool
- Transfer tube and isolation valve
- Spent fuel storage racks
- New fuel storage

Fuel Pool

The fuel pool, 37 feet wide by 41 feet long by 38 feet deep, is located in the south end of the fuel building adjacent to the reactor containment. The pool is constructed of reinforced concrete with a wall and floor lining of 1/4-inch thick stainless steel. The liner ensures another level of defense against water leakage through the structure. Welds in the liner are backed up by test channels which are piped to the spent resin pit sump. An area is provided in the fuel pool for inspection of fuel assemblies and space is also provided for a spent fuel cask. The spent fuel pool volume is approximately 59,116 ft^3.

The fuel pool is protected from being drained down to a level approximately 10[(1)] feet above the active fuel assuming a rupture of any pipe normally connected to the pool. The consequences of the siphoning incident is discussed in section 3.3.1.3. The anti-siphon device on the suction side of the fuel pool cooling pumps is safety class 3, seismically designed and periodically inspected for debris.

The design of the pool, in conjunction with analyses concluding that the MY spent fuel pool concrete structure and liner are capable of resisting thermal stresses due to prolonged elevated pool water temperatures of 212° F without failure, provides reasonable assurance that the pool water level will not be affected by credible incidents other than evaporative losses due to normal and loss of forced cooling incidents.

Transfer Tube and Isolation Valve

The fuel transfer tube penetration is provided for fuel movement between the refueling canal in the containment and the spent fuel pool. The transfer tube design prevents draindown through two independent mechanical devices; by an isolation valve (FP-21) located in the spent fuel pool and a blind flange seal at the refueling canal end.

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[1]As noted in Section 3, a branch syphon break has been installed which increases the margin to approximately 19'.

Spent Fuel Storage Racks

The 23 stainless steel, poison curtain storage racks provide for the storage of a maximum of 1568 fuel assemblies. The fuel pool is separated into two regions. Region I contains five racks to store up to 228 fuel assemblies, spaced on a minimum of 10.5 inch centers and is designed for initial enrichments up to 4.5 weight percent U-235. Region II contains 18 racks to store up to 1340 fuel assemblies, spaced on a minimum of 9 inch centers. Figure 3.3-6 shows the high density spent fuel pool layout for the two region pool. The spent fuel is stored in borated water. Water is necessary to provide shielding and cooling for removal of the decay heat being released by the spent fuel assemblies. The new and spent fuel pool structures including fuel racks are designed to withstand the anticipated earthquake loadings as Class I structures. Refer to Figure 3.3-5 for a description of restrictions for fuel which may be placed in each region.

The racks are a single tier, rectilinear array of free standing modules and are seismically designed. The design ensures that during the event, adequate distances between the assemblies is maintained to prevent a criticality event and to prevent adverse interaction between the pool structure and the racks. Structural material used in the rack design is ASME Section II, SA 240, Type 304 or ASTM A240 Type 304L stainless steel.

The poison material selected as a neutron absorber is Boral; a cement composite material made of boron carbide particles in a Type 1100 aluminum matrix. Boral Panels consist of 2 outer sheets of type 1100 aluminum that clad a sintered plate of boron carbide in a type 1100 aluminum matrix.

New Fuel Storage

The new fuel storage area, located at elevation 31 feet in the fuel building, contains facilities to receive, handle, and store new fuel assemblies up to 5.5 weight percent U-235. The new fuel is stored dry in racks that have a center-to-center spacing of 20 inches. Although not applicable to the defueled condition, there are provisions for storing 160 fuel assemblies in the new fuel storage area. New fuel may not be placed in the pool.

3.3.1.2.4 Fuel Pool Cooling, Makeup and Purification System Description

Fuel Pool Heat Exchanger

The fuel pool heat exchanger is a cross-flow heat exchanger of the shell-and-tube design. The heat exchanger is designed to cool the fuel pool water based on the following conditions:

Shell Side	Tube Side
108°F in and 111°F out	116°F in and 113°F out
2.88 x 10^6 BTU/hr heat transfer	2.88 x 10^6 Btu/hr heat transfer

The fuel pool heat exchanger is designed for 150 psig and 225°F on both the shell and tube sides. The tubes are 304 stainless steel, and the shell is of carbon steel. The above values are based on a tubeside flow of 1700 gpm, a shellside flow of 1000 gpm and an outside temperature of 87°F and the calculated heat load as of June 22, 1999.

Fuel Pool Cooling Pumps

There are two fuel pool cooling pumps designed for maximum efficiency at 772 gpm per pump at a total pump head of approximately 80 ft. The pumps may be operated at other flows and heads. The casing is designed for 125 psi at 250°F. The internal wetted surfaces of the pumps are type 316 stainless steel or an equivalent cast form. The pumps are installed for parallel operation.

Fuel Pool Purification Pump

The fuel pool purification pump is designed for 250 gpm. The casing is designed for 200 psig at 250°F. The internal wetted surfaces of the pump are type 316 stainless steel or an equivalent cast form. The pump is used for fuel pool cooling when the heat load is low, and it can also be used for skimming operations. Skimmers are provided to collect precipitants on the water surface and prevent it's build-up on the pool wall surface.

Fuel Pool Filters

The fuel pool prefilter and post filter are of the cartridge type and are located in the purification loop to remove the particulate or undissolved solids. The prefilter retains 95% of the particles nominally 1 micron, or larger, at a flow of 200 gpm. The vessel is constructed to Type 304 stainless steel and is rated at 200 psig and 250°F. The post filter retains 95% of the particles nominally 1 micron, or larger, at a flow of 200 gpm. The vessel is constructed to Type 304 stainless steel and is rated at 200 psig and 250°F.

Fuel Pool Demineralizer

A mixed bed, underwater demineralizer is located in the spent fuel pool to remove dissolved solids from the water by ionic exchange with the resin. The vessel is constructed primarily of Type 304 stainless steel and has a design pressure of 50 psig. The vessel normally rests on the fuel pool floor at the 7.5 ft. elevation and circulates fuel pool water, at a nominal rate of 100 gpm, using an internally mounted pump and motor. Fuel pool water is drawn into the vessel by the pump, through two inlet connections, and through the process bed. Water is discharged from the vessel through a hose assembly which distributes the water to a remote area of the fuel pool, ensuring proper pool circulation. The demineralizer is sized to contain approximately 28 cubic feet of process media.

Fuel Pool System Valves

All valves in the fuel pool system are austenitic stainless steel. The gate and check valves are butt-welded. The diaphragm and ball valves are socket-welded.

Fuel Pool System Piping

All the piping used in the fuel pool system is type 304 stainless steel with welded connections throughout, except for flanged connections at the pump suction and discharge.

Code Requirements

The design of all components in the fuel pool cooling system complies with the following codes and regulations:

Fuel Pool Heat Exchanger	Tube side ASME Section III Class C Paragraph UW-2(a) of Section VIII applies Shell side ASME Section VIII
Fuel Pool Cooling Pumps	No code
Fuel Pool Purification Pump	No code
Fuel Pool Filters	ASME Section III Class C Paragraph UW-2(a) of Section VIII applies
Underwater Demineralizer	ASME Section VIII, Not Stamped
Fuel Pool System Piping	ANSI B31.1

3.3.1.3 Design Evaluation

Several considerations and analyses determine the safety of fuel pool handling and storage operations. These include:

Thermal Analysis
Radiological Analysis
Fuel Assembly Structural and Material Considerations
Fuel Pool SSC Structural and Material Considerations
Criticality Analysis

The passive design features of the pool prevent significant loss of pool water and assure that sufficient time is provided for operators to identify any incident condition and take the necessary action to restore cooling, and, if required, provide makeup to the spent fuel pool. The design and administrative controls assure that the fuel is safely stored and the public and workers are adequately protected. Several incident scenarios are described below. These include:

No pumps running (loss of forced cooling)
Loss of heat sink
Siphoning
Fuel Handling Incident
Cask Drop Incident

To determine the significance of these events, each of these incidents are reviewed in terms of the design criteria and design basis requirements. An evaluation is performed for each incident to determine the time available to the operator to discover the incident and take remedial actions. The consequences of each event are then evaluated in terms of pool water level and the radiological consequences. Structures, systems, and components, credited in preventing or mitigating the event are then classified according to its safety function. Similarly, activities credited to mitigate the condition are then defined. SSCs which are safety class are contained in section 3.1.
A discussion of the analyses and considerations pertaining to the defueled condition is contained below followed by a description of the incidents and the consequences.

3.3.1.3.1 Analyses and Considerations

<u>Thermal Analysis</u>

The heat load as a function of time is provided in Table 5.5.2. The reduced heat load on 12/29/97 is calculated to be 5,420,000 BTU/hr. which approximately represents a 75% reduction in heat load from the previously analyzed full core off-load scenario when the plant was operating. The revised residual decay energy release rate was calculated using the assumptions and formulations of Branch Technical Position (BTP) ASB 9-2 "Residual Decay Energy for Light Water Reactors for Long-Term Cooling" and is consistent with the guidance contained in the Standard Review Plan (NUREG-0800), section 9.1.3. It also considers the long term uncertainty fraction for cooling times greater than 10^7 seconds and are very conservative.

Time to Boil

In this analysis, the "time to boil" evaluation was performed for three levels (elevations) of water, each at five initial pool temperature conditions. The approximate elevations shown in the following table:

Elevation (ft.)	Height from pool floor (ft.)	Height above active fuel (ft.)	Description
31.5'	23.5'	10'	This elevation corresponds to lowest point on the cooling water return piping into the SFP.
40'	32.5'	19'	This elevation corresponds to the level at which the cooling water suction piping penetrates the liner.
43'	35.5'	22'	This elevation corresponds to the water level one foot below the normal pool water (i.e., the low level instrument setting)

The time to boil is also a function of the initial temperature of the pool water. Accordingly, for each elevation, time to boil is calculated at initial pool water temperatures of 80°F, 100°F, 120°F, 140°F, and 154°F. Time to boil was calculated for each of the three water levels for differing heat loads as a function of time after shutdown and is shown in Table 5.5.2.

A spent fuel pool heat load test, hereinafter referred to as the "passive cooling" test, provides more realistic site specific results. The passive cooling test demonstrated that the heat loss due to the effects of conduction, convection, and evaporation results in a significantly longer "time-to-boil" and lower "boil-off" rate. It is therefore more representative of pool cooling than the BTP ASB 9-2 calculation results. As a rule-of-thumb, the representative "passive cooling" results are approximately 71% of the BTP ASB 9-2 results. An example of the differences are illustrated in the table below.

Initial water level = elev. 43' **Final water level = elev. 31'** **Initial Pool Temp. = 80°F**	**BTP ASB 9-2 Results** **12/29/97** **5.42E+6 BTU/hr**	**Passive Cooling Results** **12/29/97** **3.85E+6 BTU/hr**
Time to Boil	67.36 hrs. ; 2.8 days	93.5 hrs. ; 3.9 days
Boil off Rate	11.64 gpm ;1.47 ft./day	8.3 gpm ; 1.05 ft/day
Days to reach ≥10 ft. above active fuel	8.16 days	11.42 days

Radiological Analysis

An evaluation was performed to determine the upper limit radiation fields in the fuel building and at surrounding onsite and offsite locations resulting from gamma radiation from the spent fuel in the fuel pool. This evaluation was performed as a function of both decay and water level. The results are contained in Table 5.5.3.

The analyses are conservative for the following major reasons:

1. The SFP cross-sectional area was assumed to be loaded with discharged assemblies, each assembly having the same operating history as the worst case assembly currently in the pool.
2. No credit was taken for the shielding provided by the assembly and the fuel rack structural material above active fuel.
3. The sky shine radiation was based on the assumption that the gamma rays escaping from the fuel pool had a spectrum identical to the uncollided spectrum originating within the fuel, thus allowing the radiation to propagate to farther distances.

The basic findings in this analysis are as follows:

1. The dose rates on elevation 46', by the edge of the pool, are approximately a factor of 2 lower than those at the over-pool platform.
2. A water depth of about 11 feet over the fuel is needed for a dose rate less than 1 mrem/hr on the platform immediately above the pool.
3. For 23 feet of water above active fuel, the dose rate on the over-pool platform is less than .0003 μrad/hr.
4. With 4 feet of water above the active fuel and one year of decay, the sky shine radiation level at the worst case receptor analyzed (20 m from the SFP center) is approximately 15 mrem/hr: this sky shine radiation level drops to approximately 5 μrem/hr at 610 meters (exclusion area boundary).

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The siphoning incident assumes that siphoning occurs from the cooling water return piping inlet and represents the worst case scenario. This incident results in a worst case pool water level approximately 10 feet above the active fuel[1]. Radiological doses are illustrated in the table below.

Water Level at 10 Feet Above the Active Fuel			
Decay time since 12/6/96	Sky shine at 610 meters (EAB)	Radiation field on platform directly above pool	Radiation field on 46' elevation beside pool
1 year	.0003 μrem/hr	< 4 mrem/hr	< 2 mrem/hr
2 years	data not available	< 2 mrem/hr	< 1 mrem/hr
3 years	.00007μrem/hr	< .8 mrem/hr	< .4 mrem/hr

Radiological consequences are discussed for each incident resulting in a loss of water level. Consistent with the operating plant requirements, dose rates around the fuel pool can normally be expected to be less than 2.5 mrem/hour during normal spent fuel pool storage conditions. During fuel handling operations, the maximum dose rate to the operators is administratively maintained below 50 mrem/hour. This limit is preserved when analyzing the incidents below. Radiological effects are discussed in terms of direct radiation field, sky shine, pool surface evaporation, and in the case of the fuel handling incident, the decontamination factor for Iodine scrubbing. Direct radiation and sky shine are a function of shielding (e.g., water level). The radiation dose rates affected through pool surface evaporation are applicable when the water is significantly heated or boiling. It is virtually a constant which is typically in the range of 1.28 mrem/hour (whole body) and 1.8 mrem/hour (organ-lungs). The decontamination factor is a function of water level above the fuel pin. Iodine is not a factor unless there is a breach of the fuel pin (e.g., fuel handling incident).
Even then, the short half life of iodine coupled with the fuel decay time minimizes the source term dose affects contributed by radio-iodines.

[1]As noted in Section 3, a branch syphon break has been installed which increases the margin to approximately 19'.

Fuel Assembly Structural and Material Considerations

The fuel assemblies are designed to maintain their structural integrity under steady-state and transient operating conditions, as well as under normal handling, shipping, and refueling loads. The design takes into account differential thermal expansion of fuel rods, thermal bowing of fuel rods and CEA guide tubes, irradiation effects, and wear of all components. Mechanical tolerances and clearances were established on the basis of the functional requirements of the components. All components, including welds, are highly resistant to corrosion in the reactor and fuel storage environment. Each core was made up of 38,192 Zircaloy 4 or ZIRLO clad rods in 217 assemblies. The rods contain slightly enriched uranium in the form of sintered UO_2 pellets, burnable absorber, or water-filled rods. The principle design structural criteria for the fuel rods is that the predicted permanent strain of the cladding is less than 1.0% during the fuel lifetime.

Fuel Pool SSC Structural and Material Considerations

The fuel pool wall and floor slab is constructed of 6' 0" thick reinforced concrete with a wall and floor lining of 1/4-inch thick stainless steel to ensure against loss of water. The new and spent fuel pool structures including fuel racks are designed to withstand the anticipated earthquake loadings as seismic Class I structures. The spent fuel pool liner is designed QAR.

A structural evaluation was performed to determine the dead load and hydrostatic force affect on the SFP concrete wall, floor slab, and stainless steel liner when the pool is subjected to an elevated pool water temperature of 212°F. Even when revising the thermal load factor, ACI allowables were met for a differential temperature of 186°F. Further qualitative discussions showed that even if the ACI allowables were exceeded (i.e., long term loss of all SFP cooling during the coldest possible winter temperatures), the water retaining capacity of the pool would be maintained since the pool side concrete remains in compression and the liner has sufficient ductility to maintain inventory integrity. It was concluded that in the unlikely event that a loss of forced flow incident occurred, the concrete walls and base slab have adequate capacity to resist the forces and moments generated by the self weight, water pressure, and thermal effects due to the boil-off of the pool. The load factors assumed for the evaluation are for a service load condition and therefore does not restrict the elevated temperature condition to a, one-time only, limited duration incident.

The spent fuel racks are designed to Seismic Class I requirements, constructed of ASME Section II, SA240, Type 304, or ASTM A240 Type 304L stainless steel. The steel alloy is reasonably corrosion resistant to the oxidizing effects of most electrolytes at low concentrations. The steel is susceptible to corrosion in acidic solutions (pH less than 7.0) containing chloride or fluoride anions and can lead to pitting of the material. Control of the water impurities are provided by the SFP purification demineralizer and filters and a chemical control program to assure that impurities are minimal and will not affect the structural integrity of the racks, liner, piping, vessels, exchangers and pumps.

The neutron absorber poison material is "Boral"; a cement composite material made of boron carbide particles in a Type 1100 aluminum matrix. This composite is highly durable and heat resistant. Boral panels consist of two outer sheets of type 1100 aluminum that clad a sintered plate of boron carbide in a type 1100 aluminum matrix. The type 1100 aluminum material imparts sufficient corrosion resistance by forming an aluminum oxide layer on its surface when exposed to oxidizing agents. This oxide layer is stable in environments with a pH range from 4.5 to 8.5 (which is the operating range of the spent fuel pool). The Boral panels are installed snug between the outside wall of the storage cells and the 304 stainless steel sheath that is welded to the wall. Vent holes at the corners of the stainless steel sheath create a sufficient vent path for any potential hydrogen produced by a water-aluminum reaction.

The neutron absorber capability of Boral is assured by fabrication design and material selection. The quantity and configuration of the panels is verified by a visual Boral Surveillance Program

All piping used in the fuel pool system is type 304 stainless steel, as is the fuel pool purification filter and demineralizer vessels, liner plates, and the tube side of the fuel pool heat exchanger. The pumps (spent fuel cooling pumps and purification pumps) are constructed of type 316 stainless steel. Each type of stainless steel is reasonably corrosion resistant.

Criticality

Criticality of fuel assemblies is precluded by adequate design of fuel transfer, shipping, and storage facilities, and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality is that, considering the possible variations, there is 95% probability at a 95% confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 for flooded conditions as recommended in NUREG-0800. The following conditions were assumed to meet this design basis:

1. The fuel assembly contains the highest enrichment authorized without any control rods or any noncontained burnable poison and is at its most reactive point in life.
2. For flooded conditions, the moderator is pure water at the temperature within the design limits which yields the largest reactivity.
3. The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design.
4. Mechanical uncertainties are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

5. Credit is taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption.

6. Where borated water is present, credit for the dissolved boron is not taken except under postulated accident condition, where the double contingency principle of NUREG-0800 is applied which requires at least two unlikely, independent, and concurrent events to produce a criticality accident.

For fuel storage application, water is usually present. However, the design methodology also prevents accidental criticality when fuel assemblies are stored in the dry condition. For this case, possible source of moderation such as those that could arise during fire fighting operations are included in the analysis. The design basis K_{eff} is 0.98, as recommended in NUREG-0800 for optimum moderation.

The calculations which assure the criticality safety of fuel assemblies include uncertainties. The uncertainties are applied to assure that there is a 95% probability with a 95% confidence level that the specified design basis K_{eff} will not be exceeded. The total uncertainty added to the criticality calculations include the following components:

1. Method Uncertainty: based on critical experiment comparisons to establish method bias and variability.
2. Statistical Uncertainty: based on the statistics of the particular Monte Carlo calculations.
3. Fabrication Uncertainly: based on manufacturing and mechanical tolerances, such as thicknesses and spacings. Either statistically- based uncertainties or "worst case" assumptions are used.

The criticality design criteria are met when the calculated effective multiplication factor plus the total uncertainty is less than the specified design basis effective multiplication factor.

In the permanently defueled condition, the reactivity effects of a misplaced assembly represents the bounding case for determining the boron concentration required to maintain a 5% $\Delta k/k$ safety margin to criticality in the spent fuel pool. The analysis considered various misposition configurations including a fuel assembly laying horizontally on the top of the spent fuel racks, a fuel assembly placed adjacent to or on the outside corner of the fuel racks and a fuel assembly placed in the wrong spent fuel rack. A boron concentration of 670 ppm was assumed in the analysis for the postulated misplaced fuel assembly. This was determined to be required to maintain a 5% $\Delta k/k$ safety margin to criticality. This concentration is required whenever fuel assemblies are stored in the pool and a spent fuel pool assembly placement verification has not been performed since the last movement of fuel assemblies in the spent fuel pool. An administrative limit is further detailed in this safety analysis report.

Incident Evaluation

Pool water level is normally constant except for relatively minor evaporative losses. For the purposes of this evaluation, the pool water level is important from a standpoint of determining time available to the operator. The determination that adequate pool water level remains many hours or days after the incident is key to determining the severity of the event and the safety function performed by SSCs. Pool water level is important to meet the design basis requirement to maintain the fuel covered with water and for providing an end-of-incident water level which provides adequate shielding to maintain exposures below 10CFR 20 requirements.

Consistent with the SRP criteria, the spent fuel pool and cooling, makeup and purification systems design assures that, in the event of a failure in suction or return piping, the pool level is not inadvertently drained to a level approximately 10 feet above the top of the active fuel.
Significant elevations in the spent fuel pool are noted below:

Description	**Elevation** (approx.)	
Top of the spent fuel pool	46' 0"	
High Water level (alarm)	45'	
Normal water level	44'	
Low water level (alarm)	43'	
Siphon breaker (suction side)	40' 11"	
(Return side)	40'	
Bottom of gate to the fuel transfer canal	12' 0"	
Top of active fuel (approx.)		
Fuel pellets	21'	(no higher than)
Fuel pin	21'6"	(no higher than)
Bottom of the spent fuel pool	7' 6"	

The lowest elevation where piping penetrates the liner is on the cooling water return (inlet) piping. The piping penetrates the liner at the 33.5 ft elevation and turns downward terminating at the 31.5 ft. elevation. A hypothetical break in the piping is postulated to occur on the discharge side of the pump or in the non-safety skimmer piping which causes a siphoning action and subsequent draining of the pool to the 31.5 ft. elevation. This condition is analyzed later in this section. (A new siphon anti-siphon device has subsequently been installed at El. 40'-0" on the cooling water return piping)

An anti-siphon device is installed on the cooling suction outlet and is designed safety class 3. This penetrates the liner at the 40 ft elevation. The anti-siphon device extends up to just below the 41 ft. elevation. Therefore, the maximum level to which siphoning can hypothetically occur from this penetration is to the 40 ft. elevation. At this level, approximately 19 ft. of water remains above the active fuel. This anti-siphon device prevents inadvertent draining of the pool and, for this reason, is designed to seismic Class I requirements.

Several administrative controls are in place to assure the continued safe storage operations. Alarm response procedures provide assurance that off-normal conditions are quickly identified and remedied. For instance, alarms are provided to alert operators to off-normal pool conditions, including high temperature (110°F) and high/low water level (45' and 43' elevations, respectively) and radiation level. Once alerted, operations personnel intervene before significant changes in water level or temperature can occur. In the event that the water level is increased one foot above, or falls one foot below, the normal operating level, an annunciator would alert operations personnel. Similarly, a high temperature alarm in the control room would alert operations personnel to the high temperature condition. Plant operating procedures require the operator to investigate the alarm condition and initiate appropriate remedial actions. This instrumentation, as well as the area radiation alarm, is useful for alerting operators of off-normal and incident conditions; however, the alarms are not credited in the spent fuel storage incidents. In the scenarios evaluated, credit was only taken for daily direct visual monitoring by Operations personnel of the SFP temperature and level.]

Loss of Spent Fuel Pool Cooling

The following assumptions are used in this scenario:

1. Initial pool water temperature is 212°F,
2. Boil-off rate as of 12/29/97 equals 11.64 gpm (1.47 feet per day) and
3. Initial normal water level in the fuel pool (43' elevation),
4. Incident water level decreases to 10' above the active fuel (31' elevation). The 31' elevation (vs. the 31.5' elevation) is used because it corresponds to 10' above the fuel pellets, referred to hereinafter as the "active fuel" as pellet height is used to determine the radiological field and sky shine doses to receptors.

To analyze the time available to operators in the event of loss of spent fuel pool cooling, the principal functions considered include the time it takes to boil and the boil-off rate. For ease in analyzing this event, it conservatively assumes that the temperature starts at 212°F. According to the BTP ASB 9-2 results, it would take a minimum of 8 days to reach the 31' elevation. According to passive cooling test results, it would take a minimum of 11 days to reach the 31' elevation.

At ten feet above the active fuel, radiation field dose rates are expected to be no greater than 4 mrem/hour (whole body) plus evaporative dose rates of no more than 2 mrem/hour (whole body). This results in radiological dose rate of no more than 6 mrem/hour (whole body). An additional organ dose rate of 2 mrem/hour (lungs) is also expected. Historically, 50 mrem/hr is the maximum dose rate permissible for fuel transfer operations. The doses received as a result of this incident are well below this limit. Sky shine dose rates at this level are approximately 40 μrem/hr of sky shine to a receptor 20 meters from the pool center, and .0003 μrem/hr at 610 meters (exclusion area boundary).

The above numbers are very conservative from a radiological analysis standpoint as can be seen by the radiological analysis assumptions above. It is also conservative from the decay heat load standpoint as follows. Empirical data was gathered during a SFP heat up test conducted in October, 1997. In that test, the initial fuel pool temperature was 81°F and the pool level was at it's normal elevation (44 ft. elevation). The SFP cooling pumps were then secured. It took approximately 70 hours for the pool water to reach 140°F (Note that there was virtually no loss of water level during that heat-up period. In addition, note that these results are conservative with respect to the passive cooling analytical results contained in section 3.3.1.3.1 which indicate 93 hours to boil). Assuming an initial SFP temperature of 140°F, BTP ASB 9-2 calculation results show that the time-to-boil would take approximately 34 additional hours. Although not credited as part of this safety analysis, when combining the empirical and analysis data, a more reasonable estimate of the time to boil is approximately 100 hours.

<u>Siphoning Incident</u>

There are four penetrations which project into to the fuel pool. These are: 1) the fuel transfer tube, 2) a piping penetration for cooling system suction (approx. 40' elevation), and 3) a pair of piping penetrations for cooling system discharge (approx. 31.5' elevation)[1]. Failure of the transfer tube is not deemed credible due to the robust design (seismic Class I and safety class 3), it's passive function, and the existence of at least two passive mechanical devices preventing draindown. The piping penetration at the cooling system suction is designed with an anti-siphon device which is designed safety class 3 and seismic class 1. The lowest elevation to which siphoning can occur is 31.5 ft. elevation and is the most limiting siphoning incident. The associated piping systems connected to this penetration contain NNS piping which is not seismically designed.

The evaluation of this incident assumes that the siphoning event starts immediately following the daily operations area walk-through wherein pool temperature and level is checked. The pool drains down to elevation 31.5 feet due to siphoning caused by a non-mechanistic pipe break[2]. At this level, no less than 10.5 feet of water above the active fuel remains. Assuming an initial temperature of 80°F and a heat load of 5.42E+6 BTU/hr, the time to boil is 39.5 hours. No credit is assumed for the low level alarm or for operations personnel to identify the existence of the displaced fluid. Operations personnel would detect the condition on their daily round no later than 30 hours after the incident and 9.5 hours prior to boiling. With 10.5 feet of pool water above the active fuel, radiological exposures to workers in the fuel building is no greater than approximately 2 mrem/hour (whole body) for a person on the platform directly over the pool, plus evaporative whole body dose rate of no more than 2 mrem/hour.

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[1] A pre-existing piping penetration (at approximate El. 40') was utilized to create a return line syphon break.

[2]The syphon break noted above makes this evaluation conservative

At the time of identification, corrective actions are taken to investigate the cause of the incident and start makeup to the pool. Under any credible situation, several makeup sources could be initiated within a few hours of identification. Not withstanding, the consequences of this incident is further analyzed assuming that corrective action is not taken immediately following identification. The assumptions are:

a. No action is taken for another 32 hours once identified (30 hours after incident),
b. No credit is taken for the remaining 9.5 hours to boiling (since the time of identification),
c. A 25% allowance for the 24 hour surveillance is assumed (30 hours), and
d. The water is at a boiling temperature with a boil-off rate of 11.64 gpm or 1.47 ft. per day.
e. BTP ASB9-2 decay heat loads are used

With the above assumptions, after 62 hours (30 hrs. to identify the condition + 32 hrs. to effect remedial action), the water level would be no lower than 8.5 feet above the active fuel. With 8.5 feet of pool water above the active fuel, a straight line extrapolation of radiation field dose rates indicates that the operators on the platform directly over the pool would be subject to a radiation field dose rate of no more than 35 mrem/hour whole body plus evaporative whole body dose rates of no more than 2 mrem/hour, or 37 mrem total (whole body). An additional 2 mrem/hour organ (lung) dose rate would be received. By the edge of the pool, whole body dose rate is on the order of 20 mrem/hour. These totals still remain within the 50 mrem/hour maximum radiation exposure limits indicated above for fuel transfer operations. Offsite dose rates are less than 10 μrad/hr at 20 meters, and less than .003 μrad/hr at the exclusion area boundary (610 meters).

Several conservative assumptions are used in this incident. One assumption involves the 25% allowable for the routine daily round. Using the BTP ASB 9-2 heat loads, at an initial pool water temperature of 80°F, over 62 hours are available from the initial event to reach a pool water level of 8.5 feet above the active fuel. This estimate is still accurate assuming an initial pool temperature of 100°F. When assuming 24 hours to discovery, 62 hours is accurate up to and including an initial pool temperature of 130°F or less. Finally, using the passive cooling results, significantly more time is available. Using a heat load of 3.85E+6 BTU/hr, at an initial pool temperature of 80°F, the time to boil is over 93 hours. The boil-off rate is 1.05 ft. per day which allows for additional time to reach the 8.5 ft level above the active fuel. Other conservative assumptions in the radiological analysis assume that all assemblies have the same operating history as the worst case assembly. Therefore, actual doses would be lower and a reduction in water level below 8.5 ft could occur and still remain below the 50 mrem/hour whole body dose to personnel at the edge of the pool.

Assuming the conservative BTP decay heat loads and a boil-off rate of 1.47 ft./day, it would take approximately 40 hours for the level of the pool water to decrease from 8.5 feet to 6 feet above the active fuel. This corresponds to a whole body dose rate of approximately 750 mrem/hour to operators around the perimeter of the pool, and a total of 102 hours from the time of incident initiation. 6 feet of shielding above the active fuel corresponds to a dose rate of approximately .15 μrem/hour to a receptor at the exclusion area boundary. In all cases, reasonable time is available for operators to take corrective actions to restore make-up to the pool. Additionally, strict radiological controls are effected to minimize exposures to personnel ALARA.

Fuel Handling Incident

For the fuel handling incident, no credit is taken for systems or components to mitigate the incident such as fuel building ventilation systems or control room ventilation systems. Additionally, no credit is taken for scrubbing of released gases (DF=1) and the dropped assembly is based upon the release of the fuel rod gap inventory of the worst case (highest burnup, enrichment and longest operating history) composite with one year decay. The accident assumed an instantaneous puff release at ground level meteorological conditions to determine offsite exclusion area boundary and control room doses. As shown in Section 5, doses are acceptable assuming no filtration. The occupational, control room and offsite radiological doses from the fuel handling incident bounds all of the fuel pool storage incidents described above.

Cask Drop Incident

Movement of the cask has been carefully examined and in no case does the cask pass over systems or equipment important to safety. An analysis has revealed that a 100-ton, 6-foot diameter cask dropped 42 feet straight down into the fuel storage pool would puncture the steel liner and penetrate 1.5 feet into the 6-foot concrete floor. Calculations were later performed expanding the scope of the original analysis to include 125 ton 9.5 foot diameter casks. Leakage of 2 to 5 gpm may be expected due to the permeability of the crushed concrete, backfill, and bedrock. In addition, a maximum of 2.5 gpm leakage may occur through the liner leakage detection system for each leakage zone breached as the result of the cask drop. Available makeup capacity is significantly higher than the draindown rate of the spent fuel pool. Equipment location and drainage capability is such that no damage to critical equipment from this leakage would occur.

Although analyses are in place evaluating the consequences to the SFP from a cask drop as acceptable; Maine Yankee has also chosen to upgrade the fuel building yard crane, CR-3, to meet the single-failure-proof guidelines of NUREG-0612 and NUREG-0554 for cask handling over the SFP. The CR-3 upgrade provides another level of defense-in-depth as the single-failure-proof status reduces the probability of a load drop to an extremely small value.

Loss of Heat Sink

Loss of DHR cooling to the SFP heat exchanger (E-25) will result in a heat-up of the pool. DHR normally flows through the shell side of the fuel pool heat exchanger and cools the tube side pool water. If DHR is not available, temporary hoses may be connected to flanged connections provided on the shell side of the heat exchanger to allow the alignment of alternate cooling water supplies. Loss of DHR cooling is bounded by the "Loss of Spent Fuel Pool Cooling" incidents described above. An abnormal operating procedure describes the steps to initiate alternate cooling from a fire system supply hose to the E-25 inlet connection and a fire hose from the E-25 outlet connection to the fire pond. The fire system has ample capacity to provide alternate cooling to the spent fuel pool heat exchanger. The fire system also has a diesel driven fire pump.

Evaluation and Conclusion

The primary function of the spent fuel pool cooling, makeup and purification system is to ensure that the fuel remains covered and that suitable shielding exists to maintain radiological exposures below 10CFR 20 limits. This is primarily a passive design function and requires no active system intervention by the system components to prevent or mitigate the consequences of a design basis accident. In the current plant condition, the radiological source terms have decayed to a level where the postulated events would not result in offsite dose limits exceeding a small fraction of the 10CFR100.11 limits. The reduced heat load, and the assurance of pool integrity provide the margin required to credit only the passive design features of the pool. Operators are provided ample time to discover the incident conditions, effect remedial actions to restore pool cooling or provide make-up prior to uncovering the fuel, and maintain adequate shielding above the active fuel.

Although boiling is permitted as a service condition, long term boiling as a service condition is not practical due to the continuous makeup capacity required and the demineralizer temperature limits. Accordingly, normal pool water temperatures are maintained below 110°F. The pool water is cooled using the spent fuel pool cooling pumps. Additionally, the purification pump may be used to cool the pool water if conditions permit (i.e., depending on the decay heat load, the pool temperature, DHR flow rate and the DHR inlet temperature).

The spent fuel pool purification system is provided to maintain water chemistry within specifications and is not required to prevent or mitigate the consequences of an accident. For out-of-specification chemistry conditions, corrective actions are taken based upon the results of sample analyses or other indications. Refer to section 3.3.1.5 for the parameter values monitored and their frequency.

Makeup water is provided to the spent fuel pool to compensate for loss of pool water level primarily due to evaporation losses. Sufficient available make-up water capacity exists through diverse sources to allow operators to take manual actions to compensate for liner leakage or natural and incident evaporation rates due to boiling. Makeup capacity includes: the Primary Water Storage Tank (and associated pump P-SFP2), town of Wiscasset water supply system, and the water storage pond (and fire pump).

The active components of the spent fuel cooling, makeup and purification system are not relied upon to assure: 1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of an accident resulting in potential offsite exposures comparable to the applicable guideline exposures set forth in 10CFR 50.34(a)(1) or 10CFR 100.11. Due to the time available to operators to provide makeup to the pool, the active components of the spent fuel pool cooling, make-up and purification system are not safety related nor are they credited to function during or following a design basis earthquake. The time available following each of the incidents, including the worst case siphoning event, conservatively shows at least 62 hours since the initiation of the event to establish make-up without exceeding the 50 mrem/hour (whole body) limit normally established for fuel handling operations. The analysis also conservatively demonstrated that, with 6 feet of water over the active fuel, over 100 hours are available to an operator to provide make-up. With 6 feet of water, operators around the pool perimeter would be subject to a whole body dose rate of no more than 750 mrem/hour. In all cases, a suitable work environment is available, and adequate time is available to provide offsite or onsite make-up capability without relying on safety related equipment.

Fuel handling equipment is discussed in section 3.2.2. The results of the systems, functions, and actions credited relative to fuel handling are included below.

The below listed SSCs are designed to Seismic Class I requirements.

- A functional anti-siphon device on the suction side of the pool to prevent draindown,
- Fuel platform and hoist and yard crane
- Fuel pool concrete structure, liner, racks and transfer tube
- Flow limiters on the liner leakage detection system

The following system functions were credited in the evaluation.

- Capability of the pool liner and concrete structure to sustain long term boiling,
- Capability of the transfer tube to maintain its integrity, including associated valves and flanges.
- Fuel platform and hoist and yard crane design safety features to assure the safe handling of fuel and casks.

- At least one make-up water supply to the spent fuel pool is available within 24 hours after the identification of the worst case siphoning event. The flow rate shall be equal to or greater than the boil-off rate.
- The capability of the racks to maintain adequate spacing between the assemblies. This includes the placement of fuel in the racks in the proper region. Refer to Figures 3.3.5 and 3.3-6 for the proper placement of fuel in the racks as a function of burnup.

The following actions were credited during the events:
- Daily operator/handler area walk-throughs for SFP level and temperature.
- The ability to provide make-up to the fuel pool within 62 hours from the initiation of an loss of pool water siphoning incident.
- Assuring that boron concentration was adequate prior to the movement of fuel.
- Assuring that appropriate tests and inspections were conducted on fuel handling equipment prior to the movement of heavy loads and that such movements are administratively controlled to prevent movements in the vicinity of safety related equipment.
- Appropriate chemistry is maintained in the pool
- A Boral Surveillance Program is implemented
- Spent fuel cask handling will be performed with CR-3, which has been upgraded in accordance with the single-failure-proof guidelines of NUREG-0612/0554.]

3.3.1.4 Fuel Storage - System Operation

Normal Operating Specifications:

Fuel pool water level	Between 43 ft. and 45 ft. elevations
Fuel pool water level during fuel handling operations	≥ 42.2 ft elevation
Fuel pool temperature	Between 40^{0}F and 120°F
Fuel pool cooling pumps	≤1800 gpm
DHR Pump flow rate to E-25	≤1100 gpm
FHB Normal Radiation Levels (passage ways)	≤2.5 mrem/hr (whole body)
Radiation Levels during Fuel Handling operations	≤50 mrem/hr (whole body)
Water make-up	Demineralized Water

System Operation Description

The fuel pool cooling system is shown on Figure 3.3-9. This system removes the decay heat from spent fuel stored in the fuel pool by circulating the pool water through a heat exchanger. The fuel pool cooling pumps (P-17 A and 17B) take suction from the fuel pool at approximately 2 feet below the low level alarm (plant elevation 43 ft., or 35' 6" from the bottom of the pool). Each pump is capable of maintaining the pool water within normal operating parameters under all temperature conditions provided that the level is above the low level setpoint and electrical power is available. Suction flow by the spent fuel cooling pump is circulated through a heat exchanger (E-25), and returned to the fuel pool below normal water level at the 31 ft elevation. Flow from each pump may be independently throttled to obtain the desired flow. The fuel pool cooling pumps (P-17 A, B) are controlled locally via a local start/stop button at the pump on the 21' elevation of the Fuel Building.

The fuel pool cooling system has a purification loop consisting of a pump (P-85), a pre-filter (FL-2), and a post filter (FL-29) which may be operated independently of the fuel pool cooling system. The purification pump (P-85) located in the Fuel Building is controlled locally via a local start/stop switch at the local Motor Control Center on the 21' elevation of the Fuel Building.

The purification pump can take suction from the cooling pump suction line during periods when cooling is not required or from the discharge of the heat exchanger when the cooling system is operating. Flow from the discharge of the fuel pool purification pumps is directed through the fuel pool prefilter, and/or postfilter, while the return is through the fuel pool cooling return line. The purification pump can also be used for skimming operation or to circulate and cool pool water through the heat exchanger.

DHR normally flows through the shell side of the fuel pool heat exchanger and cools the tube side pool water. The design DHR flow rate is adequate to assure cooling of pool water. A discussion of the DHR water system is provided in section 3.3.3.

Spent fuel makeup capacity is provided through diverse sources. Makeup capacity includes: the Primary Water Storage Tank and the town of Wiscasset municipal water supply. Water is also available from the fire pond. None of the makeup sources are safety related. Two fire pumps are located in the fire pump house located near the water storage pond. A diesel engine drives one pump while the other is motor-driven. The building houses the diesel fuel tank, the batteries and control board required for the diesel operation. Therefore, under severe conditions, fuel pool inventory may be restored using the 2500 gpm fire pumps and the water storage pond.

During skimmer operations, the spent fuel pool purification pump takes suction from four surface skimmers to keep the pool surface free of foreign matter. The purification pump discharge is directed back to the fuel pool through the underwater return line during this operation.

The Fuel Building Area Radiation Monitor is discussed in section 4 including the response to radiation alarm conditions. The Fuel Building Ventilation System is described in section 3.3.5.

Operating procedures specify the methods and actions required to provide fuel pool makeup, cooling and purification under normal operating conditions. Operating procedures also specify the methods and actions required to restore cooling temperatures and water level in the spent fuel pool under abnormal operating conditions.

3.3.1.5 Monitoring and Instrumentation

Instrumentation

The fuel pool high or low level and the fuel pool temperature alarm conditions are annunciated in the control room.

Pressure transmitters and gauges are provided on the discharge of each pump in the spent fuel pool cooling system. This includes the differential pressure indicator located in the purification loop which may be valved-in to measure differential pressure of each filtering component. The fuel pool heat exchanger temperature indicating transmitter on the fuel pool cooling heat exchange outlet line is used to regulate the number of cooler coil fans operating in the DHR System.

Tell-tale connections are provided to aid operations personnel in monitoring for liner weld leakage. The tell-tales discharge into the spent resin pit sump. The manually operated sump pump discharges to the Holdup Tank (TK-109). The spent resin sump pump high level switch alarms on the PLC.

Upon receipt of a Spent Resin Pit high level alarm condition, an operator is dispatched to investigate the condition. If the alarm condition is due to liner leakage, actions are taken to restore level, as appropriate in accordance with operating procedures.

Chemistry

Spent fuel pool water chemistry is monitored to minimize the potential effects of corrosion which could affect the safe storage of irradiated fuel. Chemistry surveillance activities are performed within the specified interval below, with a maximum allowable extension not to exceed 25% of the specified interval. The spent fuel pool water is maintained with demineralized water and additives, as required. The water chemistry is maintained and monitored in accordance with the following values and frequency:

Parameter	Values	Nominal Frequency
pH at 25°C	4.5 to 8.5	Weekly
Chloride	less than 150 ppb	Weekly
Fluoride	150 ppb maximum	Weekly
Boron	≥ 670 ppm design limit	
	≥ 1200 ppm administrative limit	Monthly*
Gamma Isotopic	NA (for trending only)	Monthly
Suspended Solids	NA (for trending only)	Monthly

*7 Days when moving Fuel

Although dose analyses for the various spent fuel storage and handling incidents are conservative, they do not consider the effect of an increase in dose due to particulates in the spent fuel pool water or surface contaminants. In general, maintenance of water quality is necessary to prevent degradation of the spent fuel and other stored materials in the SFP. Accordingly, the chemistry program includes sampling analysis and corrective action recommendations to protect the stored materials and minimize radioactive contamination.

The underwater demineralizer and the slipstream purification pre and post filters, located in the Fuel Pool and the Fuel Building, respectively, are periodically monitored to assure that activity limits, and differential pressures, are within specifications.

<u>Ventilation</u>

The ventilation systems consists of heating and ventilation and is designed to provide a suitable environment for equipment and personnel. The ventilation system utilizes fans, filters, dampers, heating elements, and ductwork to accomplish the desired effects. Refer to section 3.3.2 for a description of requirements applicable during fuel movements. The ventilation system is NNS.

<u>Monitoring</u>

Fuel pool level and temperature is visually monitored and logged nominally every 24 hours.

Operators shall periodically monitor the temperature shown on the heat exchanger indicator to assure the temperature is within specification.

Operations shall periodically monitor for tell-tale liner leakage.

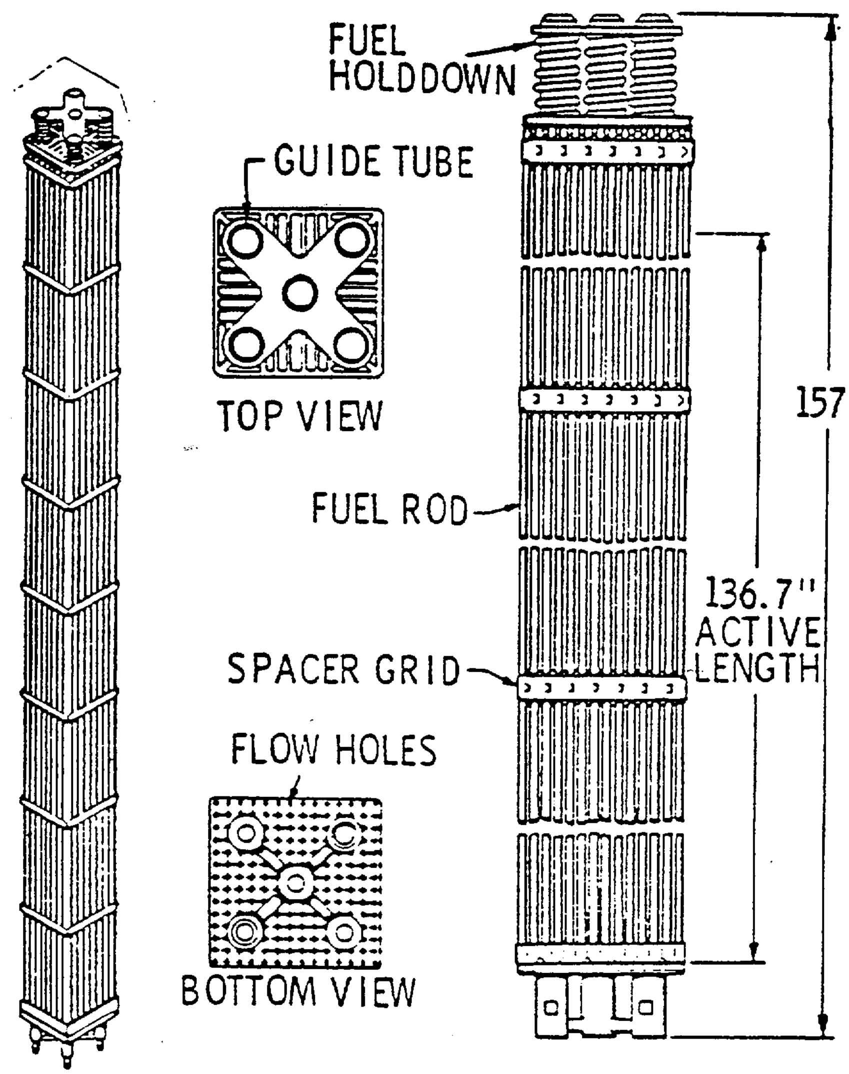
The suction siphon breakers is monitored periodically for signs of apparent or potential blockage.

Maintenance

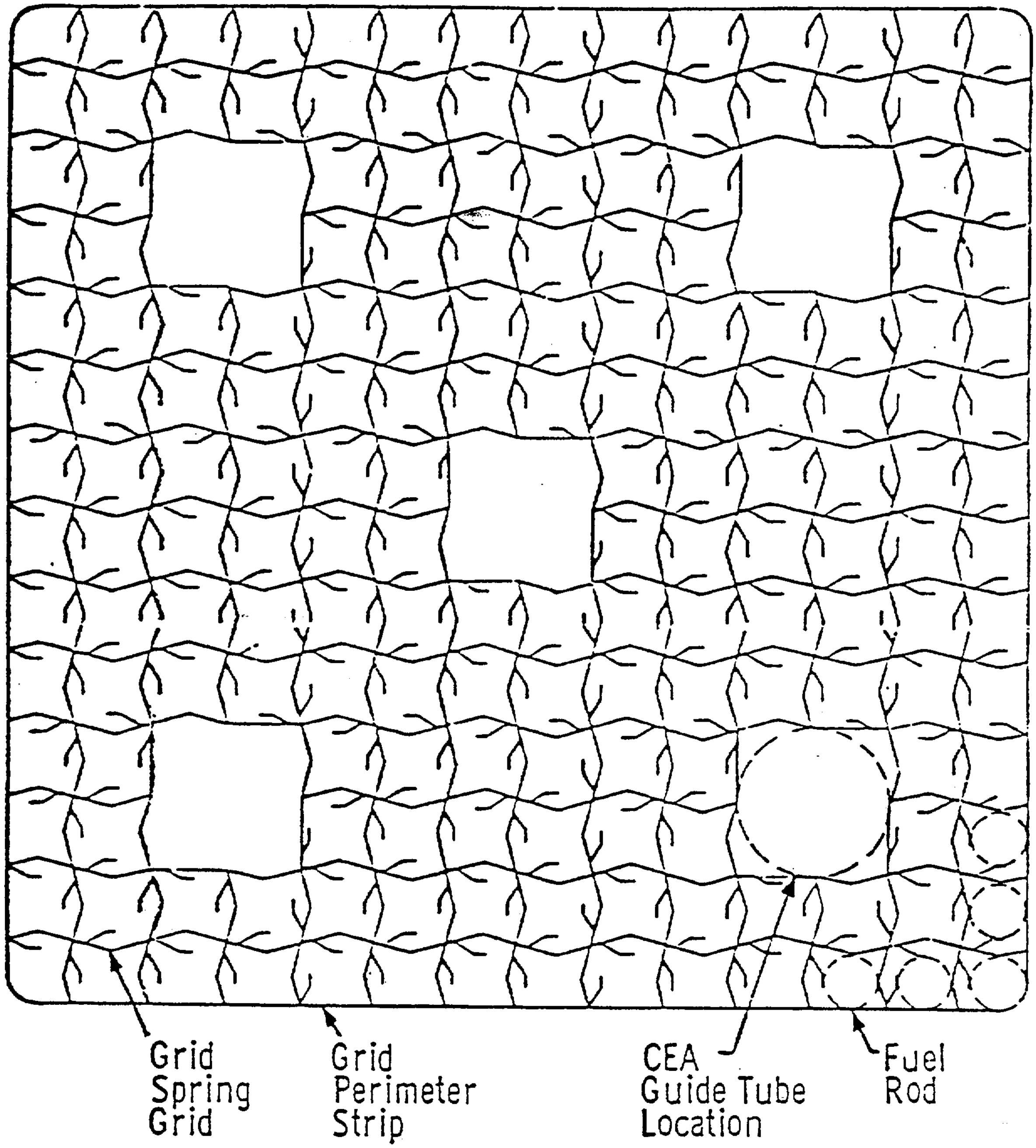
The components (flanges or valves) which assure the integrity of the fuel transfer tube are safety related and are periodically inspected to ensure that they can satisfactorily perform their intended safety function.

The suction siphon breaker at the 40' elevation is periodically inspected to assure there is no blockage.

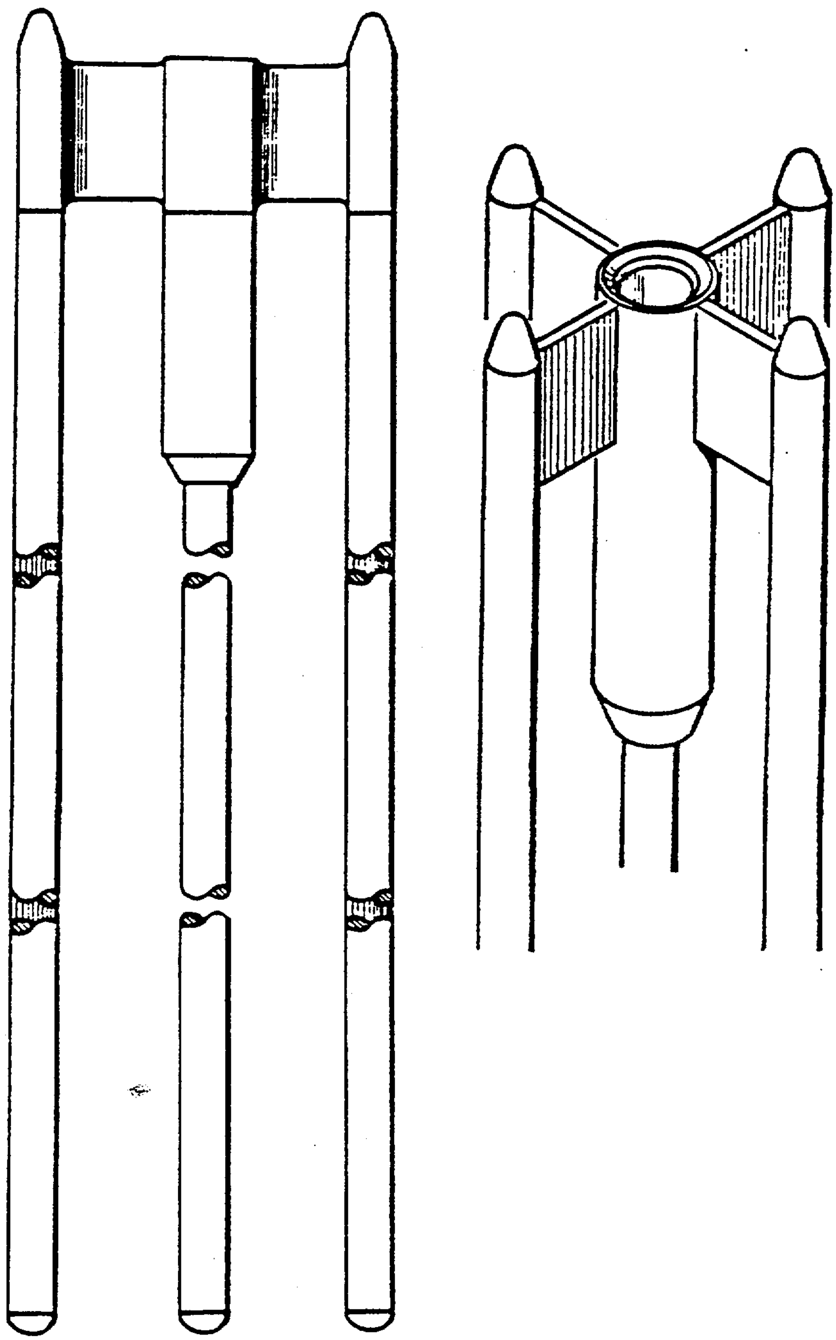
A Boral surveillance program shall be maintained.



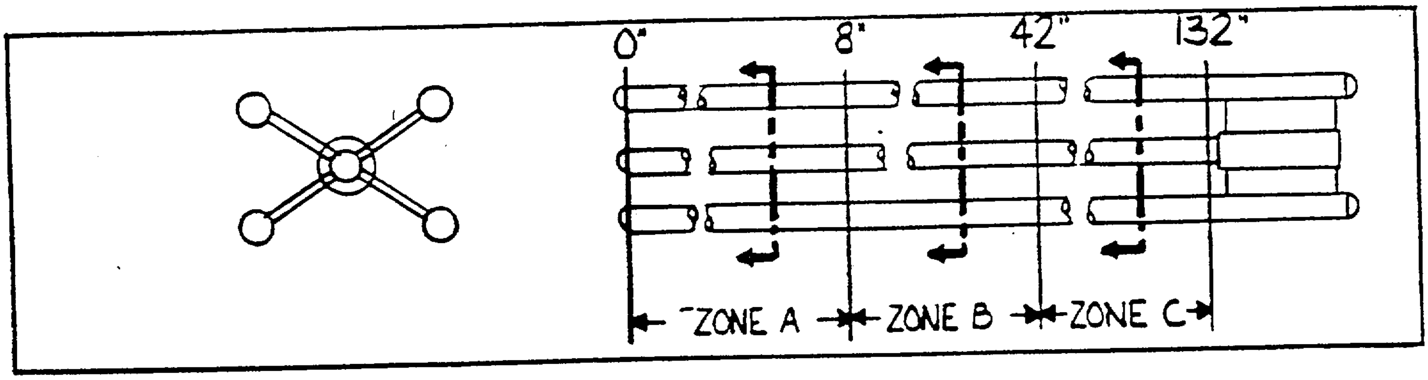
MAINE YANKEE ATOMIC POWER CO. Maine Yankee Atomic Power Station	TYPICAL Fuel Assembly	Figure 3.3-1



MAINE YANKEE ATOMIC POWER CO. Maine Yankee Atomic Power Station	TYPICAL Fuel Spacer Grid	Figure 3.3-2



MAINE YANKEE ATOMIC POWER CO. Maine Yankee Atomic Power Station	Control Element Assembly (CEA)	Figure 3.3-3



CEA TYPE	NO. OF FULL STRENGTH FINGERS	MATERIALS ZONE A	MATERIALS ZONE B	MATERIALS ZONE C
FULL-STRENGTH	5	Ag Ag B or Ag Ag Ag	B B B B B	B B B B B
PART-STRENGTH	1	S S B S S	S S B S S	S S B S S
PART-STRENGTH	2	Ag S S S Ag	B S S S B	B S S S B

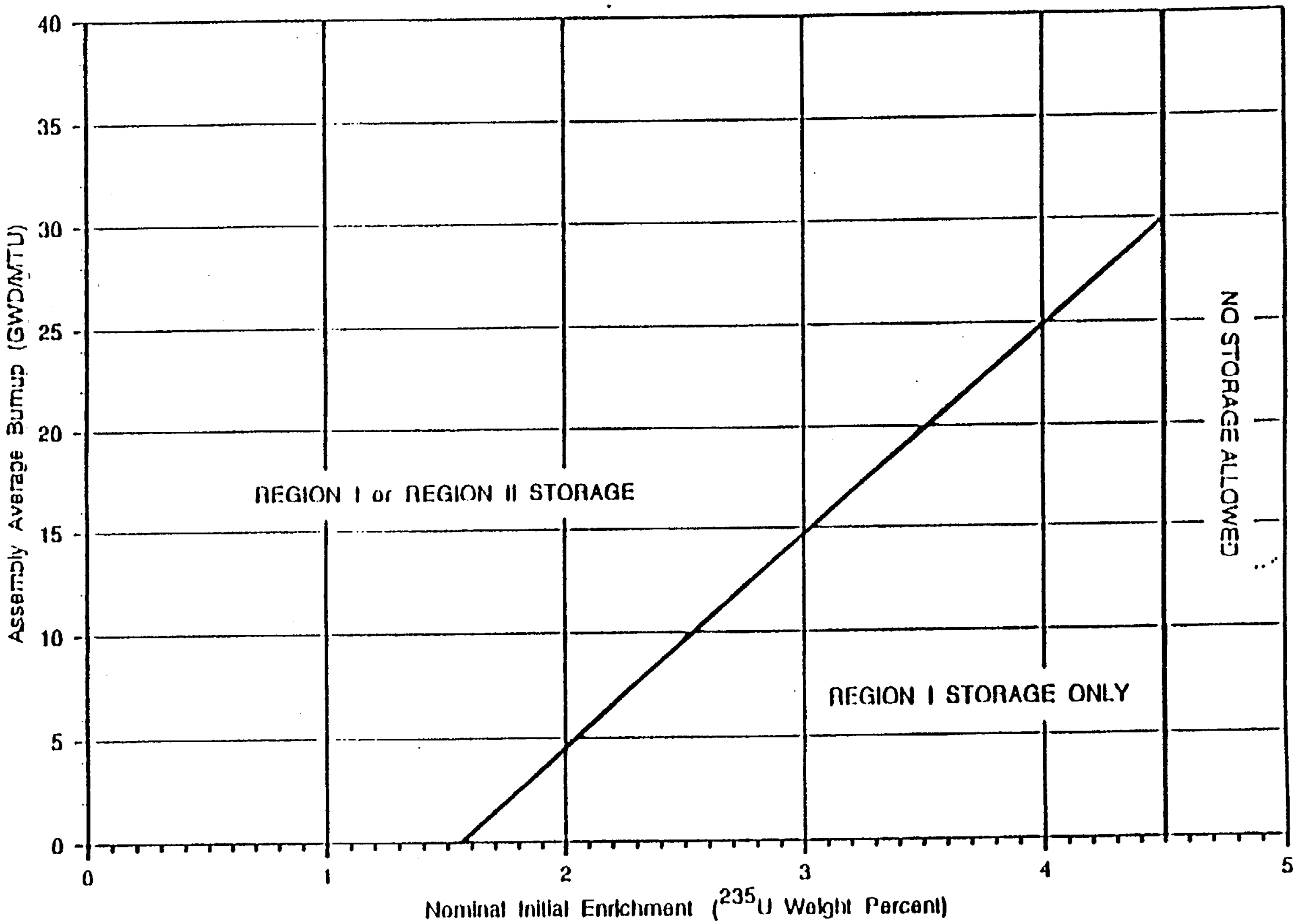
MATERIALS:

$B = B_4C$ S = STAINLESS STEEL Ag = Ag - In - Cd

$Al = Al_2O_3$

MAINE YANKEE ATOMIC POWER CO. Maine Yankee Atomic Power Station	CONTROL ELEMENT ASSEMBLIES (CEA)	Figure 3.3-4

MAINE YANKEE

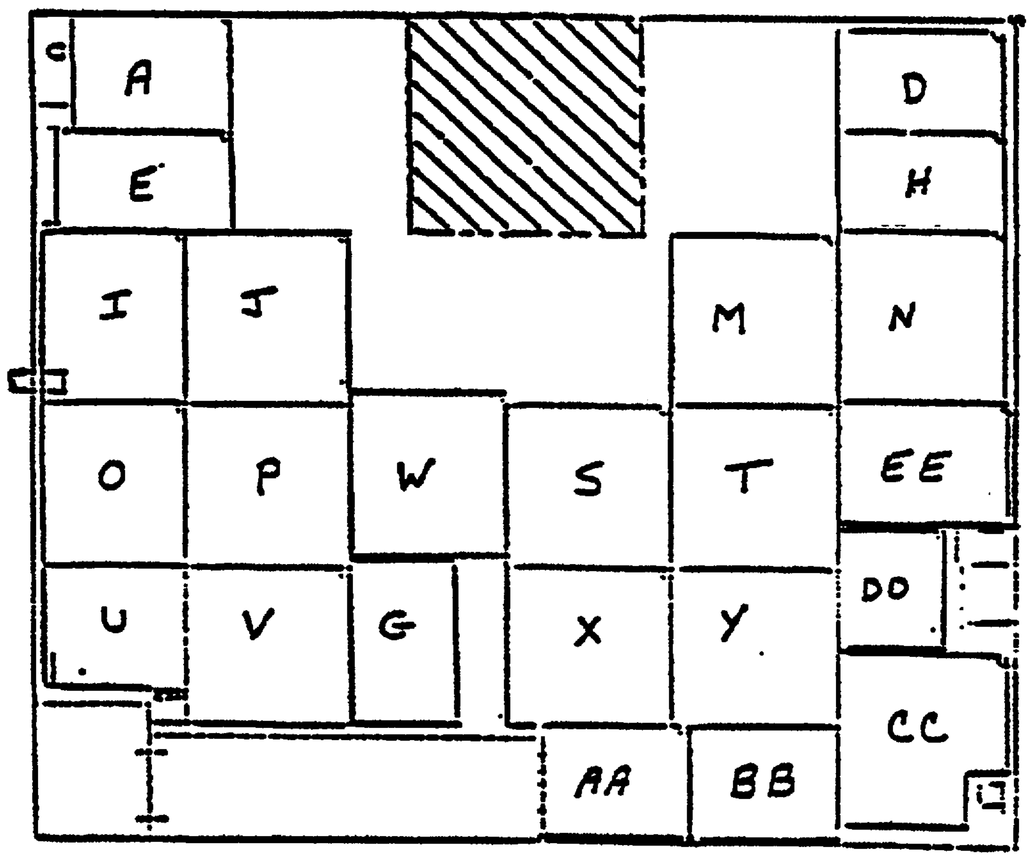


Spent Fuel Pool
Assembly Placement Limitations

Figure
3.3-5

FIGURE 3.3-6
Revision 18

HIGH DENSITY SPENT
FUEL RACK LAYOUT
FOR TWO REGION POOL



Region I racks have a double letter designation.
Region II racks have a single letter designation.

3.3.2 Fuel Handling System

3.3.2.1 Design Basis

Design Criteria

The fuel handling design criteria is as follows:

1. Criticality in new and spent fuel storage is prevented by physical design features or processes. A geometrically safe configuration is emphasized over procedural controls.
2. Appropriate shielding is provided to meet the requirements of 10CFR 20.
3. The fuel building is continuously monitored by area specific detectors. Audible and visual alarms are activated at the detector locations and the control room for radiation levels in excess of predetermined limits.
4. Spent fuel storage systems are:
   a. designed to prevent or mitigate accidents which could lead to the release of significant amounts of radioactivity affecting the public health and safety,
   b. designed, fabricated and erected to withstand, the additional forces that might be imposed by natural phenomena.

The design basis of the fuel handling system with regard to the spent fuel handling incident and the cask drop incident are contained in sections 3.3.1.3 and 5.0. Fuel building ventilation system requirements and radiation monitoring are contained below and in section 4.0. Heavy Loads program requirements are addressed in section 3.4.

Design Basis

The fuel handling design criteria is as follows:

a. Irradiated fuel shall not be consolidated until it has cooled for a period not less than 730 days after final discharge from the reactor. Prior to conducting irradiated fuel pin activities, a review is performed to assure that the licensing/design basis is adequately reflected in operating procedures. This includes a review to assure that occupational doses as a result of a pin rupture are maintained ALARA consistent with the requisite design basis ventilation flow.]
b. Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operations.
c. Fuel handling equipment is designed to preclude lifting of the assembly above a height which could cause undue radiation exposure to personnel (≥50 mrem/hr) on the platform directly over the pool.

d. The spent fuel racks are designed such that the dropping of a 2500 lb. (submerged weight) fuel assembly from a height of 18" above the top of the racks will not incur damage which could result in criticality. A subsequent evaluation has shown that the racks are able to withstand the dropping of a 2000 lb. (submerged weight) fuel assembly from a height of 22.5 inches above the top of the racks without incurring damage which could result in criticality or spent fuel cooling concerns.
e. Fuel handling equipment is designed to withstand the loadings that would occur during an Operational Basis Earthquake and will not fail so as to cause damage to any fuel elements should it occur during fuel transfer operations.
f. Placement of assemblies in racks shall be based upon the burnup rate in Figure 3.3-5.

3.3.2.2 Fuel Handling System Description

The fuel handling system provides a safe and efficient method to unload and store new fuel assemblies in the fuel building, transfer spent fuel assemblies in the fuel pool, and to ship spent fuel assemblies off-site. Equipment is also provided to transfer control element assemblies to guide tubes in each fuel assembly, as required. The refueling equipment arrangement is shown on Figure 3.3-8. The system is comprised of the following:

Spent Fuel Movable Platform and Hoist
Yard Area Crane
Communications

<u>Spent Fuel Movable Platform and Hoist</u>

The basic structure of the movable platform and hoist is a traveling bridge which spans the spent fuel pool and moves on rails over any spent fuel storage position, the new fuel elevator and the transfer system upending machine. A fuel hoist is mounted on the bridge structure. The hoist hook supports handling tools for grappling fuel assemblies below water. The rotation of fuel is manually controlled via grapple tool. All operations may be monitored by binocular viewing as required.

CR-9 has the following three basic interlocks:

1. An electrical cutout that stops movement in the north direction if CR-9 gets to close to CR-6. CR-6 has a similar interlock to prevent it from driving into CR-9.
2. An interlock that prevents concurrent operation of the CR-9 hoist with either the bridge or the trolley.
3. An interlock that limits upward travel of the hoist (two block interlock).

Yard Area Crane

The 125/10 ton fuel building yard crane, CR-3, is an overhead multiple girder bridge crane with a top-running trolley. It was originally designed to the requirements of the Electric Overhead Crane Institute (EOCI) Specification #61 for Class A service. The design was later verified to comply with the Crane Manufacturers Association of America (CMAA) Specification #70-1975. CR-3 has been upgraded in accordance with the single-failure-proof guidelines of NUREG-0612/0554, including a complete trolley replacement. The single-failure-proof main hoist has a 125 ton design rated load (DRL) with a 115 ton maximum critical load (MCL) rating. The trolley also has an auxiliary hoist with a 10 ton DRL. Although it is not single-failure-proof, the auxiliary hoist has enhanced safety features and has been given a reduced 5 ton MCL rating, doubling the mechanical/structural factors of safety for handling loads associated with spent fuel casks (e.g., cask lids). CMAA Specification #70-1999 and ASME NOG-1-1998 were used in the design of the new trolley.

The main hoist features dual (redundant) reeving systems, with each system providing independent load balance on the head and load blocks through configuration of the ropes and rope equalizers. Design of the equalizer system includes an energy-absorbing crush-pad that limits the load on the intact reeving system in the event of a broken rope. Mechanical components of the hoist system that are not redundant are designed with increased factors of safety to reduce the probability of failure. Two independent holding brake systems are included for the main hoist. The primary brake is electric and located at the motor. The secondary holding brake consists of air-operated calipers acting on discs integral with the main hoist drum. Both braking systems are fail-safe and provide the capability to manually lower a load. An overspeed limit switch is incorporated to prevent uncontrolled lowering by setting both holding brakes. Redundant upper and lower limit switches are provided for overtravel, as well as sensors for detecting proper winding of the rope on the drum and overweight conditions. An unbalanced load limit switch is also supplied that detects movement in the equalizer system, which would indicate one of the two main hoist ropes has stretched or yielded.

3.3.2.3 Design Evaluation

Fuel handling equipment is designed Seismic Class 1 as it is necessary that load bearing components and brakes operate properly to prevent situations which could cause damage to the fuel pool or the fuel. The fuel platform and hoist and yard crane are designed to Seismic Class I requirements and the interlocks are fail-safe. The design of load handling equipment complies with applicable industry standards and codes. The design of the hoists and cranes coupled with the administrative controls provide assurance that a heavy load will not be dropped which could result in damage to safety related equipment and loss of required safety functions. Rev. 18

To assure the safe handling of fuel and casks, design safety features include, as necessary: mechanical stops, brakes, limit switches, and electrical interlocks. These design controls, together with the administrative controls (e.g., operator training, periodic load tests, functional tests) provide assurance that the consequences of accidents will remain well below the 10CFR 100.11 and EPA PAG limits. In addition, administrative controls restrict handling of heavy loads in the vicinity of safety related equipment.]

The fuel handling SSCs designed to Seismic Class I requirements are the fuel platform and hoist and yard crane.

The following fuel handling system functions were credited in section 3.3.1

- The fuel platform and hoist and yard crane design safety features assure the safe handling of fuel and casks.]
- Design controls assure that the spent fuel assembly is not lifted greater than 22.5 inches over the top of the racks for loads up to 2000 lbs. And not more than 18 inches above the racks for loads in excess of 2000 lbs (but less than 2500 lbs).]

The following actions were credited during the events:

- Assuring that appropriate tests and inspections were conducted on fuel handling equipment prior to the movement of heavy loads and that such movements are administratively controlled to prevent movements in the vicinity of safety related equipment.
- Assuring the placement of fuel in the proper rack region. Refer to figures 3.3-5 and 3.3-6 for the proper placement of fuel in the racks as a function of burnup.

3.3.2.4 System Operations

The following operations controls are implemented for the spent fuel movable platform and hoist:

a. Prior to conducting fuel handling, a complete checkout, including a load test using a dummy assembly, shall be conducted on the fuel handling crane used to handle irradiated fuel assemblies (prior to such use unless checked out in the last 18 months).
b. Prior to conducting fuel handling operations (i.e., prior to handling irradiated fuel assemblies unless tested in the last 18 months), fuel handling equipment interlocks functional test shall be conducted.
c. The spent fuel hoist is controlled to assure that the assembly is not lifted more than 22.5 inches over the top of the racks for loads up to 2000 lbs. And not more than 18 inches over the top of the racks for loads in excess of 2000 lbs. (but less than 2500 lbs).

The following controls are implemented regarding the yard crane

a. All phases of fuel cask movement are under strict administrative control, with all movement done in accordance with written instruction and check-off lists. The operation is limited to trained and certified personnel or is under the direct visual supervision of an individual with training and certification in the operation. Supervisory personnel who personally direct the operation of the equipment and controls are also certified in such operations. The crane operators are trained to meet the requirements of USAS B30.2-1976, Overhead and Gantry Cranes.

b. Documented maintenance and performance checks, including a loaded operational test of the crane controls and brakes, are performed prior to each fuel cask handling evolution.

The following requirements were relocated from the technical specifications:

1. The following conditions shall be satisfied during movement of irradiated fuel within the spent fuel storage building:
    a. Radiation levels in the spent fuel storage area shall be monitored continuously.

2. Spent fuel storage racks may be moved only in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

3.3.2.5 Inspection and Testing

The yard area crane receives annual maintenance examinations by qualified personnel.

3.3.3 Spent Fuel Pool (SFP) Decay Heat Removal (DHR) System

3.3.3.1 Design Basis

Design Criteria

The Spent Fuel Pool (SFP) Decay Heat Removal (DHR) system is designed to be a reliable source of heat transfer from the SFP water. It removes decay heat from pool water to maintain temperature within specification. The system is designed to preclude the possibility of radioactive leakage from reaching beyond the plant boundary.

Design Basis

a. The secondary cooling Loop (DHR) system is an air-cooled, closed-loop cooling system.
b. The system piping and components are classified as Non-Nuclear Safety/Important To the Defueled Condition (NNS/ITDC). The system is constructed and maintained in accordance with appropriate industry codes and standards.
c. The system is protected during periods of prolonged freezing temperature conditions.
d. Alternate sources of cooling can be effected in the event that the DHR system is out of service. Evaporation and pool water makeup are sources of alternate cooling.
e. The system is capable of maintaining bulk temperatures ≤120°F with an outside temperature of 87°F with a duty of 2.88 million BTUs per hour.
f. The system is able to recover from a fuel pool boiling incident where the water in the primary and secondary system may reach 212°F.
g. Pool water bulk temperature is maintained above 40°F.

3.3.3.2 System Description

The SFP DHR system flow diagram is shown in Figure 3.3-9. SFP DHR provides cooling water service for spent fuel pool decay heat removal. SFP DHR is primarily filled with deionized water, with additives to protect against freezing and corrosion. Heat generated by the spent fuel is transferred to the fuel pool water, then to the DHR system in the SFP heat exchanger (E-25) and lastly to the air by water-to-air coolers (E-SFP 1 to 6). The DHR solution is circulated by one of two DHR cooling pumps. An expansion tank and air separator are also provided.

Local hardwired controls and indications are the primary means of controlling the Spent Fuel Pool Island (SFPI) equipment. A highly reliable non-nuclear safety related computer system, referred to as the Programmable Logic Controller (PLC), is also provided. The PLC monitors, displays, and alarms certain desired DHR SFP parameters to the operators in the SFPI control room. Limited DHR loop control functions are provided by the PLC as discussed in 3.3.3.5.

The Primary Water Storage Tank provides a makeup source to the DHR loop via a manual hose connection. In the event that makeup water is required, deionized water in the PWST is pumped via P-SFP2 located in the fuel building to a hose connection in the secondary loop.

SFP DHR Pumps

Two parallel SFP DHR pumps are provided and each deliver a nominal flow of 1000 gallons per minute. One pump is usually in service. The other is an installed spare. The pump internal wetted surfaces are stainless steel.

Water-to-Air Coolers

The DHR water-to-air cooler is composed of six parallel, finned-coiled type coolers; each cooler containing three fans. The SFP DHR coolers may be bypassed to maintain system flow balance and desired heat transfer rate. The heat transfer rate is controlled through the cycling of the cooler fans.

Deionized water is used in the DHR loop during the first and possibly future summers to maximize heat transfer. Ethylene glycol will be added for the winter months. The units are designed to 300 psig and 350°F. Normal operating conditions are anticipated to be approximately 80 psig and 85°F. The tubes and fins are constructed of stainless steel and aluminum, respectively.

DHR Expansion Tank

The DHR bladder expansion tank is sized to accommodate fluid expansion and contractions resulting from temperature variations in the secondary cooling loop. The tank capacity is 211 gallons. All wetted portions of the tank are constructed of stainless steel. The bladder is constructed of butyl rubber.

Air Separator

The air separator removes entrained air in the secondary coolant. The unit and trap are made of stainless steel.

Piping System

The piping and valves used throughout the system are constructed of stainless steel with the exceptions of a small portion of carbon steel piping used at the inlet and outlet of the secondary side of E-25 and the shell side of E-25.

The SFP DHR system uses corrosion inhibited deionized water. A connection is provided on the DHR loop to add corrosion inhibitors as required.

Codes

The design of equipment in the SFP DHR comply with the following codes:

Shell side of Hx (E-25)------------------------------------ ASME, Section VIII
Surge Tank-- ASME, Section VIII
Coolers-- Commercial Industrial Standards
Air Separator--- ASME, Section VIII
DHR Pumps--- Commercial Industrial Standards
Piping, Valves and Fittings------------------------------- ANSI, B31.1-1980

3.3.3.3 Design Evaluation

The SFP DHR system does not perform a safety related function. Analyses demonstrate that the fuel and fuel pool structural components are capable of sustaining boiling as a service condition. For operational considerations, pool temperatures are normally maintained below 120°F. In addition, adequate time is available to the operators to identify and remedy a low pool water level condition or loss of cooling incident. Standby power sources are not required.

The secondary cooling loop flow rate is controlled by the DHR cooling pumps. The typical operating flow rate is maintained below 1100 gpm; the manufacturer's maximum recommended shell side flow rate for E-25. Flow through the heat exchanger is controlled by throttling the heat exchange discharge valve. For pump protection, and as an indication of reduced system inventory, there is a suction pressure switch on each pump which trips the pump on low suction pressure.

Six water-to-air coolers in parallel provide the heat sink necessary to maximize heat transfer from the fuel pool. Each cooler may be bypassed to maintain system flow balance and desired heat transfer rate or facilitate cooler maintenance activities. Over pressure protection of the secondary loop is also provided.

Calculations were performed to determine the capacity of the DHR system. The assumptions used in the analysis are as follows:

Fuel Pool Design Decay Heat Load (6/22/99)	2.88 E+6 BTU/hr
DHR (secondary loop)	
Flow rate	1000 gpm
Shell Side Inlet Temperature	108°F
Discharge	113°F
Outside Air Temperature	87°F
Fuel Pool Cooling Pumps	
Flow Rate	1700 gpm (1800 max)
Tube Side Inlet Temperature of	116°F

The results show that the outlet temperature of the tube side of the heat exchanger would be 113°F.

Abnormal Conditions

Each of the following abnormal conditions are evaluated for its effects on the ability of the SFP DHR system to provide heat removal from the spent fuel pool heat exchanger and to preclude the possibility of radioactive leakage from reaching beyond the plant boundary.

Increasing System Temperature - Changes in SFP DHR temperature may result from seasonal changes in the air temperature and humidity. Secondary coolant in the tube side is air-cooled by the cooling fans which force ambient air across the coil to remove heat. The DHR fans are cycled to obtain the desired heat transfer rate. Under worst case heat loads, the secondary cooling system is designed to maintain the pool temperature below 120°F.

Loss of SFP DHR - Loss of SFP DHR cooling would cause the pool temperature to increase. Various make-up sources are capable of providing water in the event of maximum evaporation (i.e., boil-off). The normal deionized water make-up is provided from the Primary Water Storage Tank. Alternative makeup sources include the Wiscasset water supply and the fire water system.

In the event that normal cooling is interrupted, alignment of alternate cooling supplies to the spent fuel pool heat exchanger may be provided by an electric or diesel driven fire water pump through flanged connections provided on the shell side of the heat exchanger. The fire water pond may act as both the water source and the ultimate heat sink. Additional means include the use of the existing fire hydrant adjacent to the fire pump house which is connected to the Wiscasset water supply.

Loss of SFP DHR cooling to the spent fuel pool heat exchanger is bounded by the "Loss of Spent Fuel Pool Cooling" incident described in Sections 3.3.1 and 5.0. An abnormal operating procedure describes the steps to supply alternate cooling from the fire system through a connection to the fire pond. The fire system has ample capacity to provide alternate cooling to the spent fuel pool heat exchanger.

<u>Leakage Into/Out of the System</u> - Spent fuel pool water is periodically monitored for radiation levels. In addition, the purification loop assures that radiation levels in the pool are minimized to preclude the existence of significantly contaminated water. Leakage from the pool into the SFP DHR system is unlikely since the SFP DHR system is maintained at a higher pressure than the fuel pool cooling system. If leakage were to occur through the heat exchanger, the leakage would be diluted by the volume of SFP DHR and result in an insignificant amount of SFP DHR System contamination. In order to identify potential leakage, the SFP DHR water is periodically sampled for radioactive contaminants. Leakage into the SFP DHR system may be indicated by low pressure drop across the heat exchanger, or by loss of pool water level.

Freeze protection and corrosion control is effected in the secondary loop by the use of ethylene glycol and dipotassium phosphate, respectively.

<u>Loss of Off-Site Power</u> - If offsite electrical power is lost, the SFP DHR pumps will shutdown. To restore power to the SFP DHR pumps, the onsite diesel generator may be manually started. As described in Section 3.3.1, there is sufficient time to restore power.

3.3.3.4 System Operation

The fuel pool heat exchanger E-25 transfers heat from the primary (tube) side pool water to the secondary (shell) side SFP DHR cooling loop. The DHR pumps circulate water from the fuel poolheat exchanger to the water-to-air coolers and back to the shell side of the fuel pool heat exchanger.

The DHR bladder expansion tank accommodates fluid expansion and contraction resulting from temperature variations and an air separator removes entrained air in the secondary coolant. The pumps are located in the fuel building at elevation 21 feet. One pump is normally operating and the other is an installed spare.

The water-to-air coolers reject heat from the DHR system into the surrounding atmosphere by the use of finned tubes and forced air. The coolers are located in a diked area. A gabled roof over the diked cooler reduces the collection of precipitation inside of the dike. The diked cooler area is released to storm drains after it is determined that the fluid is compliant with waste discharge specifications. The operation of the cooler fans are staggered to adjust for heat rejection rate variations due to variation in ambient air temperature. Flow is provided to all the coolers whether or not the fans are operating. Cooler bypasses are installed to allow for maintenance. During winter operation some of the fans may be in standby as radiant and natural convection heat transfer may remove sufficient heat from the secondary system.

The DHR pumps are provided with a local "stop-run" control switch. One pump is likely to run continuously year round; however, the pump may be secured to prevent overcooling, as required. Pump run-time is a function of the decay heat in the spent fuel pool. As the decay heat decreases over time, the pump may be isolated more frequently to prevent overcooling.

3.3.3.5 Monitoring and Instrumentation

There are two automatic actions performed by the DHR System instrumentation (automatic trip of the DHR pumps on low suction pressure and cycling on/off the fan coolers) and they are both non-safety related.

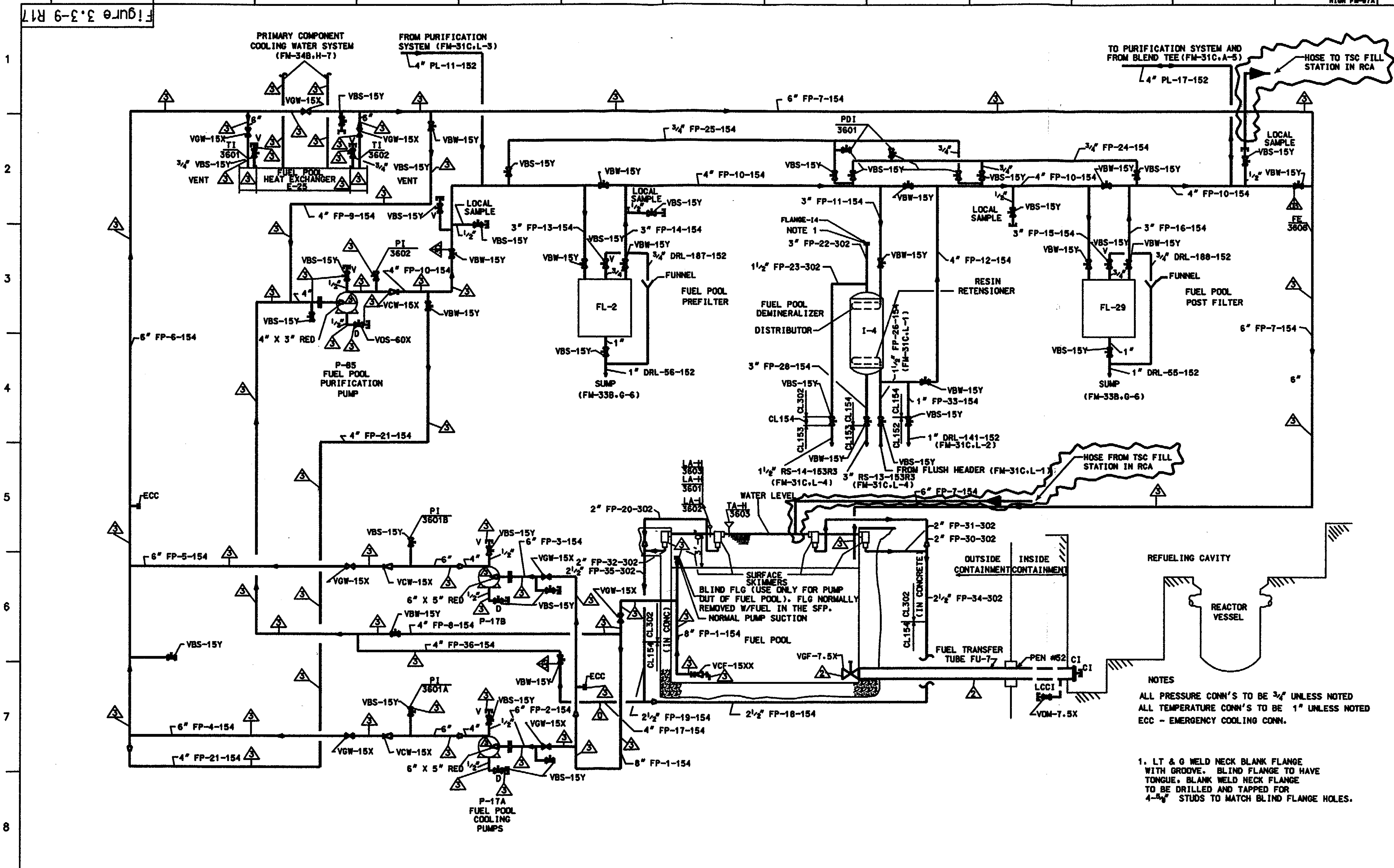
Both DHR pumps are provided with suction and discharge pressure transmitters and gauges. The suction pressure transmitters provide a low pressure alarm in the control room and initiate a DHR pump trip via the PLC in the event that a low suction pressure condition exists. In addition, pump discharge pressure transmitters provide continuous indication in the SFPI control room.

The water-to-air cooler fans are controlled from a temperature instrument on the fuel pool heat exchanger outlet. The temperature indicating transmitter information is fed to the PLC. The PLC automatically cycles the cooling fans to attain the desired cooling rate.

In the event of a programmable logic controller malfunction or out-of-service condition, the capability exists to disable the system and locally secure or initiate control of DHR pumps and fan coolers. A primary to secondary differential pressure switch is provided for local indication and low differential pressure alarm in the SFPI control room.

The system is monitored for proper concentrations of a corrosion inhibitor (dipotassium phosphate) and the degree of freeze protection capability.

In order to identify potential leakage, the SFP DHR water is periodically sampled for radioactive containments.



NOTES

ALL PRESSURE CONN'S TO BE 3/4" UNLESS NOTED
ALL TEMPERATURE CONN'S TO BE 1" UNLESS NOTED
ECC - EMERGENCY COOLING CONN.

1. LT & G WELD NECK BLANK FLANGE WITH GROOVE. BLIND FLANGE TO HAVE TONGUE. BLANK WELD NECK FLANGE TO BE DRILLED AND TAPPED FOR 4-5/8" STUDS TO MATCH BLIND FLANGE HOLES.

FLOW DIAGRAM
FUEL POOL COOLING PIPING

Maine Yankee

Figure 3.3-9 Rev. 17

3.3.4 Ventilation Systems

Ventilation systems are designed to provide a suitable environment for equipment and personnel. These systems provide heating and air conditioning. The ventilation systems utilize fans, filters, dampers, heating elements, cooling elements, and duct work to accomplish the desired effects. In the radiologically controlled areas, the fuel building and the primary auxiliary building, outdoor air is supplied to these structures. Air is exhausted in greater quantities than it is supplied to maintain the building at a negative pressure and ensure that the general flow of air is into the structure. The exhausted air is discharged past radiation monitors in the primary vent stack or Fuel Building exhaust duct.

Relocated Technical Specification Definition of Ventilation Exhaust Treatment System
The Ventilation Exhaust Treatment System includes all systems designed and installed to reduce radioactive material in particulate form in effluents by passing ventilation through HEPA filters for the purpose of removing particulates from the exhaust stream prior to release to the environment. Such systems are not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment Systems components.

Inspection and Testing

Fuel Building Ventilation System

Inside the fuel building is the fuel storage pool. The stored fuel will remain in the building until it is transferred to dry-casks or off-site. The fuel building will remain independent of other systems, structures, or components undergoing the decommissioning process. In this regard, the fuel building ventilation system will remain operational. The fuel building ventilation system has Administrative Controls established to ensure the following Inspection and Testing requirements are satisfied:

- Requirements for an operable ventilation system
- Requirements for demonstrating operability
- Requirements for testing and test frequency

Control Room and Auxiliary Ventilation Systems

The remaining ventilation systems consist of the Control Room Ventilation and Auxiliary Ventilation Systems. These systems are in buildings where decommissioning activities will occur. Maine Yankee has committed to the intent of NUREG/CR 0130. These ventilation systems will be subject to the testing and operability requirements of DSAR, Chapter 7, DECOMMISSIONING.

3.3.4.1 Fuel Building Ventilation System

3.3.4.1.1 Design Basis

The fuel building ventilation system is designed to: maintain the operability of the fuel building equipment during normal operating conditions, ensure that air flow is from outside into the building to prevent unmonitored release of radiation, and ensure that exhaust air is continuously monitored.

3.3.4.1.2 System Description

The previously existing fuel building ventilation system has been replaced with one designed to support the needs of the spent fuel pool island. Variable speed exhaust fan HV-SFP1 draws from 2000 cfm to 10,000 cfm of cooling air through a louver and filter assembly in the northwest wall,]
and discharges the air through a duct mounted on the exterior of the fuel building east wall. Before it is discharged to the outside, the exhaust air passes through HEPA filters to remove any particulate materials. Because the exhaust system is once through, the fuel building is always maintained at a slight negative pressure. The heating and ventilation equipment in the fuel building is designed to minimize moisture condensation on the walls and the roof and to limit the space temperature to a maximum 97°F and a minimum of 60°F. Exhaust from HV-SFPI & RCA]
Fan FN-30 discharges into a common duct located on the PAB roof. A radiation monitoring]
system is installed in the common exhaust duct which continuously monitors the discharge air to]
identify any potential releases. Fan FN-SFP1 supplies unconditioned building air to the heat exchanger cubicle to aid in cooling the components located there.

3.3.4.1.3 Design Evaluation

Since the consequences of the Fuel Handling Accident are significantly below 10CFR100.11 limits without taking credit for the building ventilation, the fuel building ventilation system is not designated to mitigate the consequences of a Fuel Handling Accident. Therefore, the fuel building ventilation system is not safety related.

3.3.4.2 Auxiliary Ventilation Systems

This section discusses the ventilation systems at the plant. The ventilation systems include: primary auxiliary building, front office/control room building, Wart building, containment building, containment spray building, and RCA building ventilation systems. These ventilation systems do not perform any safety functions in the permanently defueled condition.

3.3.4.2.1 Primary Auxiliary Building Ventilation System

Ventilation air to all PAB areas is drawn through the HV-2 inlet wall louver. Both upply units, HV-1 and HV-2 have been removed from service and the inlet duct of HV-2 has been modified to allow introduction of fresh unconditioned outside air through the louvered opening on elevation 36. The HV-1 wall louver has been covered on the interior side with a closure cap for maintenance of building negative pressure.]

PAB exhaust is collected from elevation 36 feet. The exhaust is filtered by a HEPA filter assembly (previously referred to as the PAB tray filter) and is normally discharged by fan FN-1B to the primary vent stack. PAB miscellaneous exhaust has been removed from service. FN-1A is normally aligned to exhaust the Containment through a HEPA filter assembly (previously referred to as the PAB safety class filter) and discharged to the Primary Vent Stack. Either fan, FN-1A or 1B, may be aligned to exhaust the Containment.]

Each exhaust fan has a nominal capacity of 27,000 cfm. Exhaust fan FN-2, which previously served the 36 foot elevation, has been removed from service. The supply and exhaust airflows are balanced to ensure that the PAB is maintained at a slight negative pressure relative to the outside, as well as to the Service Building and the new HP checkpoint walkway.

3.3.4.2.2 Containment Building Ventilation System

The Containment Building is ventilated by unconditioned outside air drawn through the containment personnel hatch, and through the HV-9 housing, by fan FN-1A or FN-1B. The effluent air is filtered through a HEPA filter assembly and discharged to the primary vent stack. HV-9, previously the containment supply air unit, has been removed from service, and its filters and steam heating coils have been removed to decrease the resistance of the airflow path. Because the exhaust system is once-through, the containment is always maintained at a slight negative pressure relative to the outside. FN-1A and 1B may also be aligned to exhaust the PAB.

3.3.4.2.3 Containment Spray Building Ventilation System

The Containment Spray Building is ventilated by exhaust fan FN-44A. Unconditioned outside air is drawn into the building through the HV-7 bypass duct. After cooling the spaces in this building, the air is passed through HEPA filters and discharged to the primary vent stack. Supply unit HV-7 has been removed from service, and exhaust fan FN-44B will not be re-powered. The Containment Spray Building is maintained at a slight negative pressure relative to the adjacent spaces as a result of the once-through ventilation system design.

3.3.4.2.4 RCA Building Ventilation System

The RCA Building will continue to be used for processing radiological waste during decommissioning. Supply unit HV-4 provides unconditioned filtered outside air to this building, and exhaust fan FN-30 exhausts this space. The air exhausted from the RCA Building is passed]
through HEPA filters and discharged to a common exhaust duct with the FB discharge from HV-]
SFPI. The supply and exhaust airflows are balanced to ensure that the RCA Building is]
maintained at a negative pressure relative to the outside. Building heat is provided by four thermostatically-controlled electric unit heaters.

3.3.4.2.5 Front Office/Control Room Building Ventilation System

The Front Office Building (Gatehouse) is served by a central air handling unit, AC-3, two roof-mounted heat pumps, and five exhaust fans. AC-3 is a six zone unit which conditions (filters and heats or cools) and recirculates the building air, while also introducing a continuous stream of fresh air to all the building spaces it serves, including the Spent Fuel Pool Island Control Room.

There are no automatic signals which isolate ventilation to the Spent Fuel Pool Island Control Room. In the permanently defueled condition, the reduced source term (Section 5) precludes the need for automatic isolation of the Control Room ventilation system. The filter bank in AC-3 consists of disposable roughing filters, and heat is provided by a multi-stage electric heating coil installed in the unit's discharge plenum. Supplementary building heat is provided by electric baseboard heaters. Ceiling exhaust fans in the PBX room, the Spent Fuel Pool Island Control Room assure positive airflow to these spaces.

3.3.4.2.6 Administration Building (WART Building)

The Administration Building houses the body count room, the chemistry count room, and the relocated RCA checkpoint on the first floor, and office spaces on the second and third floors. Each floor is ventilated, cooled, and heated by its own dedicated HVAC system. The PAB ventilation system maintains the PAB at a negative pressure relative to the RCA checkpoint walkway. This assures that any potentially radiologically contaminated air in PAB cannot enter the walkway or the RCA checkpoint.

3.3.5 Auxiliary Systems

This section discusses auxiliary systems supporting plant operating equipment. These systems do not perform any safety related functions in the permanently defueled condition.

3.3.5.1 Boric Acid Makeup

Design Criteria
Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

System Description
A self-contained batch tank mixer and pump unit is provided to increase boron concentration in the pool, if required.

Design Evaluation
Irradiated spent fuel is stored under water in a reinforced concrete pool, lined with stainless steel. Fuel assemblies are spaced and the racks are fabricated so that criticality is precluded. Although the water in the pool is generally borated, neither soluble boron nor control rods are required to maintain the irradiated fuel subcritical.

Boric acid concentration requirements are only applicable in the time period immediately preceding fuel handling movements, during fuel handling movements, and up to the time that an assembly placement verification is performed. Boric acid makeup capability is provided to account for pool water dilution due to the addition of demineralized water makeup used to compensate for losses of borated water due to pump seal, maintenance activities or other minor pool water leakage paths. The boron concentration required as a result of the analyzed "misplaced assembly" incident is discussed in section 3.2.

Bulk pool water boric acid concentration generally increases under routine or non-routine (boil-off) evaporative losses in the pool. Accordingly, when routine demineralized water is added to the pool to account for boil-off or natural evaporation, the addition of boron should not be necessary.

Major losses of pool water due to draindown, or dilution of boron due to flooding, are extremely unlikely, but are the only methods through which significant makeup water is added to dilute the pool water.

For the reasons stated above, loss of boron concentration under normal storage conditions is not a significant event from the standpoint of criticality.

Significant dilutions can occur as a result of adding demineralized water to compensate for a draindown event or from flooding from a nearby pipe break containing water. Flooding due to natural phenomena is not possible due to the elevation of the spent fuel pool.

To preclude flooding, design or administrative controls are implemented. Design considerations may include one or a combination of the following: elevation and location of significant water sources, berms, freeze protection (e.g., heat tracing, heating, draining), drains, instrumentation or valves in piping systems. For instance, due to its elevation and location, a failure of the primary water storage tank would have no flooding affect on the pool. Similarly, the fire protection system is comprised only of detection equipment and is limited only to external fire suppression capability. Therefore, a fire protection pipe break or inadvertent actuation of the system causing a flooding condition is not possible.

If a primary water pipe break (e.g., hose nozzle) occurred in the fuel building at the 46' elevation, the water would drain to the fuel building sump or into the fuel pool. During routine rounds, the operator would identify the leakage visually or by excessive sump pump run time via an integrator. In the event that the break was significant, sump pumps would pump to the Waste Holdup Tank (TK-109). In addition, a berm exists around the perimeter of the pool to preclude draining of water into the pool. Flood levels exceeding the height of the berm is prevented by the existence of other leakage paths (e.g., doors, drains, etc). In the event that, for any reason, water level in the pool was to rise, a pool water high level alarm would alert operators to the condition. In both cases, operators would be dispatched to the area to investigate the cause of the condition and effect corrective actions, as appropriate.

Administrative controls include maintaining a temperature in the fuel building to preclude pipe freezing. Routine operator walk-throughs can also be credited. The defense-in-depth of the design and administrative controls precludes flooding from being a significant incident relative to boron dilution. The design and administrative controls, including reactivity control via the Boral plates, water moderator and spacing of the assemblies, provide reasonable assurance that a provision to add boron via manual means is sufficient to meet technical specification requirements.

The following SSCs were credited in the above discussion:

- Fuel building sump pump and high level alarm
- Fuel building sump pump integrator
- Spent resin pit sump pump and high level alarm
- Waste holdup tank (TK-109)
- Fuel pool high level alarm

Tests and Inspections

Tests and inspections of instrumentation sensors and alarms are periodically conducted in accordance with plant procedures.

Daily operator rounds are conducted to identify any flooding conditions, excessive sump pump run time (integrator), and potential freezing conditions.

3.3.5.2 Primary Water System

The primary water makeup system provides a source of demineralized water for use by various locations in the plant. Demineralized water is stored in the primary water storage tank. The primary water storage tank is prevented from freezing by a combination of wall insulation and an electric heating element.

Makeup to the primary water tank is provided from the Wiscasset potable water system. The potable water is processed through a truck-mounted demineralizer system prior to being pumped to the primary water storage tank. Chemistry periodically samples the output of the demineralizer system to ensure proper water quality.

The following systems, components, or areas are supplied by the primary water system:

- Spent fuel cask decontamination area
- Fuel Building hose connections
- Spent Fuel Pool Island (SFPI) make-up

3.3.5.3 Primary Vent and Drain System

The majority of the Primary Vent and Drain System has been abandoned IAW the Maine Yankee decommissioning process. Portions of the original system have been retained and re-configured IAW MY DPR 98-003 to provide for the safe and limited collection, transfer and retention of potentially radioactive waste. The system re-configuration provides for collecting sources of waste from the Spent Resin Pit and Fuel Building Sumps and transfer of these wastes to the Waste Holdup Tank (TK-109).

3.3.5.4 Radioactive Waste Processing System

The radioactive waste processing system was originally designed to collect, store, process, monitor, and dispose of solid, liquid, and gaseous wastes from the plant. Sections 4.4 and 4.5 provide details of the Liquid Waste Treatment and Solid Waste Treatment Systems as reconfigured for the decommissioning.

The Boron Recovery, Liquid and Gaseous Wastes Systems no longer serve a function in the Decommissioning and have been abandoned by Maine Yankee. The Decommissioning Operating Contractor has reconfigured the Duratek Liquid Waste processing skid to support waste water processing from decommissioning activities.

3.3.5.5 Fire Protection System

Fire detection and suppression systems are provided to minimize the adverse radiological consequences of fires at the plant.

The fire suppression water system consists of the fire pond; fire pumps, and distribution piping with associated sectionalizing control or isolation valves. These valves include yard hydrant curb valves and the first valve ahead of the water flow alarm device, which is provided for those sprinkler systems and standpipes which are normally pressurized with water. Water is supplied to inside fire suppression systems either by underground piping, or by connection to nearby fire hydrants using fire hose.

Supply water for the fire protection system is contained in the fire pond. Two 2500 gpm, 115 psi rated fire pumps, one electric and one diesel, take suction from the fire pond and discharge to the fire system loop. The motor driven fire pump will start automatically when system pressure drops to approximately 90 psig. The diesel fire pump will start automatically when system pressure drops to approximately 80 psig. A pressure maintenance system consisting of a pump, hydro-pneumatic tank and air compressor is designed to maintain system pressure between approximately 100 and 110 psig.

The discharge of the fire pumps is routed to the yard loop, as shown on figures 3.3-21 and 3.3-22. The loop consists of 12 inch underground piping which encircles the plant. Eight fire hydrants tap off of the loop, seven of which are provided with adjacent hose houses. Six of the hydrants have block valves which permit isolation of those hydrants from the fire mains.

Vertical and horizontal fire barriers have been established in designated areas of the plant. Fire doors and fire dampers have been provided as needed in these locations. Mechanical and electrical pentrations are generally sealed with fire retardant materials within or near the fire barrier.

3.3.5.5.1 Fire Detection Systems

Fire detection systems are installed in the following area:

1. Fuel Building (Drumming Room and Heat Exchanger Cubicle)]
2. LSA Building
3. Fire Pump House
4. Wart Building
5. X-14 Station Service Transformer]
6. Staff Building

The detection systems will send a signal to the Control Room, which is a constantly manned location. Manual pull stations are provided locally within all of these buildings except the Fire Pump House and the Fuel Building.

3.3.5.5.2 Hose Stations and Hose Houses

Eight hose houses are located exterior to the plant and are supplied from the 12-inch underground yard loop. The hoses are of various sizes and lengths pre-connected to the hydrants, depending upon the location.

Hose stations are located in interior areas of the plant. Each station consists of a water shut-off valve and wall-mounted hose rack. The hoses are of various lengths, depending upon the locations and each is fitted with a spray nozzle. A wet pipe system provides hose stations for the Staff Building off from the yard loop. Dry manual hose stations protect the following areas:

1. Fuel Building
2. RCA Storage Building
3. LSA Building
4. PAB]

Water is supplied to the PAB hose stations by operating a valve at the X-14 station service transformer deluge header in the PAB. Water to all other locations is supplied by a combination of valve line-ups and connection via hose to a nearby fire hydrant.]

3.3.5.5.3 Wet Pipe Sprinkler Systems

A wet pipe sprinkler system is located in Warehouse 2/3. The piping is filled with water at all times. The sprinkler heads have links designed to melt at various temperatures from 165°F to 286°F, depending upon the specific location. An alarm check valve in the supply line to this system will activate a Control Room alarm when flow is sensed.

3.3.5.5.4 Deluge Sprinkler System

A deluge sprinkler system protects the X-14 station service transformer. The spray header supplies open nozzles which spray water onto the transformer. The deluge valve for the transformer spray header is operated automatically by a signal from heat sensors at the transformer. The valve is also operated by remote manual pull stations or locally at the valve by manually operating a handle.]

3.3.5.5.5 Dry Manual Sprinkler System

Dry manual sprinkler systems protect the following areas:

1. Fuel Building]
2. RCA Storage Building
3. LSA Building

The piping to sprinklers is normally depressurized. The sprinkler heads have links designed to melt at various temperatures from 165°F to 286°F, depending upon the specific location. Water is supplied to these areas by a combination of valve line-ups and connection via hose to a nearby fire hydrant.

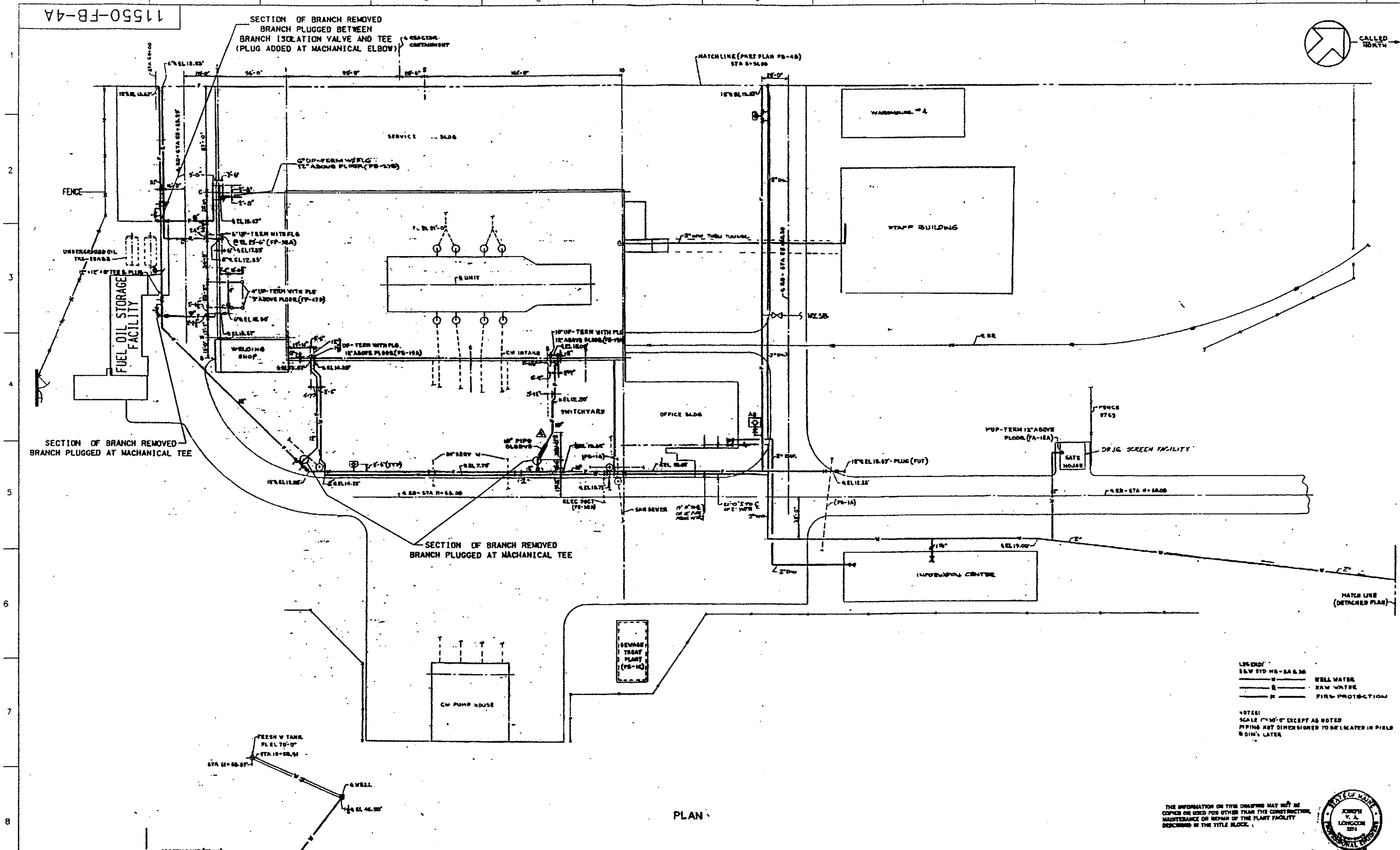


Figure 3.3-21 Rev. 17

FIRE PROTECTION SYSTEM

Maine Yankee

RELIABLE ELECTRICITY FOR MAINE SINCE 1972

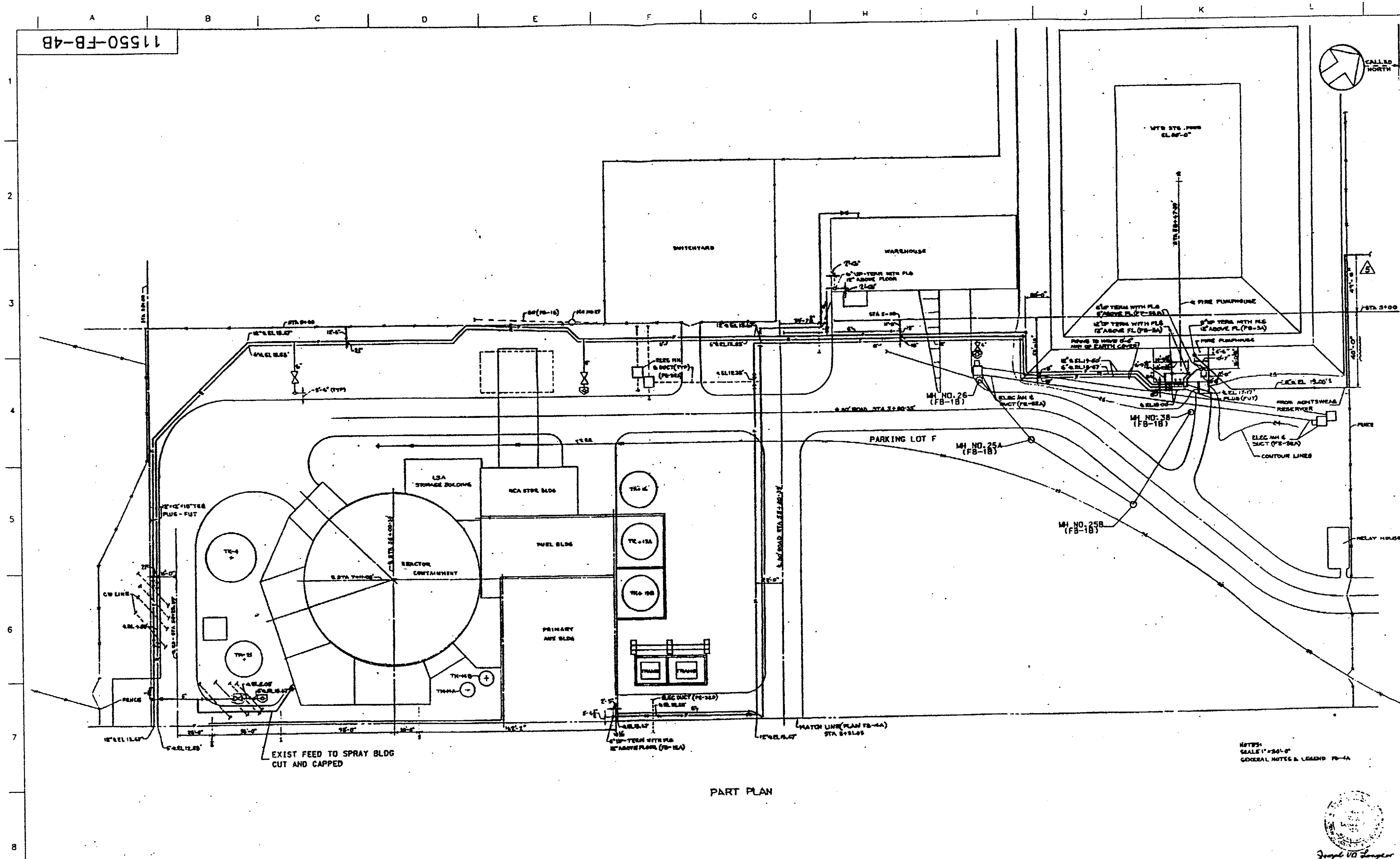


FIG. 3.3-22 REV.17

FIRE PROTECTION SYSTEM

Maine Yankee

RELIABLE ELECTRICITY FOR MAINE SINCE 1972

3.3.6 Electrical Systems

The station electrical system consists of two independent systems, one for the original plant equipment and systems and the second to support long term plant loads. These long term loads include the Spent Fuel Pool Island and station loads required during the decommissioning process. Sections 3.3.6.1 through 3.3.6.3 describes the plant electrical distribution system. Section 3.3.6.3 describes Spent Fuel Pool Island and long term plant load electrical distribution systems.

3.3.6.1 Offsite Power

Offsite power is supplied to Maine Yankee via a single 115 kV transmission line. This line is connected to the New England 345 kV power grid at the Mason and Surowiec Stations and other 115 kV lines in Topsham. This line provides two power feeds to Maine Yankee, through X-14 and X-5. The feed to X-14 is through a fused disconnect switch. X- 14 steps the 115 kV down to 4160 volts and it is connected to two 2.5 MVA stepdown transformers, X-16A and X-16B. X-16A steps the 4160 volts down to 480 volts to provide power to the SFPI loads. X-16B steps the 4160 volts down to 480 volts to power the remaining loads required to remain functional during the decommissioning . The feed to X-5 is through a fused disconnect switch. X-5 is used as a source of construction power. Figure 3.3.23 provides line arrangement.

The offsite power system is a reliable source of electrical power for plant equipment. In the defueled condition, no active systems meet the criteria for safety related systems or components as the consequences of accidents are significantly lower than the limits of 10CFR100.11. In the event of an interruption in power, the robust design of the passive systems assure the continued safe storage of fuel.

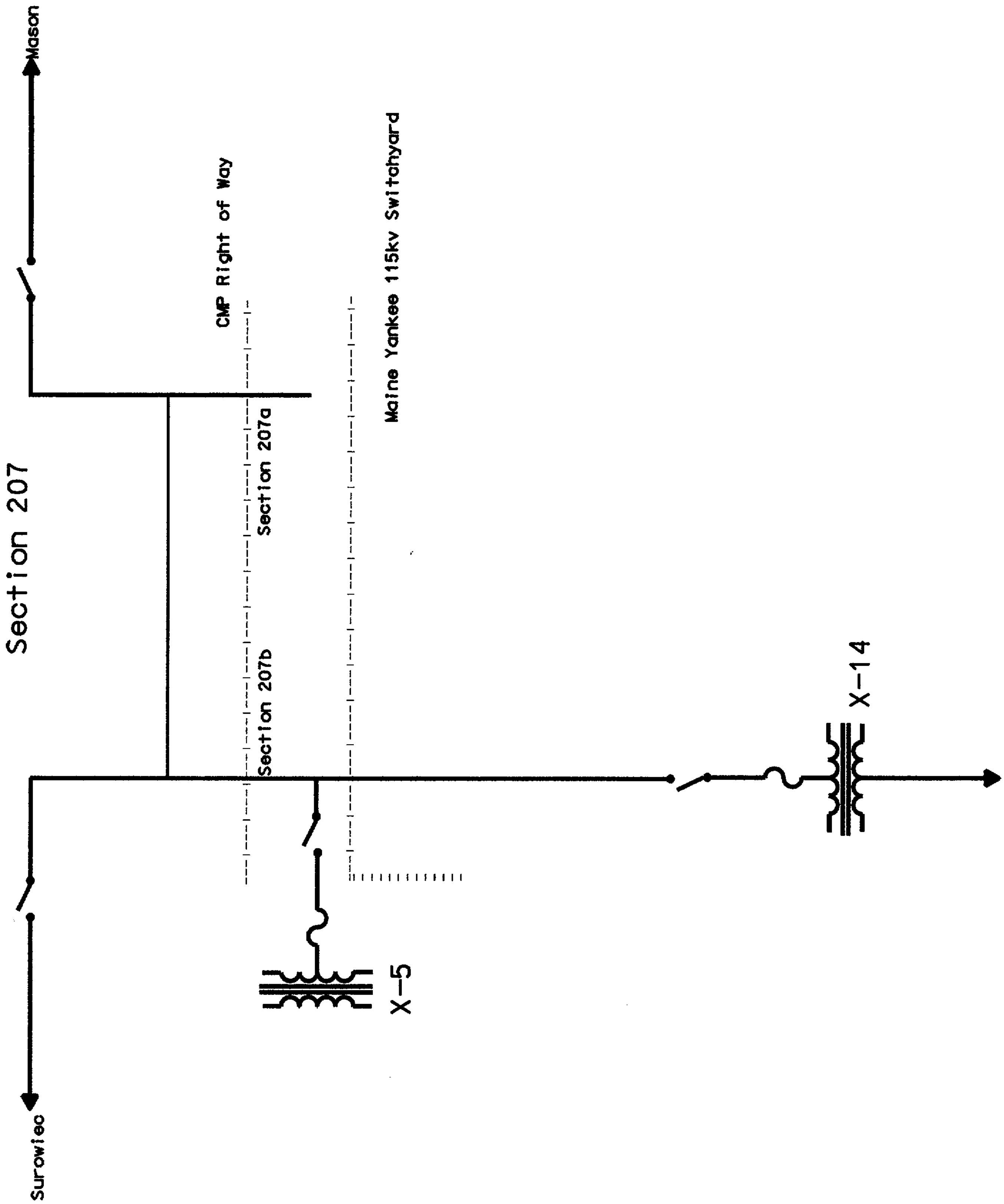


FIGURE 3.3-23
115kv Transmission
Rev. 18

3.3.6.2 Station Onsite Power

3.3.6.2.1 Auxiliary Power Systems

If the offsite power source is lost, a standby diesel generator may be used to power desired plant loads (Figure 3.3-24).

480 Volt System

There are two 480 volt bus sections, one dedicated to the SFPI and the other for balance of plant load. Normal supply to each of these sections is through individual 4160/480 volt, oil immersed self-cooled, transformers connected on the high voltage side to the 4.16 kV bus. Refer to Section 3.3.6.3 for additional detail.

3.3.6.2.2 Lighting and Heat Tracing System

Lighting System

Normal lighting for the Control Room, Spent Fuel Building, and Administration (WART) Building is supplied from the 480V electric distribution system through single-phase 480V/120-240V dry-type transformers. Battery back-up lighting is provided in vital areas for access, egress, or at pieces of equipment which may need to be operated during emergency conditions.

Heat Tracing System

Each of the lines containing water (i.e. PWST) subjected to freezing conditions is heat traced.

3.3.6.2.3 Communications Systems

Normal and emergency communication systems are described in the Maine Yankee Emergency Plan. These systems include those systems required to contact external emergency management agencies, officials and other government entities, and the general public.

3.3.6.2.4 Power and Control Cables

Cable Sizing and Rating

All cables are sized to operate within their normal rating and temperature rise by using conservative margins with respect to their current carrying capacities and insulation properties. Cable insulation was selected with due consideration to the radiation, temperature rise and humidity conditions.

The 480 volt power cables are insulated for at least 600 volt service and are suitable for use indoors and outdoors in wet and dry locations. Single conductor cables are jacketed, while three-conductor cables are triplexed or jacketed.

All control cables are rated for at least 600 volt service.

Cable Routing

Cables are separated when routed as medium voltage, low voltage, control, and instrument cables, when practical. Cable separation within the same group, but for redundant equipment or separate protective instrument channels is not required in the permanently defueled condition. There is no requirement for redundant electrical equipment.

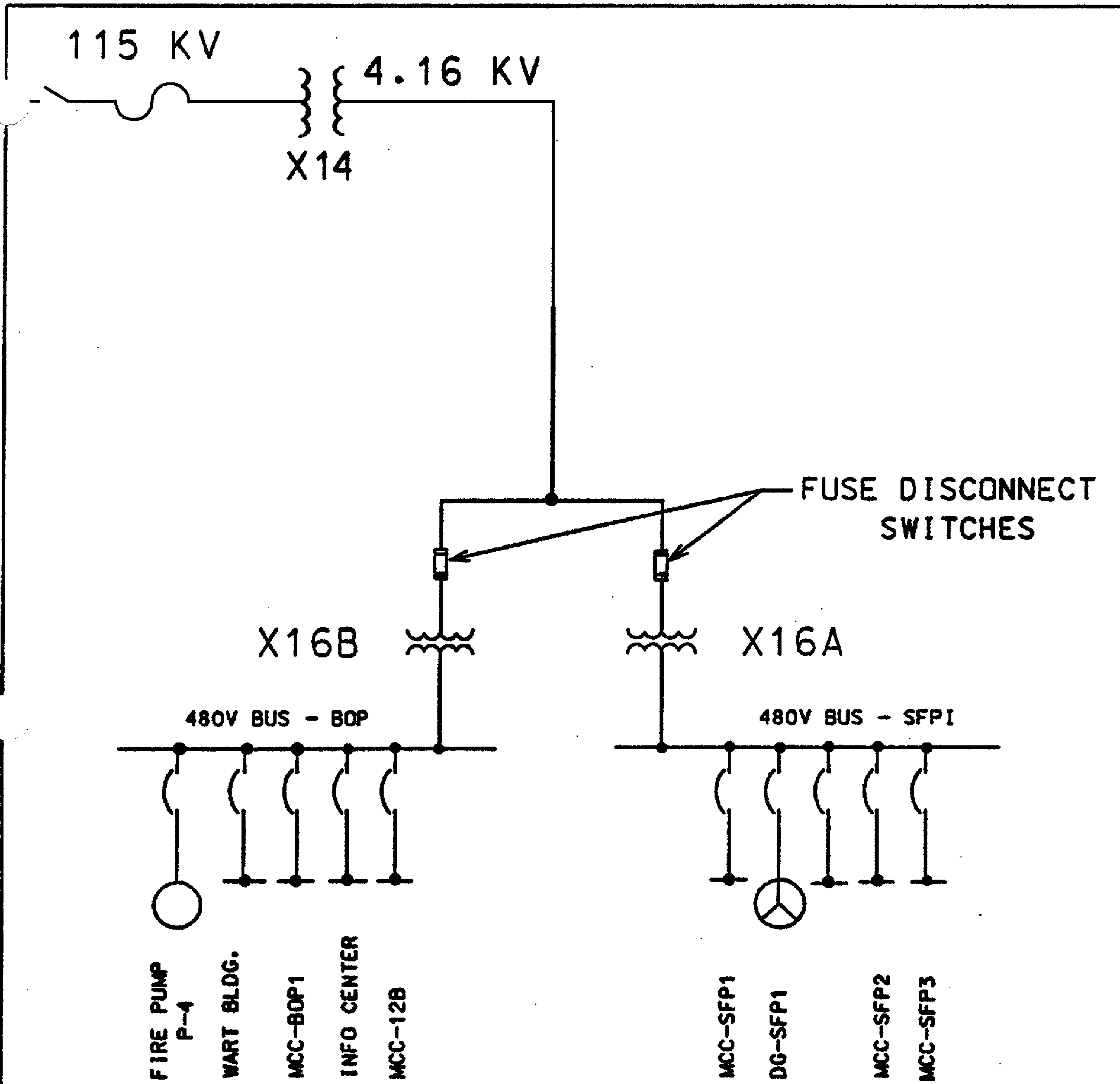
Cables are identified with their designation at regular intervals along their route and at raceway transitions

Cable Susceptibility to Fire

By routing cables in such a way as to avoid combustible materials, as well as carefully sizing and placing them in their trays, the chance of an electrical fire developing is minimized.

Control cables will not ignite from overloading or grounds, since the maximum fault is insufficient to heat the insulation to the flash point. Fire in control cables can, therefore, only occur as the result of another fire. Good housekeeping and sufficient care in cable routing reduces to a minimum the chances of external fires.

Power cables can carry sufficient fault current to reach the flash point of the cable insulation: however, protective relaying on the switchgear circuits will respond to fault currents and open the circuit before enough heat has occurred to damage the cable insulation and start a fire.



MAINE YANKEE ATOMIC POWER CO.	ONE LINE DIAGRAM TRANSMISSION AND UTILITY INTERCONNECTIONS WITH MAINE YANKEE	FIGURE 3.3-24 REV. 18

3.3.6.3 Spent Fuel Pool Island and Balance of Plant Electrical Distribution System

The Spent Fuel Pool Island and Balance-of-Plant electrical distribution system (SFP/BOP) provides power for the equipment to support the Spent Fuel Pooling Island operation and plant buildings and systems which will remain operational during the decommissioning process. The SFP/BOP electrical system is supplied by the 12 MVA, 4 kV winding of the original Maine Yankee station service transformer X-14, shown in Figure 3.3-24, from the 115 kV system described in section 3.3.6.1.]

The SFP/BOP electrical system is comprised of the following equipment: Station Service transformer X-14 and transformer protection is provided by line fuses installed on the 115 kV dead-end tower adjacent to X-14. The 4 kV winding is used to power two outdoor 480 V unit, substations Bus-SFP and Bus-BOP. Each unit substation is equipped with a set of 400 ampere fused disconnect switches, and a 2500 kVA, 4160 - 480/277 V oil-filled transformer. The Staff Building is powered directly from the BUS-BOP fused disconnect at 4160 volts.]

The unit substations are located on a concrete slab just to the west of transformer X-14.]

<u>480 V Bus-SFP</u>

The unit substation Bus-SFP is equipped with two electrically operated and three manually operated breakers. The substation is used to supply the following loads:

- MCC-SFP1 Fuel Building MCC (previously MCC-11B)
- MCC-SFP2 Fuel Building MCC (new)
- MCC-SFP3 Office Building MCC (previously MCC-13B)

]

<u>480 V Bus-BOP</u>

The unit substation for Bus-BOP is equipped with two electrically operated and five manually operated breakers. The substation issued to supply the following loads:

- WART Building
- Info Center
- MCC 12B (Fire Pump House)
- P4 (Fire Pumps)

Standby Power System

A portable diesel generator is on site with enough fuel to provide approximately 250 KW for at least 24 hours. This portable generator is normally wired to BUS-SFP and electrically isolated by a manual breaker that has a mechanical interlock to prevent parallel operation with the normal power supply. The generator must be manually started and manually switched to supply back-up power for limited BUS-SFP loads.

The diesel engine and associated diesel fuel tank is enclosed to prevent oil spills. Unprotected fuel transfer hoses are only used while an operator is present to control the transfer of fuel between tanks.

This portable diesel generator is also furnished with an electric engine heater, a generator heater and a battery charger that are energized as necessary to ensure operational readiness.

Uninterruptible Power Supply

Uninterruptible Power Supplies (UPSs) are provided to support continuous operation for the SFP Island Programmable Logic Controls (PLCs) and security system. The UPSs are rated for four hour operation. This will provide adequate time to re-establish ac power from the stand-by diesel generator.

120 VAC Distribution System and Lighting

120 VAC power is provided using original and new lighting panels. Emergency lighting for the Fuel Building is provided by self-contained, battery-powered emergency lighting units.

Raceway and Cable System

Since the equipment, raceways and cables associated with the SFP Island are intended to be operational through most of the decommissioning activities, a means of identification is essential.

Orange, day-glow orange, iridescent brand, etc. paint, tape, tags, labels, etc. is used on electrical components. Raceways and/or supports are painted orange, and existing tray side rails and conduits are painted orange at 3 to 5 foot intervals. Cables and equipment identification tags have an orange background. The load name plates for loads associated with the SFP Island are orange.

The following cable service separation is maintained in the raceways systems associated with the SFP Island.

- Large 480 VAC power (cables requiring spacing)
- 480 VAC power, 120 V and motor status circuits by will be installed in the same raceway without physical separation.
- Instrumentation circuits
- Control circuits

Instrumentation, Control and Alarms

Motor status indication is provided locally and, in some cases, at the SFP Island Control Room. Pump motor control is provided locally with remote stop capability also provided in the SFP Island Control Room.

Electrical and Spent Fuel Pool cooling system parameters and alarms are provided in the SFP Island Control Room.

Heat Tracing

Heat tracing panel HT-SFP-A and its transformer, located in the Spent Fuel Building, is powered from MCC-SFP2 to support SFPI heat tracing loads. A common heat tracing trouble alarm is provided in the SFPI Control Room. The heat tracing on the Spent Fuel Pool Island diesel fuel oil lines and the Decay Heat Removal (DHR) structure drain have local trouble indication.

3.3.6.4 Programmable Logic Controller (PLC)]

The PLC contains a microprocessor based controller which accepts equipment status information, operating parameter data, and instrumentation outputs. Information is provided to the PLC and is compared with set operating limits used in controlling equipment automatic functions. The plant operators monitor the PLC through an Operator Work station and have the capability to operate the plant equipment through the PLC.

The Programmable Logic Controller (PLC) monitors effluents from the Spent Fuel Pool Island (SFPI), operating equipment of the SFPI, and performs some limited control functions of the] equipment. The data is provided to a Operator Workstation in the New Control Room (NCR) via a serial connection from the PLC and to the Local Area Network (LAN) for display and control on workstations. Additional displays are provided in the Technical Support Center (TSC) and any other workstation connected to the LAN. The same data display screen appear on all workstations; however, only the Operator Workstation in the new control room has the capability to provide equipment control.

The physical components of the system are the:

1 The PLC is located in the new control room. The PLC is designed to be able to operate without operator interface. The PLC also communicates with the field devices in this system.]

2 2-I/O cabinets located in the spent fuel pool area. These cabinets contain the system interfaces for input and output signals in the Fuel Building.

3. 2-PowerTrac modules in the SFPI switchgear and 1-PowerTrac module in the BOP Switch Room for monitoring the electrical parameters.

4. Operator Station located in the new control room. This PC is has control and server software which interface with the PLC to provide control and indication of the SFPI equipment and to serve the LAN for display to the other PCS.]

5. 2-Genius I/O modules at the Fire Pump House for inputing status of the fire water system.

6. 3-Genius I/O modules at the New Plant Ventilation Stack Sampling Skid for inputting the PVS system status.]

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Section 3.3 - References

1. USNRC Letter to MYAPC dated March 16, 1999, Amendment No. 162]

2. YNSD Calculation MYC-892, " Spent Fuel Cask Drop," Revision 5, Dated September 5, 1991.]

3. YNSD Calculation MYC-1052, " Spent Fuel Pool Leak Rate Due to Cask Drop," Revision 1, Dated August 20, 1991.]

4. YNSD Calculation MYC-1060, "Spent Fuel Cask Drop Accident," Revision 0, Dated June 7, 1988.]

3.4 Control of Heavy Loads

The controls for the handling of heavy loads at Maine Yankee satisfy the guidelines of NUREG-0612 “Control of Heavy Loads at Nuclear Power Plants” as documented in References 2 through 7 (Phase I). In Reference 9, the NRC concluded that Phase II actions submitted in Reference 8 were not required to reduce risks associated with the handling of heavy loads. While not a requirement, the NRC encouraged the implementation of any actions identified in Phase II regarding the handling of heavy loads considered appropriate. Some of the controls and actions described in References 1 through 9 remain applicable to the control of heavy loads at Maine Yankee with the reactor permanently shutdown and defueled.

The design of load handling equipment at Maine Yankee complies with applicable industry standards and codes. Administrative controls restrict handling of heavy loads in the vicinity of equipment important to the defueled condition. The design of the hoists and cranes coupled with the administrative controls provide assurance that a heavy load will not be dropped which could result in damage to equipment important to the defueled condition.

Load handling equipment that was considered under the NUREG-0612 review that remains applicable to the defueled condition includes:

- fuel building yard crane (CR-3)
- fuel building crane (CR-6)
- mobile platform-hoist (CR-9)

These load handling systems, with the exception of the mobile platform-hoist as described below, were evaluated against the seven guidelines established by the NRC in NUREG-0612. These seven guidelines consisted of:

1. Safe Load Paths
2. Load Handling Procedures
3. Crane Operator Training
4. Special Lifting Devices
5. Lifting Devices (Not Specifically Designed)
6. Cranes (Inspection, Testing, and Maintenance)
7. Crane Design.

The two special lifting devices previously identified by Maine Yankee for the reactor vessel head and the reactor internals are no longer important to the defueled condition. However, the guidelines for special lifting devices are applicable to items such as the spent fuel transfer cask lift yoke and the spent fuel rack lift rig. The conclusions and requirements resulting from the evaluation of the remaining guidelines (1 through 3 and 5 through 7) remain applicable for the load handling systems specified above (CR-9, CR-3 and CR-6), as described in References 2 through 7.

<u>Fuel Building Yard Crane (CR-3)</u> - In 1975 the Commission reviewed Maine Yankee's analysis of a postulated spent fuel cask drop accident in the spent fuel pool and concurred with our evaluation that no safety related equipment was beneath the path for cask travel. In addition, CR-3 has been upgraded in accordance with the single-failure-proof guidelines of NUREG-0612/0554. Spent fuel storage racks and transfer cask may be moved only in accordance with procedures which ensure that they are not moved over fuel assemblies. The travel path of the Fuel Building Yard Crane is now under administrative control instead of mechanical stops and electrical interlocks.

<u>Fuel Building Crane (CR-6)</u> - The travel path of the fuel building crane, the path of the fuel pool cooling pump power cables and the critical volume associated with the fuel pool cooling system are depicted in Reference 4. The fuel pool cooling pump power cables are routed beneath a structural girder and are thus protected from a load drop. The fuel pool cooling system is entirely beneath the new fuel storage area, separated from the fuel building crane by one or more floors. The critical area is clearly marked using safety striping and warning signs and the handling of loads within this area is administratively controlled.

<u>Mobile Platform-hoist (CR-9)</u> - The mobile platform and hoist is used to move fuel assemblies and is operated over the spent fuel pool and on rail extensions to the north of the SFP. This two ton rated hoist generally does not lift loads greater than the weight of a new fuel assembly with handling tools. CR-9 has design features which assure that operation of the mobile platform-hoist can be conducted with very little probability of a load drop. Two upper limit switches of different types prevent two-blocking from occurring. Two load holding brakes of different types, each rated at 150% of the hoisting motor's full load torque, assures that the load can be dependably retained. The platform superstructure was designed to withstand an earthquake of magnitude .10g horizontal and .06 vertical while under rated load without loss of structural integrity or function. The hoist is also seismically qualified. Interlocks prevent trolley movement while the hoist is being operated.

Unless completed within the last eighteen months, a complete checkout of CR-9, including a load test and a functional test of the refueling system interlocks, shall be conducted prior to using CR-9 to handle irradiated fuel assemblies.

Section 3.4 - References:

1. USNRC Letter to MYAPC dated December 22, 1980 Control of Heavy Loads

2. MYAPC Letter to USNRC dated September 18, 1981 (FMY-81-141) Control of Heavy Loads, Section 2.1 submittal.

3. MYAPC Letter to USNRC dated July 7, 1992 (MN-82-131) Control of Heavy Loads

4. MYAPC Letter to USNRC dated August 27, 1982 (MN-82-169) Control of Heavy Loads

5. MYAPC Letter to USNRC dated December 7, 1982 (MN-82-242) Control of Heavy Loads

6. MYAPC Letter to USNRC dated October 6, 1983 (MN-83-221) Control of Heavy Loads, Phase I Report

7. USNRC Letter to MYAPC dated December 30, 1983 Control of Heavy Loads, Safety Evaluation Report.

8. MYAPCO Letter to USNRC dated March 19, 1984 (MN-84-22) Control of Heavy Loads, Phase II Report

9. USNRC Letter to MYAPC dated June 28, 1985 Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612 (Generic Letter 85-11)

SECTION 4.0
RADIATION PROTECTION

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SECTION 4.0
RADIATION PROTECTION

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Zone Description	Maximum Dose Rate mrem/hr	Typical Locations
Periodic Access (Zone II)	2.5	Reactor Containment Charging Floor and above,outside of crane wall and adjacent to CEA change station
Controlled Access (Zone IV) General	>15	Other areas in the reactor containment and inside primary auxiliary building shielded compartments
Periodic Access (Zone III)	<50	At fuel handling hoist platform or the Walkway around the pool
Periodic Access (Zone III)	<80	At the surface of the fuel pool water

Secondary Shielding

Secondary shielding consists of shielding for personnel activities related to fuel handling, auxiliary building activities, control room occupation, and yard activities. Spent fuel residing in the spent fuel pool and activated corrosion and fission products are the radioactive sources for which secondary shielding is required.

Fuel Handling Shielding

The fuel handling shielding, comprising both water and concrete, attenuates radiation from spent fuel assemblies and control element assemblies to design levels, and permits the storage, removal and transfer of spent fuel and control rods within the spent fuel pool in the fuel building. Shielding for fuel handling operations in the spent fuel pool is provided by both the presence of borated water in the pool and the walls of the spent fuel pool structure. The fuel pool wall provides shielding to personnel and is composed of 72 inch thick concrete.

Auxiliary Building Shielding

Auxiliary building shielding is designed to protect personnel in the sample room and in the vicinity of the waste disposal system. Where required, piping is located in shielded pipe trenches. The auxiliary building is compartmented so that equipment areas may be entered without having to shutdown adjacent operating systems or equipment.

4.5 SOLID WASTE TREATMENT

4.5.1 Design Basis

The solid waste treatment system is designed to store, process, monitor and dispose of solid radioactive wastes from the plant. The principal design objective is to insure that the general public is protected from exposure to radioactive waste products in accordance with 10CFR20. The normal sources of radioactive wastes are activated corrosion products and fission products generated during previous plant operation, wastes generated from maintaining spent fuel pool water chemistry, and wastes generated during decommissioning activities.

4.5.2 System Description

4.5.2.1 Filter Handling

The fuel pool system filters are removed from service when the pressure drop across the filters becomes excessive or when the radiation level exceeds a predetermined level. The expended filter cartridges are moved to a shipping cask filter container or to storage in the underground RCA storage bunker. Small, low activity filter cartridges are packaged for ultimate disposal in approved containers. In each case, the procedure conforms to DOT regulations for shipping to an NRC approved disposal site.

4.5.2.2 Solid Wastes

Noncompressible solid wastes such as contaminated metallic materials and highly contaminated solid objects are placed inside approved shipping containers and stored until they are disposed of at an NRC approved disposal site.

4.5.3 Design Evaluation

The solid waste treatment system provides adequate handling and storage capabilities for continued spent fuel pool operation and to support decommissioning activities, as required.

4.6.2.2 Process Monitoring System

The following requirements for radiation process and effluent monitors have been relocated from Technical Specifications by Proposed Change PC-207:

Instrument Operation and Source Checks:

a. Daily* Check: Internal test signals used to check instrument operation. The Liquid Waste Effluent Monitor performs a self-diagnostic check without operator action.

b. Quarterly* Functional Test: Expose the detector with either an internal or an external radiation source or an electronic signal to verify instrument operation.

c. 18 Month Calibration: Exposure to known radiation source.

* When required to be operable

4.6.2.3 Fuel Building Ventilation Exhaust Radiation Monitoring Skid

This equipment takes a continuous isokinetic air sample from an exhaust duct on the PAB roof common to the FB & RCA and, using a vacuum pump, draws it through a sealed system to particulate cartridge, a continuous gas monitor, a tritium collector system, and a automatic flow control as identified below. The sample is then discharged inside the Fuel Building.

The fuel building process radiation monitoring system provides early warning of a plant malfunction; warns personnel of increasing radiation which could result in a radiation health hazard; channels include locally operated check sources, readout and alarm in the spent fuel pool control room and are recorded by the PLC-SFP-01. The PLC indication consists of a log count rate meter and an alarm. All channels also include readout and alarm at the detector station.

Particulate Cartridge

A cartridge assembly, which collects a particulate sample from the isokinetic sample flow, is periodically removed and checked for activity.

Continuous Gas Monitor

A beta scintillation detector mounted within a fixed volume shielded chamber continuously measures the ventilation exhaust gas sample activity. This measurement is used in off-site dose projections performed per the ODCM.

SECTION 5.0
ACCIDENT ANALYSIS

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SECTION 5.0
ACCIDENT ANALYSIS

5.1 Introduction

Earlier sections of this report describe the major systems and components of the plant from the perspective of safe spent fuel handling, spent fuel storage, and other decommissioning activities as would be appropriate to a permanently defueled plant.

This section uses the previous information and examines the potential consequences of accidents and incidents, notwithstanding the precautions taken to prevent their happening, to assess the adequacy of the plant design in minimizing or mitigating potential consequences of such occurrences. Additionally, the accident analyses presented in this section provide assurance that the health and safety of the public is protected from the consequences of even the most severe of the hypothetical incidents analyzed.

With the permanent defueling of the Maine Yankee facility and the certification of the cessation of operations, the postulated accidents associated with reactor operation are no longer applicable and need not be considered. Likewise, the unirradiated nuclear fuel has been removed from the Maine Yankee site and therefore accidents involving new fuel assemblies are also no longer applicable. However, those accidents associated with the storage or handling of irradiated fuel or radioactive waste storage or processing remain applicable and are discussed within this section.

The general classification of accidents for the permanently defueled condition are limited. These groupings are listed as follows:

1. Inadvertent criticality of the stored spent fuel,
2. Fuel assembly handling accident,
3. Spent fuel cask drop in the spent fuel pool,]
4. Loss of spent fuel decay heat removal capability,
5. Loss of spent fuel pool inventory,
6. Radioactive release from a subsystem or component, or
7. Low level waste storage accident.

The evaluation of these accidents is based on a conservative set of initial conditions and analytical methodologies. The key selected initial conditions are from worst case operating conditions

5.3 Fuel Handling Accident

The purpose of this section is to assess anticipated spent fuel pool fuel handling operations in order to arrive at the accident which would result in appropriately conservative off-site and control room radioactive release effects. Fuel handling incidents originating in the containment are not applicable to the permanently defueled condition. Fuel handling operation associated with the use of a spent fuel cask are addressed in section 5.4.

The likelihood of a fuel handling incident in the spent fuel pool is minimized by implementation of appropriate and long standing administrative controls and physical limitations imposed on fuel handling operations. All fuel handling operations are conducted in accordance with prescribed procedures under the direct surveillance and supervision of qualified personnel.

The fuel handling equipment and facility are designed for the transfer and handling of a single fuel assembly at any time, and movement of equipment when handling the fuel is administratively restricted. The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. In the spent fuel pool, the design of fuel storage racks and manipulator equipment, in conjunction with appropriate administrative controls, is such that:

1. Fuel is always maintained by mechanical restraint. Fuel at rest is positioned by positive restraints in a subcritical geometrical array, with no credit for boric acid in the water.
2. Fuel can be manipulated only one assembly at a time.
3. Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.
4. The spent fuel transfer cask does not pass over spent fuel during transfer operations.]

The fuel assembly is immersed continuously while in the spent fuel pool. Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel handling equipment and spent fuel pool are described in detail in Section 3.

Inadvertent disengagement of the fuel assembly from the fuel handling equipment is prevented by design, mechanical, and procedural interlocks. Consequently, the possibility of dropping and damaging a fuel assembly is unlikely. The combination of these safeguards make the probability of a fuel handling incident very low.

5.4 Spent Fuel Cask Drop

Spent fuel casks will be handled with the fuel building yard crane (CR-3). CR-3 has been upgraded in accordance with the single-failure-proof guidelines of NUREG-0612 and NUREG-0554. Therefore, in accordance with NUREG-0612, an accident analysis of a spent fuel cask drop is not required and does not supply safety analysis limits.]

5.6.3 Spent Resin Liner Drop Accident]

Although no longer in use, the Low Level Waste Storage Building (LLWSB) provided the scene for the bounding accident in the radiological impact analysis. The accident is defined as the dropping of a highly loaded spent resin liner within the LLWSB, resulting in liner failure, spillage of spent resin and the release of a fraction of the radioisotopic contents in a cloud. The contents of this cloud form the basis for determining the radiological source term at the site boundary. The key analysis assumptions for this accident are as follows:]]]]]]

1. The liner is loaded with spent resin at the allowable Department of Transportation Low Specific Activity (LSA) limits per 40 CFR 173.403, except for the isotopes of I^{129} , I^{131}, and Co^{60}.
2. The amount of I^{129} is assumed to be equivalent to 0.08 Curies per cubic meter, which is in excess of the limits allowed by 10 CFR 61 LLW burial criteria.
3. The amount of I^{131} is assumed to be 800 Curies in the liner.
4. The amount of Co^{60} is assumed to be 1,627 Curies in the liner.
5. The liner contains 148 cubic feet of spent dewatered resin at 40 lbs/cubic foot.]

The spent resin liners are stored in High Integrity Containers (HICs) which are designed to survive a 20 foot drop while fully loaded. For purposes of this analysis, it was assumed that the loaded HIC falls and breaks open, thus spilling the spent resin. It was assumed that 1% of the activity of the liner non-mechanistically forms an aerosol and that 10% of the aerosol is non-mechanistically released outside the LLWSB. This aerosol acts as a "puff" release in assessing the potential doses at the EAB. The activity of the release was assumed to be dispersed over an arc of 225 degrees at 700 meters from the LLWSB.]]]]

Other initiating events were examined for potential radiological release pathways. Assessments of seismic events, tornados, floods, and hydrogen gas generation and explosions on the potential to yield a more severe radiological release have not identified a more severe case than the dropping of a loaded HIC.

The calculation of the doses at the EAB was performed in accordance with the NRC Regulatory Guide 1.109, Appendix C, using the code ATMODOS. The results of this calculation for a two hour period at the EAB are as follows:

1.	Maximum Organ Dose (Teen Lung)	262 mrem
2.	Thyroid Dose	70.52 mrem
3.	Whole Body Dose	7.08 mrem

The principal radionuclide contributors to the maximum organ doses are:

Nuclide	Dose (mrem)
Co^{60}	76.6
Ce^{144}	58.0
Ag^{110}	29.3
Sb^{124}	16.7
Sb^{125}	11.9

Based on these dose calculations, the doses for the bounding accident, i.e., the dropping of a fully loaded HIC with spent resins at or above regulatory limits, are below small fractions of the values in the applicable regulation, 10 CFR 100, "Reactor Site Criteria" and the EPA Protective Action Guidelines.]

SECTION 6.0
CONDUCT OF OPERATIONS

Maine Yankee Atomic Power Company is responsible for all aspects of plant operation, including maintaining the plant in the permanently defueled condition through termination of the existing Operating License. As a portion of this responsibility, Maine Yankee is also responsible for employing and training qualified personnel to operate the plant.

Technical or managerial assistance in the satisfaction of this responsibility may be provided through a number of external resources. Such assistance is provided by Entergy Nuclear, Inc, Stone & Webster Engineering Corporation, ABB-Combustion Engineering, Inc., Westinghouse Electric Company, and Duke Engineering & Services (formerly Yankee Atomic Electric Company, Nuclear Services Division). Other support, as required, may be arranged from time to time.

6.1 Responsibility and Organization

The functional organization and key lines of responsibility for the Maine Yankee plant staff are shown in Figure 6.1-1.

The on-site organization includes the technically trained personnel necessary to support all aspects of plant operation. Each member of the facility staff meets or exceeds the minimum qualifications of Regulatory Guide 1.8, dated September 1975.

6.1.1 Duties and Responsibilities of the Operating Staff Personnel

Director Operations

The Director Operations directs the daily plant activities and is responsible for the overall safe and efficient operation of the plant.

Additionally, the Director Operations is responsible for the compliance of operations with the requirements of the facility license and Technical Specifications. This responsibility includes oversight of the Operations/Maintenance and Radiation Protection Departments and the security and training functions. The Director Operations may delegate these responsibilities to the appropriate Manager(s) in his absence. The Director Operations will assume responsibilities of the generic title "Plant Manager" as defined in ANSI N18.7-1976.

The Director Operations reports directly to the Vice President & Chief Nuclear Officer.

Manager, Operations /Maintenance

The Manager, Operations/Maintenance directs the activities of the Operations Department including the operating shifts, personnel in training, and staff.

The Manager, Operations/Maintenance is responsible for coordinating the activities of the Operations Department and plant operation, with other plant functions. Additionally, the Manager, Operations/Maintenance is also responsible for maintaining plant operation records as required by the facility license and Technical Specifications. The Manager, Operations/Maintenance is responsible for the training and qualifications of the plant operations personnel.

The Manager, Operations/Maintenance directs and coordinates the scheduling and supervision of mechanical, electrical, and instrument and control maintenance work with other plant functions in the handling of routine and non-routine maintenance assignments.

The Manager, Operations/Maintenance is responsible for routine inspections, preventative maintenance practices and record keeping as required by the facility license and Maine Yankee policies and practices. Additionally, the Manager, Operations/Maintenance is responsible for the training and qualifications of maintenance personnel.

Unless otherwise designated, the Manager, Operations/Maintenance is assigned the responsibility of the ISFSI Manager.

The Manager, Operations/Maintenance reports to the Director Operations.

Shift Manager

The Shift Manager is the Senior Management Person onsite during backshifts and weekends.

Unless otherwise designated, the Shift Manager is assigned the responsibility of the ISFSI Shift Supervisor.

The Shift Manager Reports to the Manager, Operations/Maintenance

Radiation Protection/Rad Waste Manager

The Radiation Protection/Rad Waste Manager directs and coordinates the activities of the radiological controls and radiological program activities, including overall responsibility of implementation of the ALARA Program, at Maine Yankee.

As such the Radiation Protection/Rad Waste Manager is responsible for storage and handling of radioactive material, and radiation safety. Additionally, the Radiation Protection/Rad Waste Manager is responsible for the training and qualifications of radiation protection personnel.

Unless otherwise designated, the Radiation Protection/Rad Waste Manager is responsible for the ISFSI RP functional requirements.

The Radiation Protection/Rad Waste Manager reports to the Director Operations

6.1.2 Duties and Responsibilities of the Support Staff

President

The President shall have corporate responsibility for Plant nuclear safety and shall ensure acceptable performance of the staff in operation, maintaining, and providing technical support to ensure the safe storage of irradiated fuel.

Vice President & Chief Financial Officer

The Vice President & Chief Financial Officer has oversight responsibility for the development, maintenance, and implementation of policies, practices, and procedures addressing Budgets, Cash Management, Trusts, Information Technology, Accounting, and Administrative Support.

The Vice President & Chief Financial Officer reports to the President.

Vice President & Chief Nuclear Officer

The Vice President & Chief Nuclear Officer has oversight responsibility for the development, maintenance, and implementation of policies, practices, and procedures addressing safe storage of spent fuel, plant operations, compliance with applicable Federal, State, and Local laws & regulations; Quality Programs; and Public Affairs.

The Vice President & Chief Nuclear Officer reports directly to the President.

Nuclear Safety & Regulatory Affairs Director

The Nuclear Safety & Regulatory Affairs Director has overall responsibility for activities required to maintain the permits and licenses required for the plant including production, maintenance, and interpretation of licensing documents. This includes oversight of the plant's design/licensing basis, compliance with applicable federal, state and local laws and regulations, and the company's interaction with state and federal technical regulatory agencies, including Emergency Preparedness. Additionally, the Nuclear Safety and Regulatory Affairs Director is responsible for ensuring implementation of the Quality Assurance Program, Corrective Action Program, and Worker Concerns Program.

The Nuclear Safety and Regulatory Affairs Director shall provide an independent overview of ISFSI operation through the Quality Programs Department.

The Nuclear Safety & Regulatory Affairs Director reports to the Vice-President & Chief Nuclear Officer.

Quality Programs Department Manager

The Quality Programs Department (QPD) Manager is responsible for providing quality control coverage of maintenance and modification activities and developing and maintaining the Quality Assurance Program.

The QPD Manager has the authority and independence to identify quality problems; initiate, recommend, or provide solutions to quality problems through designated channels; and verify implementation of solutions to quality problems. The QPD Manager also has the authority and responsibility to initiate stop work orders to responsible management, as necessary, for any condition adverse to quality.

Additionally, the QPD Manager has overall responsibility for development, maintenance, and implementation of the Corrective Action Program.

Unless otherwise designated, the QPD Manager is responsible for the ISFSI QA functional requirements.

The QPD Manager reports directly to the Nuclear Safety & Regulatory Affairs Director. Additionally, the QPD Manager is a functional report to the President.

Vice President of Decommissioning

The Vice President of Decommissioning is responsible for developing and implementing projects and programs necessary to ensure the safe and economical decommissioning of the plant.

The Vice President of Decommissioning reports to the President.

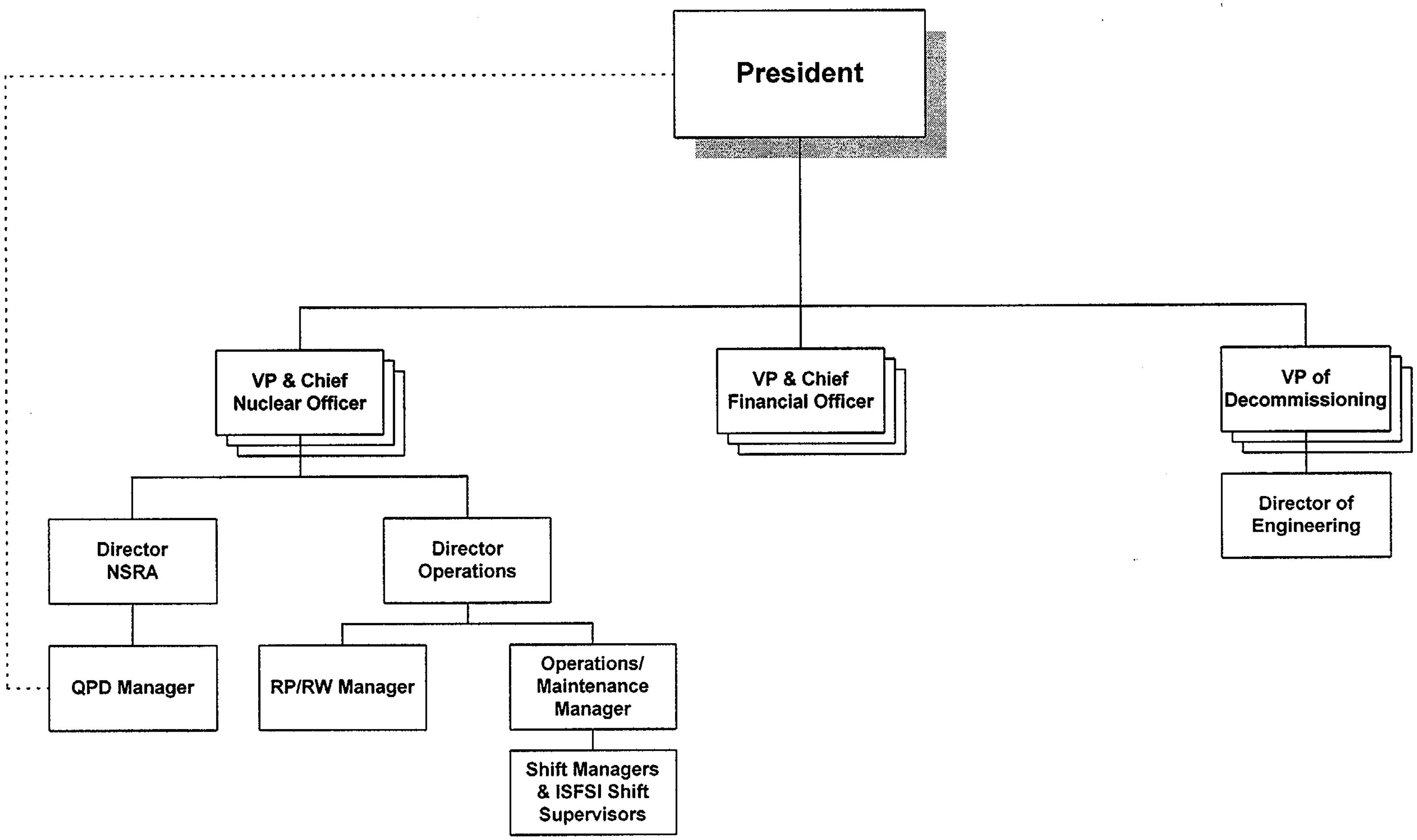
Director of Engineering

The Director of Engineering is responsible for providing engineering assistance to the plant operations and support staffs, initiation and implementation of key plant modifications, maintenance of the plant design basis, design configuration data and documentation.

Unless otherwise designated, the Director of Engineering is assigned the responsibility of the ISFSI Engineer.

The Director of Engineering reports to the Vice President of Decommissioning.

MYAPC
Facility Organization



Maine Yankee will maintain and protect systems and areas critical to the storage of spent fuel.

7.1.2.2 Low Level Waste

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

7.1.3 Radiation Exposure Monitoring

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

For criticality monitoring during spent fuel loading operations, the requirements of 10 CFR 50.68 and 72.124(c) shall be utilized in lieu of the requirements of 10 CFR 70.24.]

APPENDIX A
METEOROLOGICAL DATA SUMMARIES

A.1 **Initial On-site Meteorological Program**

A.1.1 Introduction

The initial data collection program was undertaken at the site of the Maine Yankee Atomic Power Station to provide information from which to base meteorological conditions for accident analyses in the original Final Safety Analysis Report (FSAR). Data for one year, from July 1, 1967 to June 30, 1968, was evaluated from that program and used for dispersion analysis in the original FSAR. This section contains an analysis of the July 1967 - June 1968 data collected from the initial monitoring program.

A.1.2 Description of the Monitoring Program

A 149-foot meteorological tower was installed at Bailey Point at an elevation of approximately 45 feet. The instrumentation consisted of a recording "Aerovane" wind system at the 149-foot tower elevation to record wind speed, wind direction and directional variability, and three temperature sensors at the 5, 73 and 143-foot tower elevations.

Information regarding missing and valid data for the period July 1967 and June 1968 is shown in the following table.

Month	No. Of Missing Observations	No of Valid Observations
July 1967	96	627
August	19	725
September	0	689
October	0	667
November	13	705
December	1	740
January 1968	101	643
February	95	601
March	94	650
April	141	569
May	61	682
June 1968	56	662
TOTALS	677	7,960

MYAPC

The longest continuous period of missing data lasted 116 hours and occurred at the end of March and beginning of April 1968.

Each observation represents an average of the parameter of interest (wind speed, wind direction or temperature difference) of the thirty-minute period preceding each hour. A missing observation refers to lack of either wind speed or direction data and a valid observation includes wind speed, wind direction and temperature difference. The stability classes, as defined below, were determined by the temperature difference between 5 and 143 feet.

The following data was collected on a monthly, seasonal and annual basis:

1. Joint frequency distributions of wind speed and direction for each stability classification and for all classes combined.

Wind speed ranges (mph):	calm, 1-3, 4-7, 8-12, 13-18, 19-24, 25-31
Wind directions:	16 point compass
Stability Classes:	
Moderately Stable	$\Delta T>1.5°C/100$ m
Slightly Stable	$-0.5\ \Delta T<1.5$
Neutral	$-1.5\ \Delta T<-0.5$
Unstable	$\Delta T<-1.5$

2. Percent occurrence of each stability classification.

3. Inversion persistence with maximum duration and percent of occurrence.

4. Wind direction persistence summary (22-1/2° and 67-1/2° sectors) for all observations.

5. Frequency distribution, of $\sigma_\theta\acute{u}$, for each stability class and wind speed direction summarized according to the wind speed ranges listed above.

In addition to the above data, the most persistent monthly inversions and wind speed and direction during these periods, and an annual precipitation/no precipitation summary have been tabulated.

A.2 Current On-site Meteorological Program

A.2.1 Introduction

The on-site meteorological data collection program was upgraded in late 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. This report describes the on-site monitoring program applied during plant operation and presents wind and stability data summaries for one full year of operation; January 1, 1979 through December 31, 1979. A discussion of the data summaries is included, and a comparison is made between the initial (July 1967 - June 1968) and the current (January 1979 - December 1979) data bases. It is concluded that results from both programs are compatible, and that both programs produced data bases which are representative of site meteorology.

A.2.2 Description of the Monitoring Program

The meteorological monitoring system utilized a guyed 200-foot tower located on-site as shown in Figure A.2-1. Instrumentation on the tower was located on booms at the 33-foot and 195-foot levels. Wind measurements were observed at heights of 35 feet and 197 feet above the tower base (two feet above both booms). Both wind speed and wind direction were measured at each height. Ambient temperature difference was measured on the tower between 32 feet and 194 feet (one foot below both booms). Ambient temperature was also measured by this system for the 32-foot level.

A digital recording system was the primary data collection mechanism for the Maine Yankee Meteorological System. The data collection mechanism was provided by the Control Room PLC.

A.2.3 Results

The data base used to compile the 1979 data summaries which follow represents hourly averaged data digitized from strip charts (the 1979 data base was collected before the plant computer began collecting data digitally). The 1979 data recovery rates, which are well above the Regulatory Guide 1.23 goal of 90 percent, are presented in Table A.2.1.

DSAR – APPENDIX (B)

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SECTION 1.0

INTRODUCTION AND SUMMARY

1.1 Introduction

The Maine Yankee nuclear power plant is a single unit pressurized water reactor located in Wiscasset, Maine. The plant began operation in December 1972 and permanently shut down in August 1997. The plant last operated on December 6, 1996. By June 20, 1997, the reactor had been completely defueled and all spent fuel resided in the spent fuel pool. On August 6, 1997, the Maine Yankee Atomic Power Company Board of Directors voted to permanently cease power operations and initiate the decommissioning process. On August 7, 1997, Maine Yankee provided written certification to the Nuclear Regulatory. Commission, pursuant to 10 CFR 50.82 (a)(1)(i) and (ii), that the Maine Yankee Atomic Power Company permanently ceased operations of the Maine Yankee Atomic Power Station and that all nuclear fuel had been permanently removed from the reactor (Ref. 1).

The issuance of this certification fundamentally changed the licensing basis of the Maine Yankee plant in that the NRC-issued 10 CFR 50 (Ref. 2) license no longer authorized operation of the reactor or emplacement or retention of fuel in the reactor vessel. Therefore, as of August 7, 1997, only those conditions or activities associated with the safe storage of fuel and radiological protection (including waste handling, storage, and disposal) are applicable to the defueled Maine Yankee plant.

The Maine Yankee ISFSI is being licensed under the "general" license approach in accordance with Subpart K of 10 CFR Part 72 (Ref.20). The ISFSI general license allows a 10 CFR Part 50 power plant licensee to use a cask storage system that is pre-approved by the NRC, provided that all requirements of the cask vendor's license are complied with. The cask storage system selected for use at Maine Yankee is the Universal MPC System (UMS™), which is designed and licensed by NAC International (NAC). To use this system under the general license approach, the ISFSI licensee must comply with the requirements contained in NAC UMS™ Safety Analysis Report (SAR) (Ref. 10) and the NRC issued Certificate of Compliance to the NAC UMS™ system.

The Independent Spent Fuel Storage Installation (ISFSI) provides safe, economic, and long-term storage of used nuclear fuel and facilitates the decommissioning and dismantlement of the existing Maine Yankee nuclear power plant. The ISFSI provides onsite dry cask storage for the total inventory of used nuclear fuel that is currently stored in the Maine Yankee spent fuel pool. The ISFSI also stores the Greater-Than-Class C (GTCC) waste, which consists of the irradiated reactor internals. Placing the used fuel and GTCC waste in dry storage allows the fuel building, including the spent fuel pool, to be decommissioned and dismantled in the same manner as all other plant structures. The used fuel and GTCC waste will be stored safely in the spent fuel ISFSI storage system, consisting of sealed (airtight) metal canisters placed inside of concrete casks.

The ISFSI will remain operational until the US Department of Energy (DOE) or another licensed entity has temporary storage or a permanent high-level waste disposal facility to accept the Maine Yankee spent fuel and GTCC waste, or until another suitable disposal facility becomes available.

Construction of the ISFSI facility, was started in November 1999 and is scheduled for completion by June of 2001. Spent fuel and GTCC material transfer to ISFSI is scheduled for March 2001 through to November of 2002.

1.2 General Description of ISFSI Facility

1.2.1 General

The ISFSI is located within the site boundary, of the Maine Yankee nuclear power plant in the Town of Wiscasset, Lincoln County, Maine. The site selected for the ISFSI is an open area approximately 1200-ft north of the power plant that was being used as a vehicle parking lot. The ISFSI is located outside of the plant exclusion area but within the owner-controlled area. There are no permanent occupants that reside in the owner-controlled area. The ISFSI occupies a land area of approximately 10 acres.

1.2.2 Principal Characteristics of Site

The Maine Yankee nuclear power plant layout and the ISFSI site location are shown on Figure 1.2-1. Spent fuel will be loaded into canisters inside the Fuel Building. The casks are then moved to the adjacent RCA Building for draining, drying, welding and helium backfilling activities. The canisters are then placed into storage casks that are hauled from the fuel building area to the ISFSI using a custom heavy haul tractor-trailer. The haul route is shown on Figure 1.2-2.

The ISFSI consists of the storage system and concrete storage pads, a Protected Area (PA) for spent fuel storage, and a Security/Operations Building for equipment and staff. The PA is surrounded by a 12-ft. high security fence and encompasses an area of approximately 3 acres. Additional security is provided by a second 8-ft high nuisance fence outside the security fence with a 20-ft wide isolation zone between the two fences. The PA contains 16 concrete cask storage pads (each being approximately 31-ft wide by 31-ft long) with driveway access around the pads. Each pad holds up to four spent fuel or GTCC concrete storage casks. A partial earthen berm is provided around the facility to reduce the visual impact of the facility.

A short access road (approximately 175-ft long) connects the PA with an existing plant road. Adjacent to the access road are a group of visitor parking spaces on the south side and a vertical concrete cask (VCC) fabrication pad on the north side.

The ISFSI site plan is shown on Figure 1.2-3. The ISFSI site grading is shown on Figure 1.2-4.

The facility is being designed and sized to be only large enough for the existing licensed spent fuel and GTCC at Maine Yankee. Maine Yankee will not store any spent fuel or GTCC from other generators.

The planned storage capacity of the ISFSI will utilize a total of 64 concrete storage casks, with one sealed canister per storage cask. The types of canisters stored in the storage casks include 60 for spent fuel and four for GTCC waste. Damaged fuel will be stored in special cans having mesh screens, in the same canisters with intact fuel. Sixteen storage pads of equal size are provided for uniformity. The storage pad details are shown on Figure 1.2-6. The ISFSI site is arranged to provide maneuvering room around the storage pads for access to each cask with the heavy haul tractor-trailer.

The security/operations building is approximately 10,500 square feet in area approximately (68-ft x 154-ft) and approximately 40-ft high. A ground floor plan of the security/operations building is shown on Figure 1.2-7.

1.2.3 ISFSI Facility Principal Design Features

1.2.3.1 ISFSI Site

The ISFSI facility principal design features include the design codes, standards, and design requirements applicable to the site, earthen berm, cask storage pads, and the Security/Operations Building.

In May of 1999 Maine Yankee submitted to the Maine Department of Environmental Protection (MDEP) a major amendment application (Ref. 3) to the existing MY Site Location of Development Permit. The major amendment application was to allow construction of an ISFSI for the interim storage of Maine Yankee's spent fuel and GTCC. The application, pursuant to State laws, provides detail design information relative to State environmental compliance in areas, such as, solid wastes generated by the development, anticipated air emissions and noise impact during development, assessment of site soils for the ISFSI, ISFSI impact on groundwater, and stormwater management during site development and ISFSI operation.

The ISFSI site area is above the 100-year flood plain, as shown on the "Flood plain Map" prepared for the Maine Yankee Land Use by Robert G. Gerber, Inc. and dated February 6, 1997 (Ref. 4).

The ISFSI site area is also above the Probable Maximum Flood (PMF) elevation as described in Section 2.3.3 of the DSAR, and as such, flooding is not a consideration in design of the facility.

Snow Load - A design ground snow load of 60 psf per ASCE-7 (Figure 7-1) (Ref. 5) has been considered for the site. Snow load data for the Bailey Point area is identified in the site Defueled Safety Analysis Report (Section 2.2.5.1) as 40 psf (10- yr storm), 60 psf (max. recorded), and 80 psf (Estimated max. accumulation plus weight of max. possible storm).

Wind Loads - Site calculated wind loads have been determined in accordance with ASCE-7 using a basic wind speed of 105 mph (Figure 6-1 and linear interpolation), Occupancy Classification III.

Temperature Range - Temperature extremes at the Maine Yankee site are also described in the Defueled Safety Analysis Report (Section 2.2.4 and Table 2.2.2) as approximately -40°F to +100°F. The average annual temperature is approximately 45°F.

Tornado Design Criteria - The tornado design criterion for the plant site is described in the site DSAR, Section 3.1.2.2. The design basis tornado for the vertical concrete casks (VCC's) is based on Regulatory Guide 1.76 (Ref. 25) and NUREG - 0800 (Ref. 26).

Seismic Design -The Maine Yankee ISFSI site seismic analysis is based on a 0.18g-NUREG/GR-0098 ground response spectrum for the plant site and on an evaluation of the effects associated from appropriate soil amplification and soil-structure interaction (Ref. 27 & 28).

Load combinations, allowable stresses, and factors of safety, are in accordance with NUREG-1567 (Ref. 6), ANSI/ANS-57.9 (Ref. 7), ACI-349 (Ref. 8), ACI-318 (Ref. 11), and/or ASCE-7, as applicable to the design element under consideration.

1.2.3.2 ISFSI Berm

The ISFSI is provided with an earthen berm on three sides to minimize the visual impact of the facility. The earthen berm is classified as a QA Category III structure since it is not used for shielding credit in the site dose assessment. The seismic design of the berm conforms to BOCA NBC Section 1610 (Ref. 9). The seismic design need not necessarily qualify that the berm remains intact after a Design Basis Earthquake (DBE). After such an event, repair of the berm could be performed in a short period of time. However, the berm arrangement is such that a failure of the berm does not have any effect on the storage casks or pads. Design and construction of the berm follow QA Category III requirements.

1.2.3.3 ISFSI Storage Pad

The cask storage pads are provided to support the storage casks that contain the sealed metal spent fuel canisters. The storage technology selected for use at Maine Yankee is the UMS™ storage system designed by NAC International (NAC).

A design for a light water reactor spent fuel dry cask storage was submitted to the U.S. Nuclear Regulatory Commission (NRC) for licensing approval in the NAC Topical Safety Analysis Report (Ref. 10).

The cask storage pads are capable of adequately supporting the storage casks/canisters under all load conditions. The cask storage pads are designed and constructed to assure the pad/soil system is "soft" enough to limit the deceleration forces encountered by the canister during the hypothetical cask tip over event. As such, the cask storage pads were classified as QA Category II which provided additional quality assurance and quality control requirements during design and construction of the pads to ensure compliance with the NAC SAR requirements. Pad construction documents (drawings and specifications) were marked as QA Category II, with specific QA/QC requirements identified therein. Pad calculations were prepared in accordance with QA Category I requirements. All geotechnical field investigation work, laboratory testing of soil samples, and geotechnical calculations were classified as QA Category I to assure a conservative approach to data gathering and establishment of the design bases.

The critical attributes identified by the NAC UMS™ Certificate of Compliance and associated technical specifications (e.g., concrete strength, thickness, surface roughness, subgrade soil properties, etc.) are identified on the construction drawings and specifications. The compressive strength for the concrete meets the specified 3,000 psi at 28-days nominal requirements.

The quality attributes described above are based upon the designer's (S&W) quality program definitions. The pads were classified as ITDC/NSS in accordance with the Maine Yankee QA program during design and construction to assure that the required engineering attributes were verified. The storage pads are presently classified as "Not Important to Safety"(NITS) under the MY ISFSI QA Manual classification system.

The storage pad thickness is a maximum of 3 feet with the top of concrete set approximately 2'-6" above grade to accommodate cask transfer. The existing soil beneath the pads has been replaced with compacted structural bedding material (non-frost susceptible) to a depth of 4'-6" with a tolerance of (+6"/-6").

NAC has identified the size and weight of the storage system components to be used in the storage pad design in the NAC Safety Analysis Reports (Section 3.2). The storage pad design considers the different load combinations associated with the progressive placement of storage casks.

The cask storage pads are designed in accordance with ANSI/ANS 57.9 and ACI-349. Computer methods were used to perform the static and dynamic (seismic) analysis of the storage pads. Soil-structure interaction was considered in the computer model and follows recommendations identified in ASCE-4 (Ref. 12).

1.2.3.4 Security/Operations Building

The Security/Operations building is a steel framed metal sided building with a 20-ton bridge crane running the full length of the building. The building is approximately 68 feet wide by 154 feet long and approximately 48 feet high. The ground floor, pile-supported, concrete slabs have an original floor design load of 1800 psf. The Security/Operations building provides the functional requirements for the ISFSI, including security, maintenance, and spare parts storage.

The design codes and standards applicable to the design of this facility include the BOCA National Building Code, NFPA 101 (Ref. 13) and security requirements contained in NUREG 0508 (Ref. 14).

The Security/Operations Building houses the ISFSI security staff, security equipment, communications equipment, and provides site access control and issuance of dosimetry for persons entering the Protected Area. The Security/Operations Building also provides offices and work space for the operating and maintenance personnel, including a health physics area, lunch/conference room, restrooms, document control room, truck access bay, spare parts storage area, and a diesel generator for emergency power.

Site Access is controlled from the Security/Operations Building with provisions for issuing security badges and dosimetry for personnel entering the ISFSI Protected Area.

Fire Protection - Fire protection is provided in the Security/Operations Building in accordance with the requirements of the BOCA National Building Code (NBC). Fire protection systems were reviewed and approved by the insurance underwriter of the Maine Yankee nuclear power plant.

Fire protection consists of a clean agent fire extinguishing system in the document control room, installed in accordance with NFPA 2001 (Ref. 15), and portable fire extinguishers installed in accordance with the requirements of NFPA 10 (Ref. 16), UL-299 (Ref. 17), and UL-154 (Ref. 18).

Fire detection consists of smoke detectors with local alarms. A new fire hydrant has been installed outside of the building to provide protection from fires for the building. All fire protection equipment is UL or FM approved for use in fire extinguishing systems and fire protection systems. Design of piping, connections, and pipe supports are in accordance with the requirements of applicable NFPA standards.

Emergency Power - Emergency backup power is provided for the ISFSI security system and consists of an Uninterruptable Power Supply (UPS-batteries) and a diesel generator. The diesel generator is sized to support the required security loads and emergency exit lighting in the Security/Operations Building. The UPS supports the security loads until the diesel starts and comes up to speed.

The generator is sized to power the required portions of the ISFSI security system and all of the facility emergency lights. The generator is sized for a minimum of 25% spare capacity above the anticipated load per IEEE 692 (Ref. 19), Section 4.3.4. This system is designed to operate for a minimum of 24 hours continuously in accordance with IEEE 692, Section 4.3.2.

An automatic transfer switch is supplied with the diesel generator unit. This switch transfers the security and emergency loads to the generator once the generator comes up to speed. The transfer switch is supplied with voltage sensing and timing devices to facilitate the transfer to and from the generator source.

1.2.3.5 Principal Environmental Design Parameters

The principal environmental design criteria and parameters for the ISFSI are shown in Table 1.2-1. As previously mentioned, the ISFSI is designed to store Maine Yankee generated spent fuel and GTCC radioactive waste.

1.2.4 Facility Operation Waste Products Generated

It is anticipated that only minimal solid waste will be generated during operation of the ISFSI facility. There are no gaseous or liquid wastes produced under normal operation. Most of the solid waste generated will be associated with the operation of the Security/Operations facility. Waste generated will be handled through licensed waste processing or disposal facilities.

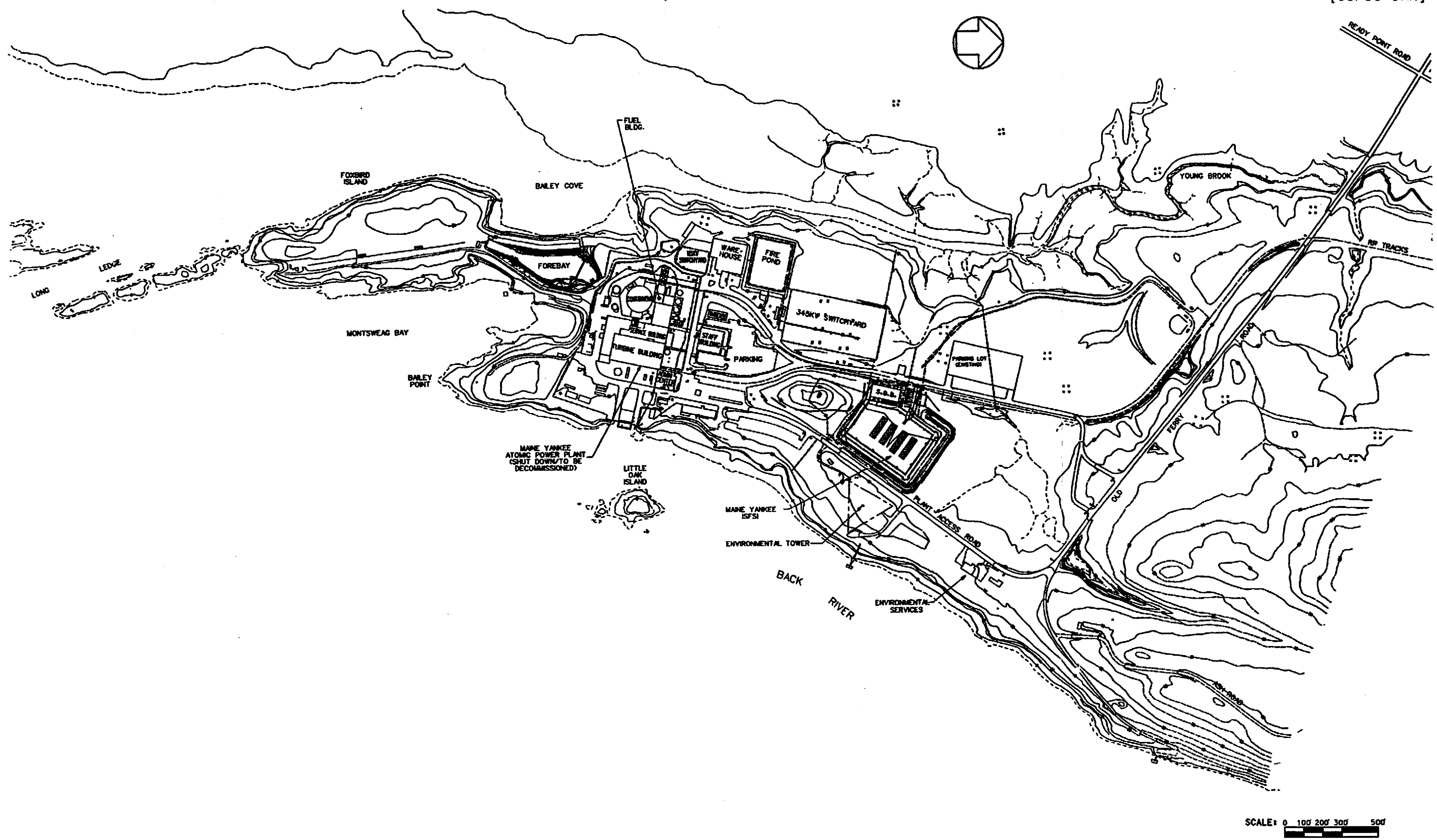
Table 1.2-1

Maine Yankee Principal Environmental ISFSI Design Parameters

Parameter	Value
Tornado & Wind Loads	Regulatory Guide 1.76 (Ref. 25) and NUREG-0800 (Ref. 26)
Ambient Temperatures	- 40 degrees F to 106 degrees F (133 degrees F for Accident – Extreme Heat)
Flood Level	El. 11 feet for "100 year flood" El. 19.9 feet for "maximum site water level due to probable maximum hurricane"
Design Basis Earthquake	0.26g for NAC Cask Design 0.18g for Cask Storage Pad Design(Note 1)
Snow and Ice Loads	100 psf for NAC Cask 60 psf for ISFSI Site

Note (1): The site ground response (0.18g NUREG/CR-0098 Median Spectra) must be modified to account for local soil and soil-structure interaction concerns (Ref. 28).

DSAR-APPENDIX (B)
[ISFSI-SAR]



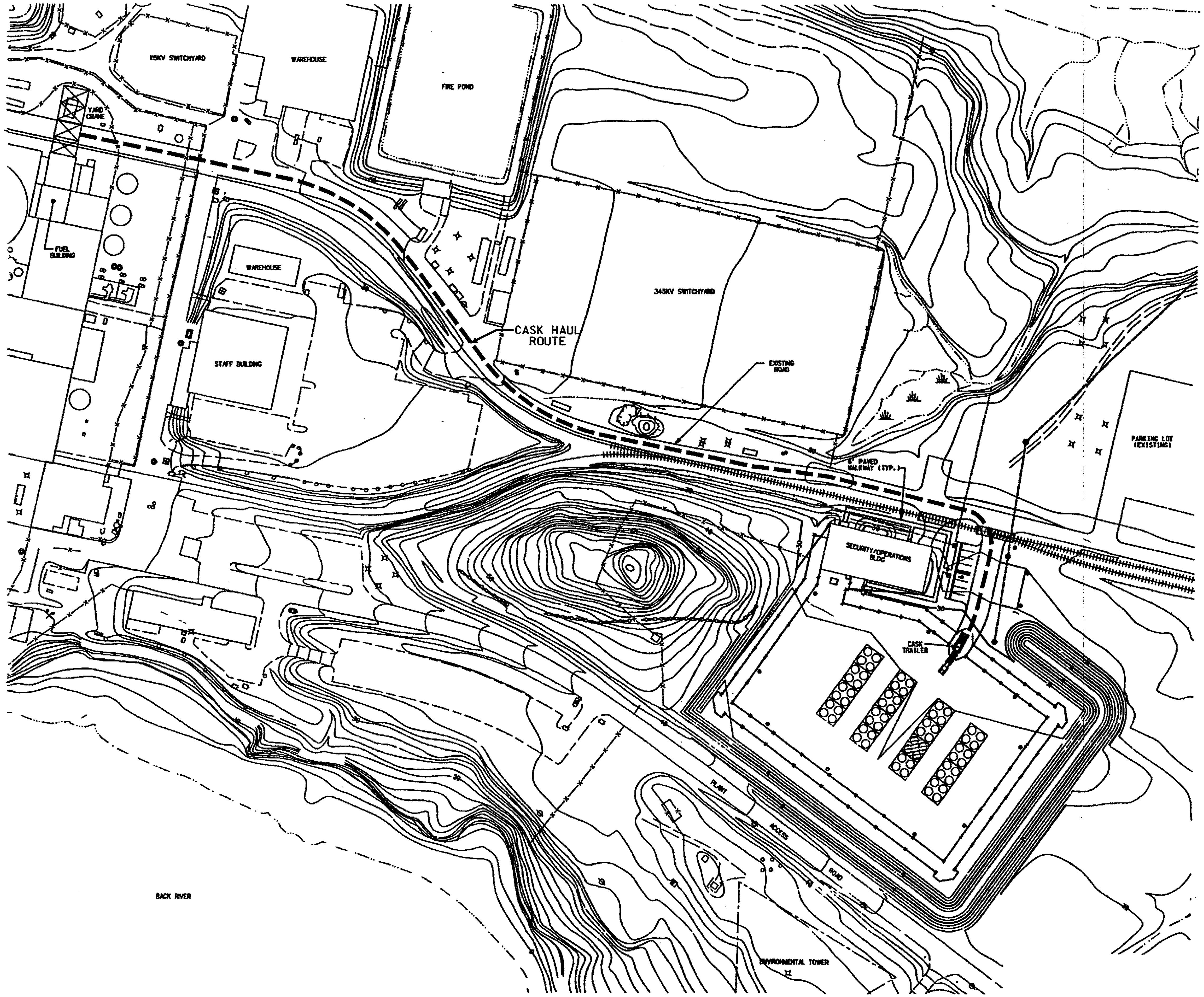
REV.	DATE	BY	CHKD	APPD
0	05/23/01	MMM		

ISFSI SITE LOCATION PLAN

Maine Yankee
RELIABLE ELECTRICITY FOR MAINE SINCE 1972

DRAWING NUMBER FIG. 1.2-1, REV.0

DSAR-APPENDIX (B)
[ISFSI-SAR]



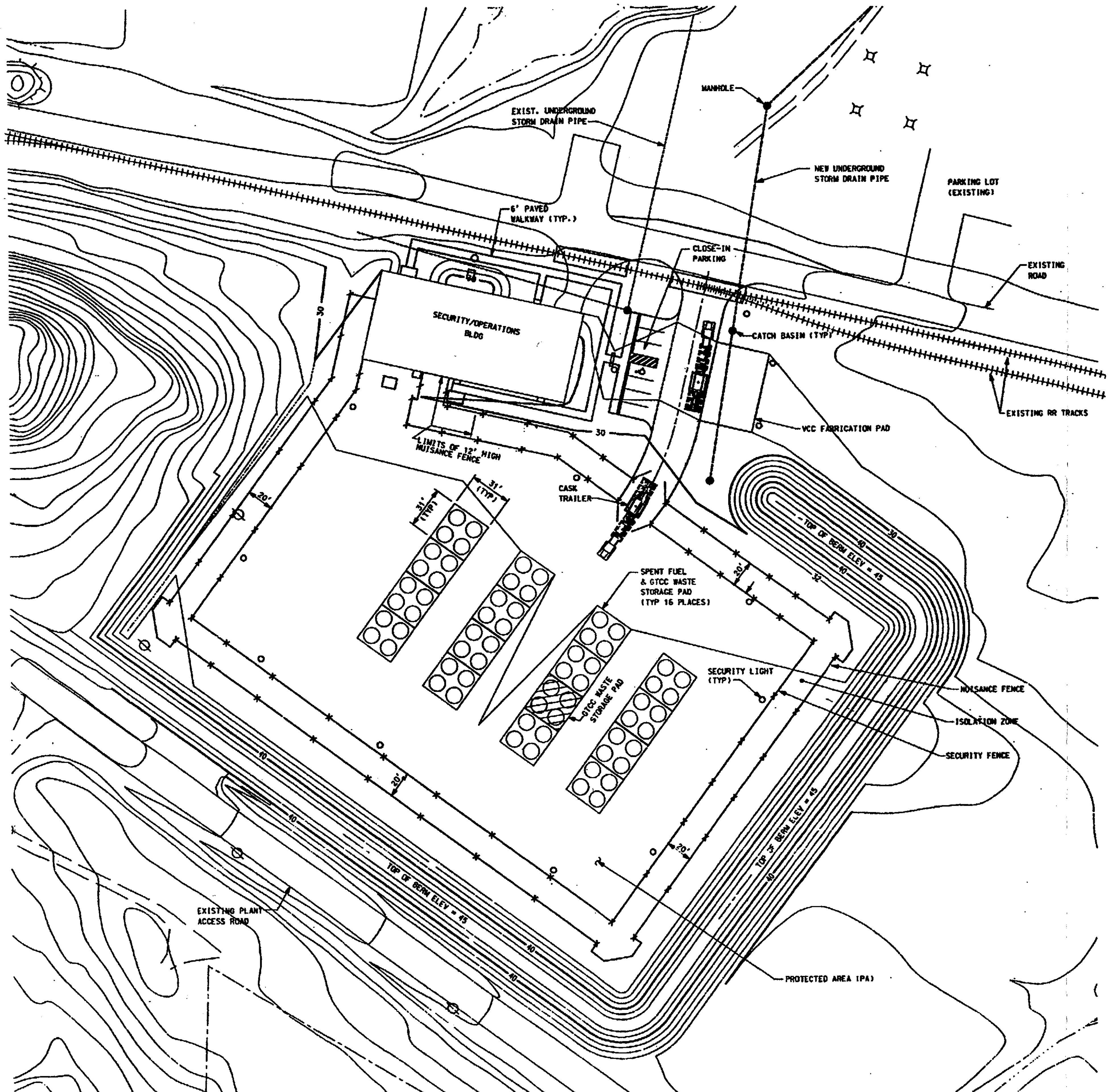
SCALE: 0 50' 100' 150' 200'

ISFSI GENERAL ARRANGEMENT AND HAUL ROUTE

Maine Yankee

DRAWING NUMBER FIG. 1.2-2, REV. 0

REV.	DATE	BY	CHKD	APPD
0	05/23/01	MMM		

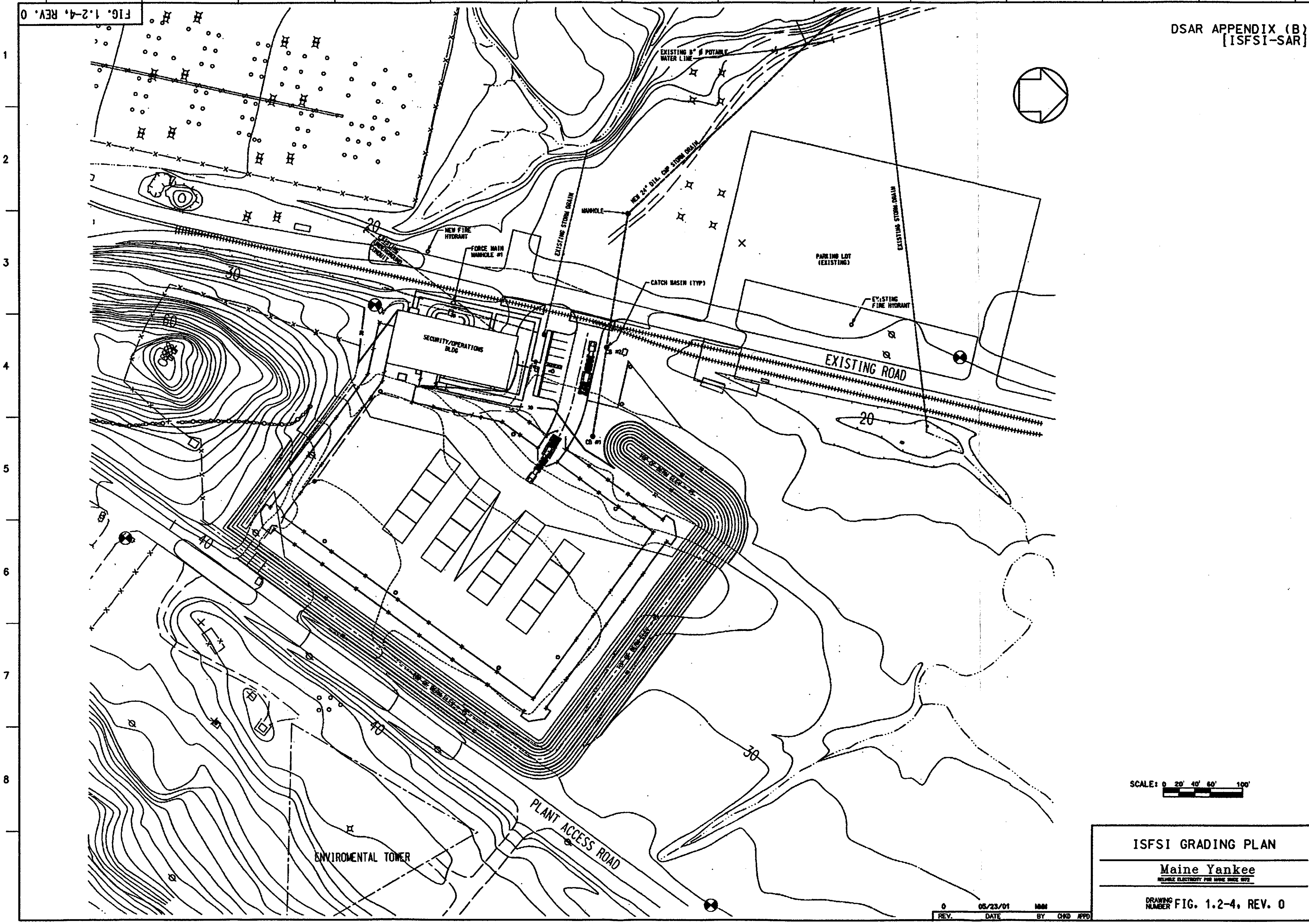


ISFSI SITE PLAN

Maine Yankee

REV.	DATE	BY	CHKD	APPD
0	05/23/01	MMM		

DRAWING NUMBER FIG. 1.2-3, REV. 0

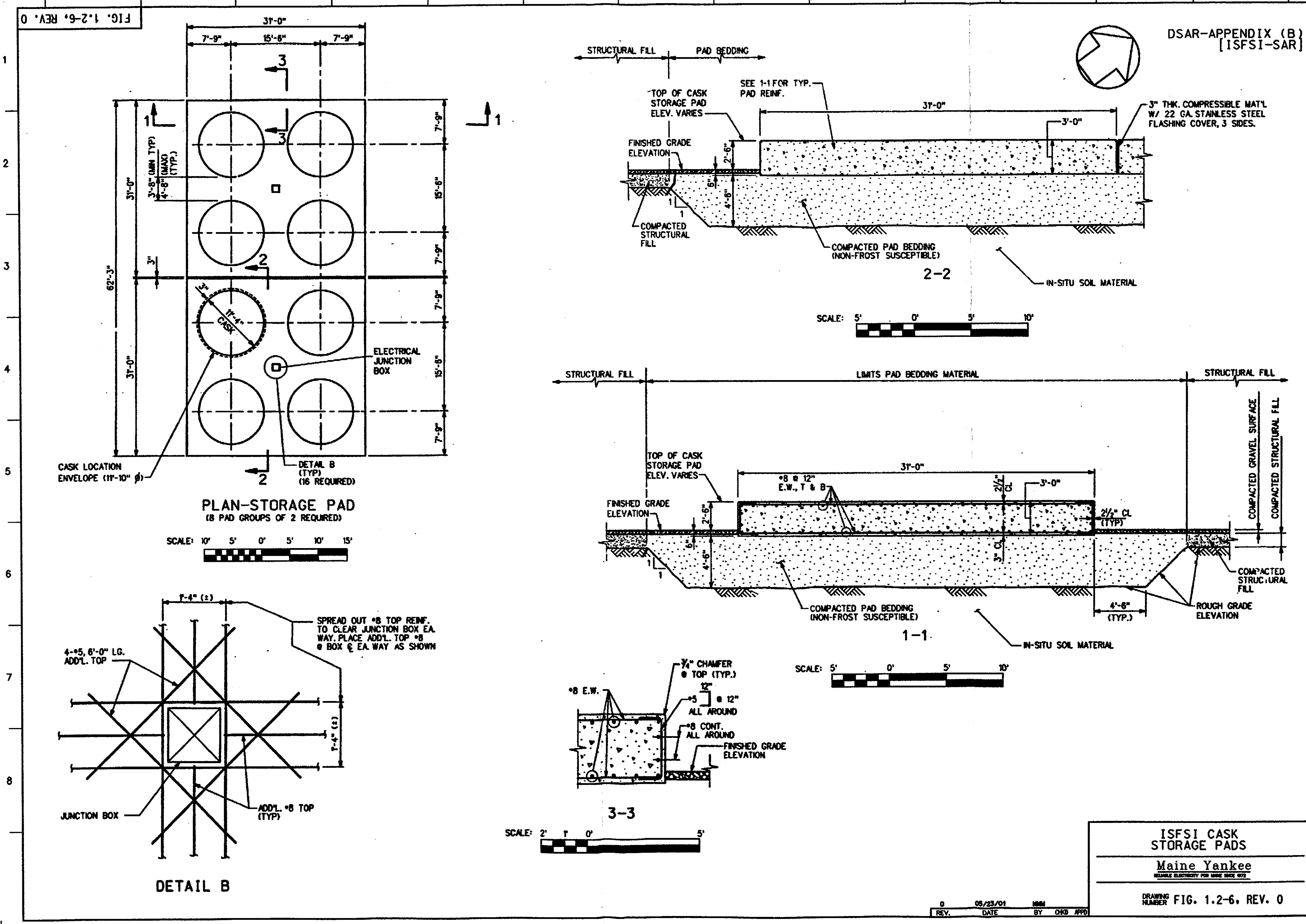


ISFSI GRADING PLAN

Maine Yankee

REV.	DATE	BY	CHKD	APPD
0	05/23/01	MMM		

DRAWING NUMBER FIG. 1.2-4, REV. 0



DSAR-APPENDIX (B)
[ISFSI-SAR]
FIG. 1.2-6, REV. 0
PLAN-STORAGE PAD
(8 PAD GROUPS OF 2 REQUIRED)
31'-0"
7'-9"
15'-6"
62'-3"
3'-8" (MIN TYP)
4'-8" (MAX) (TYP.)
11'-4" CASK
ELECTRICAL JUNCTION BOX
CASK LOCATION ENVELOPE (11'-10" ø)
DETAIL B (TYP) (16 REQUIRED)
SCALE: 10' 5' 0' 5' 10' 15'
DETAIL B
7'-4" (±)
SPREAD OUT #8 TOP REINF. TO CLEAR JUNCTION BOX EA. WAY. PLACE ADD'L. TOP #8 @ BOX ℄ EA. WAY AS SHOWN
4-#5, 6'-0" LG. ADD'L. TOP
JUNCTION BOX
ADD'L. #8 TOP (TYP)
2-2
STRUCTURAL FILL
PAD BEDDING
TOP OF CASK STORAGE PAD ELEV. VARIES
SEE 1-1 FOR TYP. PAD REINF.
3" THK. COMPRESSIBLE MAT'L W/ 22 GA. STAINLESS STEEL FLASHING COVER, 3 SIDES.
3'-0"
FINISHED GRADE ELEVATION
2'-6"
6"
4'-6"
COMPACTED STRUCTURAL FILL
COMPACTED PAD BEDDING (NON-FROST SUSCEPTIBLE)
IN-SITU SOIL MATERIAL
SCALE: 5' 0' 5' 10'
1-1
LIMITS PAD BEDDING MATERIAL
#8 @ 12" E.W., T & B
2½" CL
2½" CL (TYP)
3" CL
4'-6" (TYP.)
COMPACTED GRAVEL SURFACE
COMPACTED STRUCTURAL FILL
ROUGH GRADE ELEVATION
3-3
#8 E.W.
¾" CHAMFER @ TOP (TYP.)
#5 @ 12" ALL AROUND
#8 CONT. ALL AROUND
SCALE: 2' 1' 0' 5'
ISFSI CASK STORAGE PADS
Maine Yankee
DRAWING NUMBER FIG. 1.2-6, REV. 0
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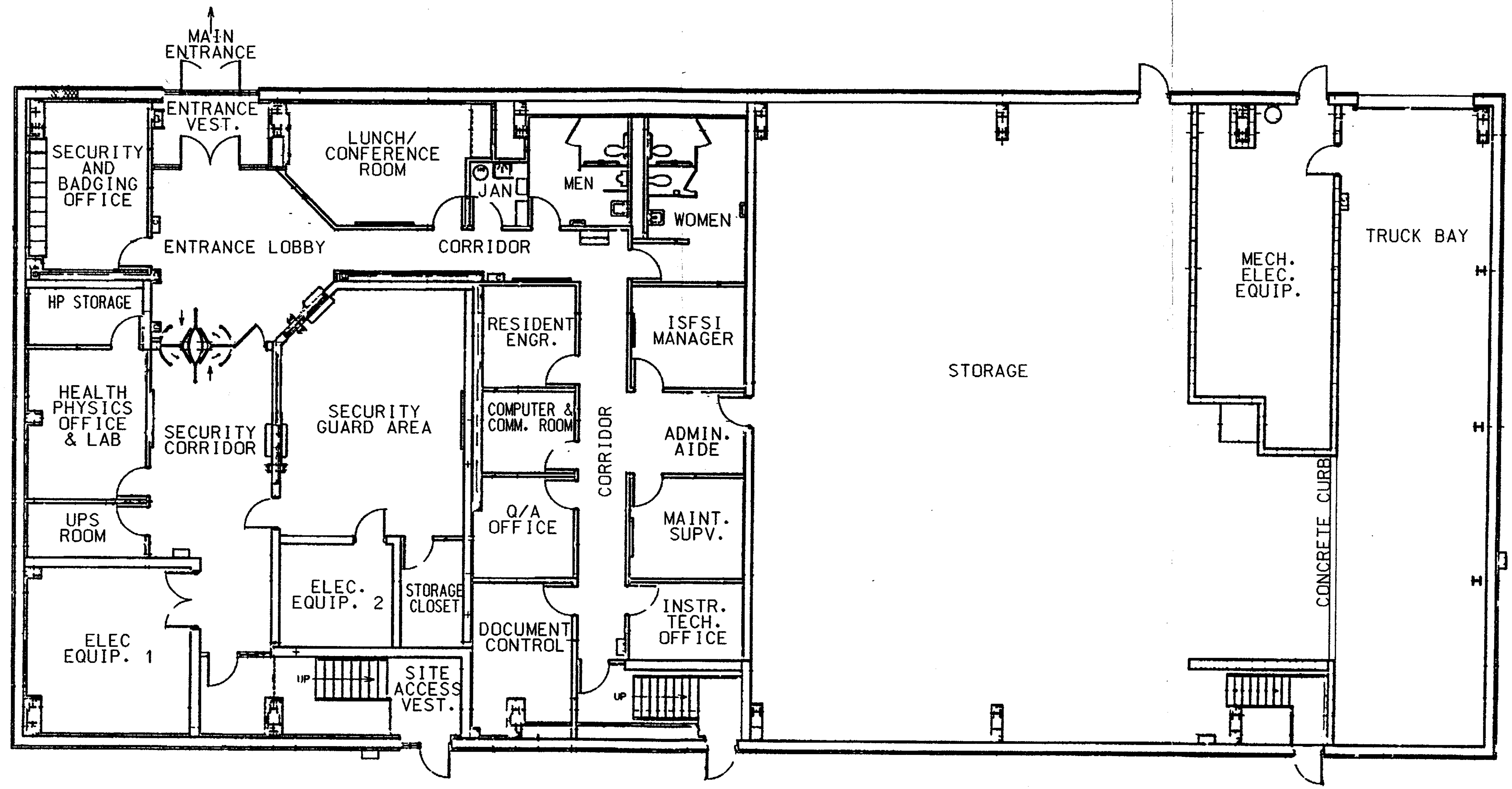
REV. DATE BY CHKD APPD

FIG. 1.2-7, REV. 0

DSAR-APPENDIX (B)
[ISFSI-SAR]



0 2' 4' 6' 8' 10'

GROUND FLOOR PLAN

Finished Floor Elev. 26'-0"

GROUND FLOOR DESIGN LIVE LOAD
-1800 psf in Storage Area
-150 psf in Office/Security Area

CEILING SLAB (EL. 42'-0") DESIGN
-Design Live Load = 15 psf

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SECURITY/OPERATIONS BUILDING
GROUND FLOOR PLAN

Maine Yankee

DRAWING NUMBER FIG. 1.2-7, REV. 0

1.3 General ISFSI System Description

1.3.1 ISFSI Storage System - General

The ISFSI utilizes the Universal MPC System (UMS™) developed by NAC International. The UMS™ is a canister-based multi-purpose canister (MPC) system designed for both storage and transportation of spent nuclear fuel and GTCC waste. The ISFSI will operate under the provisions of a general license utilizing the UMS™ spent fuel storage/transportation system. NRC Certificate of Compliance (CoC) No. 1015 has been issued, approving use of the NAC-UMS™ system for storage of spent nuclear fuel under the general license provisions of 10 CFR 72.210.

The canister-based spent fuel storage system is a simple and passive system which utilizes an outer concrete cylinder called a "storage cask" to protect and shield the inner sealed metal canister. The storage cask is vented for natural convection cooling and has no moving parts.

Metal canisters that will store spent fuel will be loaded in the spent fuel pool, vacuum dried, helium backfilled, and sealed and placed into the storage cask utilizing a metal transfer cask and the115 -ton Fuel Building - Yard Crane. The crane has been upgraded to single-failure-proof design. Canisters storing GTCC waste will be loaded in a similar manner except that they are being loaded in the Containment structure using the 250-ton Polar Crane(CR-1). The loaded concrete storage cask will be hauled to the ISFSI with a custom designed heavy haul tractor-trailer and offloaded onto the concrete storage pad.

1.3.2 ISFSI Universal Storage System Components

The Universal Storage System consists of three principal components

- Transportable Storage Canister (including PWR fuel basket),
- Vertical Concrete Cask, and
- Transfer Cask.

1.3.2.1 Transportable Storage Canister

The Transportable Storage Canister consists of a stainless steel canister that contains the fuel basket structure and contents. The canister is defined as confinement for the spent fuel during storage and is provided with a double welded closure system. The welded closure system prevents the release of contents in any design basis normal, off-normal or accident condition. The basket assembly in the canister provides the structural support and primary heat transfer path for the fuel assemblies while maintaining a subcritical configuration for all normal conditions of storage, off-normal events, and hypothetical accident conditions.

The major components of the Transportable Storage Canister are the shell and bottom, a basket assembly, shield lid, and structural lid. The canister and the shield and structural lids provide a confinement boundary during storage and lifting of the TSC.

The Transportable Storage Canister is designed to the requirements of the ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Division I, Subsection NB (Ref. 21). It is fabricated and assembled in accordance with the requirements of Subsection NB to the maximum extent practicable, consistent with the conditions of use.

1.3.2.2 Fuel Baskets

The transportable storage canister contains a fuel basket which positions and supports the stored fuel in normal, off-normal, and accident conditions. The fuel basket is designed and fabricated to the requirements of the ASME Code, Section III, Division I, Subsection NG (Ref. 22). However, the basket assembly is not Code stamped and no reports relative to Code stamping are prepared. Consequently, an exception is taken to Article NG 8000, Nameplates, Stamping and Reports.

The fuel basket is contained within the transportable storage canister. It is constructed of stainless steel, but incorporates aluminum disks for enhanced heat transfer. The fuel basket design is a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks. The fuel tubes include Boral sheets on all four sides for criticality control.

1.3.2.3 Vertical Concrete Cask

The Vertical Concrete Cask is the storage overpack for the Transportable Storage Canister. The concrete cask provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage.

The concrete cask is a reinforced concrete (Type II Portland cement) structure with a structural steel inner liner. The concrete wall and steel liner provide neutron and gamma radiation shielding for the spent fuel. Inner and outer reinforcing steel (rebar) assemblies are contained within the concrete.

The reinforced concrete wall provides the structural strength to protect the canister and its contents in natural events such as tornado wind loading and wind driven missiles. The concrete cask is shown in Figure 1.3-1.

The Vertical Concrete Cask forms an annular air passage to allow the natural circulation of air around the canister to remove the decay heat from the spent fuel. The air inlets and outlets are steel-lined penetrations that take nonplanar paths to the concrete cask cavity to minimize radiation streaming. The decay heat is transferred from the fuel assemblies to the tubes in the fuel basket and through the heat transfer disks to the canister wall. Heat flows by radiation and convection from the canister wall to the air circulating through the concrete cask annular air passage and is exhausted through the air outlet vents. This passive cooling system is designed to maintain the peak cladding temperature of the Zircaloy-clad fuel well below acceptable limits during long-term storage.

The top of the Vertical Concrete Cask is closed by a shield plug and lid. The shield plug is approximately 5 in. thick and incorporates carbon steel plate as gamma and neutron radiation shielding. A carbon steel lid that provides additional gamma radiation shielding is installed above the shield plug. The shield plug and lid reduce skyshine radiation and provide a cover and seal to protect the canister from the environment and postulated tornado missiles. The lid is bolted in place and has a tamper indicating seal on two of the installation bolts.

1.3.2.4 Transfer Cask

The transfer cask, with its lifting yoke, is primarily a lifting device used to move the canister. It provides biological shielding when it contains a loaded canister. The transfer cask is used for the vertical transfer of the canister between work stations and the concrete cask or a transport cask.

The transfer cask incorporates a multiwall (steel/lead/NS-4-FR/steel) design, which limits the contact radiation dose rate to less than 300 mrems/hr. The transfer cask design also incorporates a top retaining ring, which is bolted in place that prevents a loaded canister from being inadvertently removed through the top of the transfer cask. The transfer cask has retractable bottom shield doors. During loading operations, the doors are closed and secured by pins so they cannot inadvertently open. During unloading, the doors are retracted using hydraulic cylinders to allow the canister to be lowered into a concrete cask for storage or into a transport cask. The transfer cask is shown in Figure 1.3-2.

The transfer cask is a heavy, lifting device. Accordingly, it is designed, fabricated, and load tested to meet the requirements of NUREG-0612 (Ref. 23) and ANSI N14.6, (Ref. 24).

1.3.2.5 Auxiliary Equipment

This section presents a brief description of principal auxiliary equipment needed to operate the Universal Storage System in accordance with its design.

1.3.2.5.1 Adapter Plate

The adapter plate is a carbon steel table that bolts to the top of the Vertical Concrete Cask or the Universal Transport Cask and mates the transfer cask to either of those casks. It has a large center hole that allows the Transportable Storage Canister to be raised or lowered through the plate into or out of the transfer cask. Rails are incorporated in the adapter plate to guide and support the bottom shield doors of the transfer cask when they are in the open position. The adapter plate also supports the hydraulic system and the actuators that open and close the transfer cask bottom doors.

1.3.2.5.2 Heavy Haul Trailer

The heavy-haul trailer is used to move the Vertical Concrete Cask. A special trailer is designed for transport of the empty or loaded concrete cask. The design incorporates a built-in jacking system that facilitates the leveling of the trailer to the storage pad. The trailer incorporates both reinforcing to increase the trailer load-bearing area and design features that reduce its turning radius. However, any commercial double-drop-frame trailer having a deck height approximately matching that of the storage pad could be used.

1.3.2.5.3 Temperature Instrumentation

The Vertical Concrete Cask has four air outlets near the top of the cask and four air inlets at the bottom. Each outlet is equipped with a permanent remote temperature detector mounted in the outlet air plenum to monitor spent fuel in storage. The detector is used to measure the outlet air temperature, which can be read at a junction box located on the outside surface of the concrete cask or at a remote location.

1.3.2.6 Operational Features

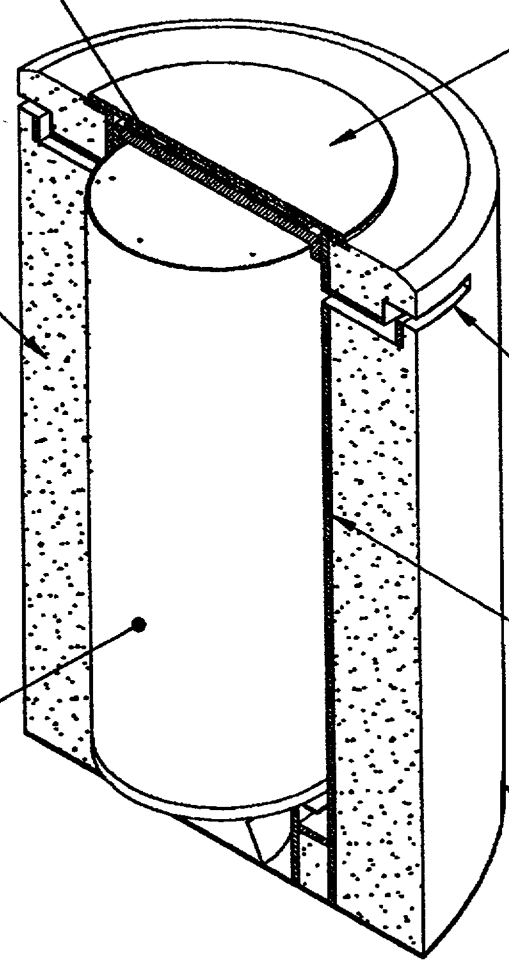
Temperature monitoring of the Vertical Concrete Cask outlet air is the only active system used for monitoring the spent fuel in storage. This temperature is recorded daily as a check of the thermal performance of the concrete casks. This system does not penetrate the confinement boundary and is not essential to the safe operation of the Universal Storage System.

The principal activities associated with the use of the Universal Storage System are loading the canister with spent fuel, closing the canister, and loading the canister in the concrete cask. The transfer cask is designed to meet the requirements of these operations. The transfer cask holds the canister during loading with fuel, provides for canister exterior surface flushing with non contaminated water while in the spent fuel pool, provides biological shielding during closing of the canister, and provides the means by which the loaded canister is moved to and installed in the concrete cask.

The canister consists of four principal components: the canister shell (side wall and bottom), the shield and structural lids, the vent and drain ports (together with the vent and drain port covers), and the basket assembly. A drain tube extends from the shield lid drain port to the bottom of the canister. The vent and drain ports allow the draining, vacuum drying, and backfilling with helium necessary to provide a dry, inert atmosphere for the contents. The vent and drain port covers, the shield lid, the canister shell, and the joining welds form the primary confinement boundary. A secondary confinement boundary is formed over the shield lid by the structural lid and the weld that joins it to the canister shell. The structural lid contains the drilled and tapped holes for attachment of the swivel hoist rings used to lift and lower the loaded canister.

The step-by-step procedures for the operation of the Universal Storage System are presented in Chapter 8.0 of the NAC-SAR.

The auxiliary equipment needed to operate the Universal Storage System is described in Section 1.3.2.5. Other items required include miscellaneous hardware, connection hoses and fittings, and hand tools typically found at a reactor site.



VERTICAL CONCRETE CASK

Maine Yankee
RELIABLE ELECTRICITY FOR MAINE SINCE 1972

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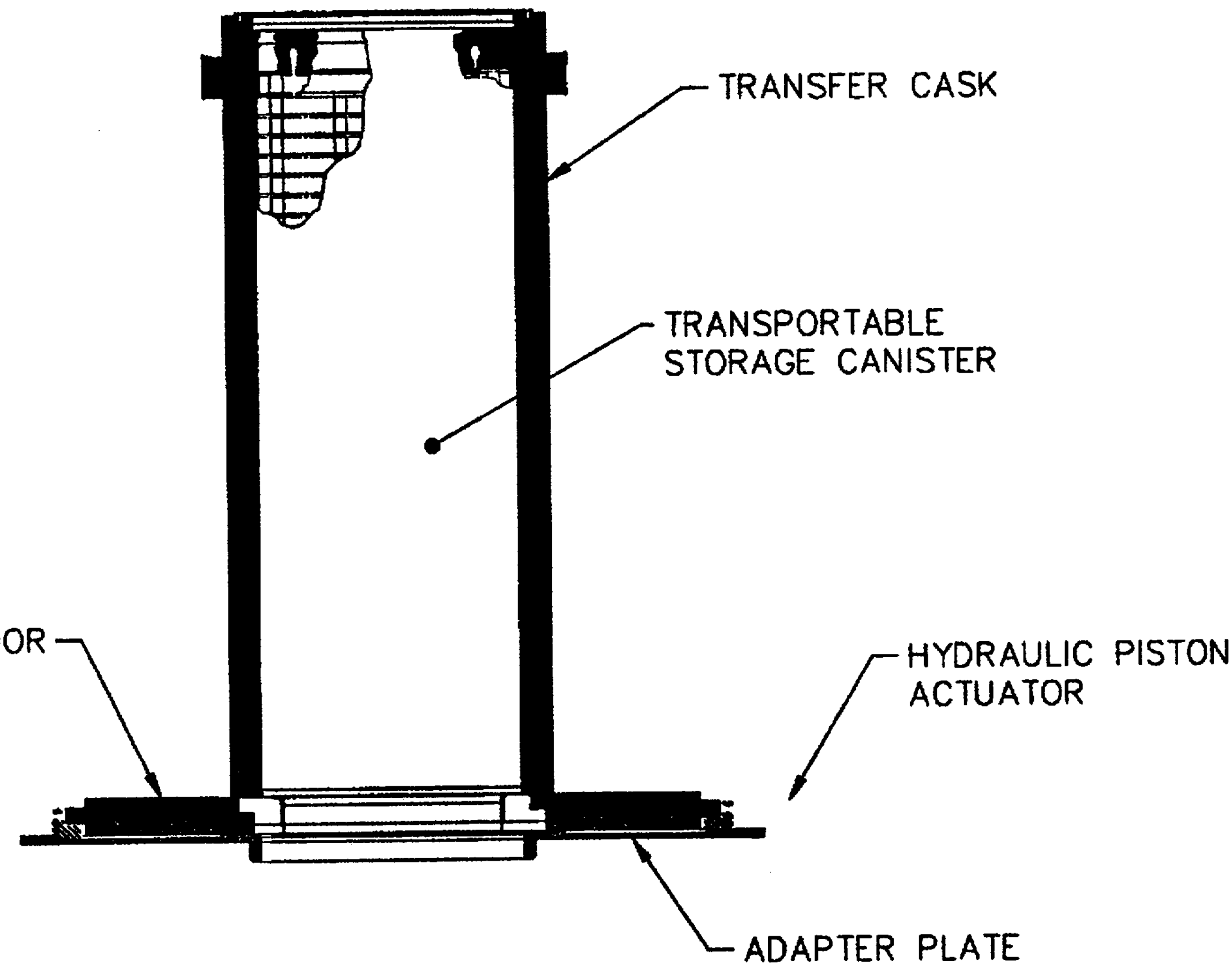

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TRANSFER CASK

Maine Yankee
RELIABLE ELECTRICITY FOR MAINE SINCE 1972

DRAWING NUMBER FIG 1.3-2, REV 0

1.4 Identification of Agents and Contractors

The prime contractor for the design of the ISFSI facility is Stone & Webster Engineering Corporation. In this capacity, Stone & Webster has provided the design of the ISFSI site, facility berm, cask storage pads, Security/Operations Building, and associated support systems. The prime contractor for the design and licensing of the canister-based Universal MPC System (UMS^{TM}) is NAC International Inc.

1.5 Material Incorporated by Reference

1. Safety Analysis Report for the UMS™ Universal Storage System, #EA790-SAR-002, Docket No. 72-1015.

2. Those items listed in the Reference Section for each ISFSI SAR Section.

1.6 References

1. Maine Yankee Letter to NRC dated August 7, 1997, "Certification of Permanent Cessation of Power Operation and That Fuel Has Been Permanently Removed from the Reactor."

2. 10CFR50, "Domestic Licensing of Production and Utilization Facilities."

3. Site Location of Development Permit #L-17973 Application for Amendment, dated May 5, 1999.

4. Maine Yankee Land Use Report by Robert G. Gerber, dated February. 6, 1997.

5. ASCE - 7, "Minimum Design Loads for Buildings and Other Structures," 1995, American Society of Civil Engineers.

6. NUREG - 1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," March, 2000.

7. ANSI/ANS - 57.9 - 1984, "Design Criteria for an Independent Spent Fuel Storage Installation."

8. ACI - 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 1997, American Concrete Institute.

9. BOCA National Building Code (NBC) - 1996, published by the Building Officials & Code Administrators International, Inc.

10. NAC - SAR, "NAC Safety Analysis Report for the UMS^R Universal Storage System," # EA790-SAR-002, Docket No. 72-1015.

11. ACI - 318-95, "Building Code Requirements for Structural Concrete."

12. ASCE - 4, "Seismic Analysis of Safety Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures," 1986, American Society of civil Engineers.

13. NFPA - 101, "Life Safety Code," National Fire Protection Association, 1996.

14. NUREG - 0508, "Design Methodology for the Physical Protection Upgrade Rule Requirements for Fixed Sites," June 1980.

15. NFPA - 2001, "Clean Agent Fire Extinguishing Systems," 1998.

16. NFPA - 10, "Portable Fire Extinguishers," National Fire Protection Association, 1994.

17. UL -299, "Dry Chemical Fire Extinguishers," 1990.

18. UL - 154, "Carbon-Dioxide Fire Extinguishers," 1990.

19. IEEE - 692, "IEEE Standard Criteria for Security Systems for Nuclear Power Generating Stations," 1986.

20. 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

21. ASME Boiler and Pressure Vessel Code, Division I, Section III, Subsection NB, "Class I Components," 1995 Edition with 1995 Addenda.

22. ASME Boiler and Pressure Vessel Code, Division I, Section III, Subsection NG, "Core Support Structures," 1995 Edition with 1995 Addenda.

23. Nuclear Regulatory commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980.

24. ANSI N14.6-1993, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping containers Weighing 10,000 Pounds (4,500 kg) or More," American National Standards Institute, Inc., June 1993, 1993.

25. USNRC Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," April 1974.

26. USNRC, “Standard Review Plan,” NUREG - 0800, April 1996.

27. USNRC Letter (NMY 87-48), Patrick M. Sears (Project Manager) to J. B. Randazza, dated March 26, 1987.

28. Stone & Webster Calculation No. 08196.16 -SG-7, “Seismic Analysis of Cask Storage Pads - NUREG-0098 Earthquake.”

1.7 LIST OF FIGURES